3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The U.S. EPR systems, components, and equipment retain their structural and functional integrity when subjected to dynamic loads that can occur during normal operation, plant transients, and external events, such as earthquakes. This is confirmed through analyses and startup testing, which verify that the systems, components, and equipment meet the regulatory requirements for the dynamic loads that are postulated to occur during both normal plant operations and transient conditions. This section describes the general startup functional tests and the vibration and dynamic analyses to be performed on specified high-energy and moderate-energy piping and the associated piping supports and restraints, and on reactor internals to verify they meet structural and functional requirements. Section 14.2 contains test abstracts that describe in general terms the planned tests that will be performed on the U.S. EPR. Section 14.2 also describes the programmatic controls that will be used to develop the individual U.S. EPR startup tests. The individual startup tests will contain review and acceptance criteria imposed by the detailed design.

The tests, inspections, and analyses described in this section comply with the following regulations:

- GDC 1 and 10 CFR 50.55a require that systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions performed. In addition, 10 CFR 50, Appendix B addresses the issue of QA as it applies to the dynamic resting and analysis of systems, structures, and components (SSC). The NRC has approved the U.S. EPR Quality Assurance (QA) Program (refer to Section 17.5). The vibration, thermal expansion, and dynamic effects test programs for startup functional testing of high-energy and moderate-energy piping and their supports and restraints described in this section comply with this approved QA program, as described in Section 3.1. Results obtained from flow-induced vibration analyses of the reactor pressure vessel (RPV) internals are confirmed by prototype testing. Dynamic analyses methods are described in this section for Seismic Category 1 systems, components, equipment, and their supports (including supports for conduit and cable trays and ventilation ducts).

- GDC 2 requires that systems and components important to safety be designed to withstand the effects of expected natural phenomena, combined with effects of normal and accident conditions, without losing the ability to perform their safety functions. In addition, 10 CFR 50, Appendix S requires systems and components important to safety withstand certain vibratory ground motions associated with design basis earthquakes. This section describes vibration testing programs for safety-related systems and components and presents dynamic analysis methods for Seismic Category I systems, components, equipment, and their supports. These tests, analyses, and comparisons are in accordance with sound engineering practices and demonstrate that these systems and components are designed to withstand natural phenomena in combination with normal and accident conditions.

- GDC 4 requires that the nuclear power plant systems and components important to safety be designed to accommodate the effects of and be compatible with the environmental conditions of normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). The test programs described herein and in Section 14.2 verify the ability of components and systems to withstand the temperatures, pressures, vibrations, and thermal expansions associated with normal plant operation and maintenance as well as the transient conditions arising from anticipated operational events, such as valve and pump actuations. Testing and analysis to confirm that the safety-related systems and components will withstand anticipated loads are described in this section, Section 3.6.2, Section 3.9.3, and in Appendix 3C.

- GDC 14 requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. Dynamic testing of RCPB components is performed to demonstrate that they will withstand the applicable design-basis seismic and dynamic loads, in combination with other environmental and natural phenomena loads, without leakage, rapidly propagating failure, or gross rupture.

- GDC 15 requires that the reactor coolant system (RCS) be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. The RCPB is designed to resist seismic, LOCA, and other appropriate environmental loads individually and in combination. Dynamic analyses are described to confirm the structural design adequacy of the RCPB. Vibration, thermal expansion, and dynamic effects testing are also described to verify the design.

The requirements of GDCs 1, 2, 4, 14, and 15 are also satisfied through vibration, thermal expansion, and dynamic effects testing conducted during startup functional testing for high- and moderate-energy piping and their supports and restraints. The purposes of these tests are to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service and to confirm that no unacceptable restraint of normal thermal motion occurs.

Other FSAR sections that interface with this section are:

- Computer programs used in the analyses addressed in this section are described in Section 3.9.1 and Appendix 3C.

- The design of ASME Code Classes 1, 2, and 3 components, component supports, and core support structures is described in Section 3.9.3.

- The design of reactor vessel internal components is described in Section 3.9.5.

- The seismic qualification testing of Seismic Category I mechanical equipment is described in Section 3.10.

- The thermal and hydraulic design of the RPV internals is described in Section 4.4.

- Consideration of design loads from the dynamic effects of pipe rupture is described in Section 3.6.2 and Appendix 3C.

- The number of earthquake cycles to be considered in Seismic Category I subsystem and component design, as well as the seismic system analysis, is described in Sections 3.7.2 and 3.7.3.

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

Pipes are subject to damage from a variety of events that can cause movement of the pipe or of ancillary components, such as supports or snubbers. These include routine operational events, such as vibrations caused by fluid flow or movement arising from differential thermal expansion. Transient operational events (e.g., valve closures and flow instabilities) can also induce pipe vibration or movement.

This section addresses the preoperational testing, initial fuel loading, and pre-critical testing to be performed to verify that: vibration and thermal expansion and contraction of piping systems meet design requirements; design of the piping systems tested will prevent excessive vibration; and snubbers, restraints, and supports function as intended during these events. The piping systems tested include:

- ASME Code, Section III, Class 1, 2, and 3 piping systems (Reference 1). Table 3.2.2-1—Classification Summary identifies the ASME Code, Section III, Class 1, 2, and 3 systems for the U.S. EPR. Analysis of the vibration of Class 1, 2, and 3 piping arising from seismic events is described in Sections 3.12 and Appendix 3C.

- High-energy piping systems inside Seismic Category I structures, or those whose failure would reduce the safety level of a Seismic Category I SSC (see Table 3.2.2-1).

- Seismic Category I portions of moderate-energy piping systems located outside of containment (see Table 3.2.2-1).

Piping systems are validated through a series of checks, inspections, and tests:

- During construction, pipe and equipment supports are checked for correct assembly and their initial positions under cold conditions are recorded. This phase also includes the initial operation of many system components as the systems are flushed to remove construction debris and meet cleanliness requirements.

- When the plant is first heated to normal operating temperatures the systems are operated in an integrated mode. Expansion data for piping systems are recorded, compared with the cold position data, and evaluated against system expansion acceptance criteria. Piping systems are checked for expansion-induced damage and for improperly restrained expansion and contraction. Clearances at gapped restraints are monitored during thermal expansion and contraction of the tested systems to identify any unanticipated contact between the fluid systems and their restraints. Snubber and deadweight support spring travel are also monitored to confirm that they will prevent unanticipated loading of the system.

- Performance tests are run on systems to verify operation and to check the performance of critical pumps, valves, controls, and auxiliary equipment. This phase of testing includes transient tests to identify unacceptable movement, noise, vibration, and damage caused by rapid valve opening and closing, safety valve discharge, pump operation, and other operational transients. During this phase, the piping and piping restraints are observed for vibration and expansion response and the automatic safety devices, control devices, and other major equipment are observed for indications of overstress, excessive vibration, overheating, and noise. Each system test includes critical valve operation during anticipated transients.

The initial test program is described in Section 14.2. The vibration, thermal expansion, and dynamic effects elements of this test program, summarized below, are performed during Phase I, Preoperational Testing, and Phase II, Initial Fuel Loading and Pre-Critical Testing. The initial test program includes the selected flow modes of operation and the transients to be simulated during testing, per RG 1.68. Test procedures for the initial test program describe the methods for establishing and measuring the reference values, including information on the instrumentation. Specific information concerning the locations where visual inspection or measurements are to be taken is also addressed in the applicable test procedures. These procedures also include details on the pipe monitoring displacement transducers or scratch plates, and strain gage or load cell locations. Information accuracy, measurement range requirements, and the criteria for evaluating the data are also addressed in the test procedures.

Phase I – Preoperational Testing

The piping systems to be tested are those described above (i.e., ASME Class 1, 2, and 3 pipes; high-energy pipes that involve seismic Category I structures; and seismic Category I portions of moderate-energy piping systems located outside containment). This testing includes ASME instrumentation lines up to the first support in each of the three orthogonal directions from the process pipe or equipment connection point.

Preoperational tests are intended to demonstrate that the components comprising these piping systems meet functional design requirements, that piping vibrations are within acceptable levels, and that proper allowance for thermal contraction and expansion is provided. In addition, these tests verify that integrated systems are operating correctly, that system operating procedures are correct, and that system components and safety equipment are operational prior to fuel loading. The end of hot functional testing (HFT) marks the end of Phase I testing. Before fuel loading commences, the results of preoperational tests are evaluated by plant operations and technical staff. If test acceptance criteria are not satisfied, appropriate corrective actions and retesting occurs.

Phase II – Initial Fuel Loading and Pre-Critical Testing

Initial fuel loading and pre-critical tests (refer to Section 14.2.12) are similar to Phase I tests, but occur after the initial reactor core is loaded. Phase II tests establish that the RCS vibration levels and piping reactions to transient conditions (e.g., pump starting and stopping and valve opening and closing) are acceptable. Phase II testing is completed, evaluated, and any required corrective actions taken prior to initiating Phase III (Initial Criticality and Low-Power Physics Testing). If excessive vibration levels are detected during testing, consideration is given to modifying the design specification to re-verify applicable code conformance using the measured vibration as input. If testing and subsequent analysis reveal that additional restraints are needed to reduce stresses to acceptable levels, they are installed.

As described in U.S. EPR Piping Analysis and Pipe Support Design (Reference 2), the U.S. EPR uses snubbers to support piping systems that require free thermal movement but restrained movement due to dynamic loadings. The proper installation and operation of snubbers is verified through visual inspections, hot and cold position measurements, and observation of thermal movements during Phase I and II startup testing. Section 3.9.6.1 provides the preservice testing (PST) and inservice testing (IST) requirements for snubbers. The IST program incorporates Phase I and II startup testing. Snubber use and locations are determined during detailed design using the analytical methods presented in Reference 2, as described in Section 3.9.6.

3.9.2.1.1 Piping Vibration Details

Piping vibration testing and assessment is performed in accordance with the ASME Standards and Guides for Operation and Maintenance of Nuclear Power Plants (Reference 3) including the addenda. Reference 2 describes the code requirements, acceptance criteria, analysis methods, and modeling techniques for ASME Class 1, 2, and 3 piping and pipe supports. The Phase I and II tests described above demonstrate that the piping systems withstand vibrations arising from Level A (Normal) loads and Level B (Upset) loads.

Level A loads are sustained loads encountered during normal plant startup, operation, refueling, and shutdown. The vibrations arising from such loads are considered steady-state or constant vibration. If excessive system vibration is evident during Phase I or Phase II tests arising from Level A loads, 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 vibration does not cause the piping in question to exceed stress or fatigue acceptance criteria.

Level B loads are infrequent loads with a high probability of occurrence but which cause no damage or reduction in function. The vibrations arising from these loads are transient and are usually set in motion by rapid actuation of control, relief, and check valves, or from the rapid start or trip of a pump or turbine. These dynamic responses are tested or simulated during the preoperational testing. If excessive system vibration is evident during Phase I or Phase II tests arising from Level B loads, 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 vibration does not cause the piping in question to exceed stress or fatigue acceptance criteria.

The systems and transients included in the test program are described in Section 3.9.1 and Section 14.2. The Phase I and Phase II tests do not address vibrations arising from Level C (Emergency) or Level D (Faulted) loads.

3.9.2.1.2 Piping Thermal Expansion Details

Thermal expansion testing verifies that the design of the piping systems tested prevents constrained thermal contraction and expansion during Normal and Upset transient events, and that the component supports (including spring hangers, snubbers, and struts) can accommodate the expansion of the piping for these modes of operation. In those systems where startup testing leads to unacceptable thermal expansion, the systems or their restraints are modified and retested.

Section 14.2.12 provides descriptions of selected planned piping thermal expansion measurement tests. Test specifications for thermal expansion testing of piping systems during preoperational and startup testing are in accordance with Reference 3. The standard also provides guidance for developing acceptance criteria, instrumentation, and measurement techniques, as well as corrective actions and methodologies for reconciling movements that differ from those specified by the acceptance criteria.

3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

This section describes the seismic system analysis and qualification of Seismic Category I systems, components, and equipment performed to confirm functional integrity and operability during and after a postulated seismic occurrence. Appendix 3A describes the design criteria for piping and supports (including cable trays and ventilation ducts). Additionally, Section 3.2 provides a list of Seismic Category I equipment.

3.9.2.2.1 Seismic Qualification Testing

The methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment and a description of the seismic operability criteria are described in Section 3.10.

3.9.2.2.2 Seismic Sub-System Analysis Methods

Descriptions of the U.S. EPR seismic analysis methods for safety-related piping are provided in Reference 2 and AREVA NP letter NRC:07:028 dated July 13, 2007 (Reference 4). Methods for seismic analysis of systems, components, and equipment are also addressed in Section 3.7.3, Appendix 3A, and Appendix 3C.

3.9.2.2.3 Determination of Number of Earthquake Cycles

See Section 3.7.3.

3.9.2.2.4 Basis for Selection of Frequencies

See Section 3.7.3.

3.9.2.2.5 Three Components of Earthquake Motion

See Section 3.7.3, Appendix 3C, and Section 4.2 of Reference 2.

3.9.2.2.6 Combination of Modal Responses

See Section 4.2.2.3 of Reference 2.

3.9.2.2.7 Analytical Procedures for Piping

See Appendix 3C and Section 4.2 of Reference 2.

3.9.2.2.8 Multiple-Supported Equipment Components with Distinct Inputs

See Section 4.2.2.2 of Reference 2.

3.9.2.2.9 Use of Constant Vertical Static Factors

See Section 3.7.3.

3.9.2.2.10 Torsional Effects of Eccentric Masses

See Section 5.2 of Reference 2.

3.9.2.2.11 Buried Seismic Category I Piping Conduits, and Tunnels

See Section 3.7.3 and Section 3.10 of Reference 2.

3.9.2.2.12 Interaction of Other Piping with Seismic Category I Piping

See Section 4.4 of Reference 2.

3.9.2.2.13 Analysis Procedure for Damping

See Section 3.7.3 and Section 4.2.5 of Reference 2.

3.9.2.2.14 Test and Analysis Results

See Sections 3.9.2.2.1 and 3.9.2.2.2 above.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

Vibration characteristics and behavior due to flow-induced excitation are complex and not readily determined by analytical means. Thus, the assessment of the vibrational response of the U.S. EPR RPV internals includes a combination of analytical evaluations and testing. Scale model analyses confirmed by scale model tests are used to verify the analytical methods and design inputs that are used for the full-scale design analysis. During preoperational testing, the full-scale analytical results are confirmed, or the analysis inputs are adjusted to achieve agreement and understanding of the response of the RPV internals to flow-induced excitation mechanisms. The results of this comprehensive vibration assessment program are recorded, consistent with the guidance of RG 1.20.

The design of the U.S. EPR RPV internals is similar to that of the AREVA NP SAS 4-loop N4 units and the German Konvoï units; however, the U.S. EPR design also incorporates international design evolutions. AREVA NP designates the classification of the RPV internals for the Olkiluoto-3 reactor, which is the first EPR to be constructed, as prototype according to the RG 1.20 classification guidance. The U.S. EPR internals are classified as prototype per the guidance of RG 1.20. Subsequent to HFT and inspection of the Olkiluoto-3 RPV internals and successfully fulfilling other RG 1.20 requirements for a prototype design, the U.S. EPR internals may be re-classified as non-prototype Category I.

Reactor components are excited by flowing coolant, which causes fluctuating pressures on their surfaces. The integration of these pressures over the applied area provides the forcing functions used in the dynamic analysis of the structures. Due to the complexities of the geometries and the random character of the pressure fluctuations, a closed form solution of the vibration problem by the integration of the differential equations is not always practical or realistic.

The determination of forcing functions as a direct correlation of pressure fluctuations cannot be practically performed independent of the dynamic characteristics of the RPV internals. The objective is to establish the characteristics of the forcing functions that determine the response of the various structures that make up the internals. By analyzing the dynamic properties of the structures obtained from analysis and an experimental scale model flow test, the characteristics of the forcing functions are deduced to provide verification of the design inputs for the full-scale design. These analyses indicate that the important forcing functions are flow turbulence and pump-related excitation. The relevance of such excitation depends on factors that include the type and location of components and the specific flow conditions.

RPV internal vibrations are induced by flow turbulences, the fluid–structure interactions (e.g., vortex shedding and leakage-flow-induced vibrations), and acoustic sources. The high-frequency acoustic sources from the reactor coolant pumps (rotation frequency and blade passing frequency) and the low-frequency acoustic sources from loop oscillation can induce vibrations of the RPV internals. The RPV internals are evaluated for the primary flow turbulence and fluid-structure interaction vibration mechanisms consistent with full-power, steady-state normal operating conditions with the reactor coolant pumps operating.

Since the spectral density of the pressure fluctuations decreases rapidly as frequency increases, the lower RPV internals vibrations are primarily due to flow turbulences in the downcomer that induce random, low-frequency excitation (typically from 0 to 30 Hz). The low-frequency loop oscillations are expected to have an influence on the response of the lower internals.

The design of the lower RPV internals does not include baffle and former plates; these have been replaced with the heavy reflector. As such, the stiffness of the lower RPV internals has been reduced and its weight has increased, reducing the beam and shell mode natural frequencies. The RPV lower internals are monitored during preoperational testing to verify that the shell and beam mode frequencies and amplitudes agree with the analytical solutions of these parameters.

For the upper RPV internals, including the support columns in the upper plenum, vortex shedding and fluid-elastic instability are potential sources of strong vibrations. Support column designs similar to those in the U.S. EPR have proven operating experience in the German Konvoï plants. The incore instrumentation guide tubes attached to the column supports in the upper internals are somewhat shielded from the coolant flow in the upper plenum, but nonetheless are evaluated for flow-induced vibrations (FIV). The instrumentation guide tubes are subjected to fuel assembly outlet nozzle turbulence, which is judged to be less than the inlet nozzle turbulence to which incore instrumentation thimbles in previous plants are subjected.

The FIV performance and the design of the flow distribution device and the control rod guide assemblies (CRGAs) have been optimized based on flow tests of these components. The results of these tests are used to optimize the design of these components to minimize vibration levels and the formation of vortices. These test results are also used to optimize the design of the rod control cluster assemblies, which function within the confines of the control rod guide assembly column supports, to minimize wear. Analytical evaluations of these full-scale components have demonstrated that they have acceptable vibrational behavior.

The vibration assessment program demonstrates that the vibration levels of the RPV internals conform to RG 1.20. The U.S. EPR is equipped with a vibration monitoring system (VMS), which provides information on the vibratory behavior of the RPV internals during operation.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Currently, the U.S. EPR RPV internals are classified as a prototype per RG 1.20. AREVA NP plans to rely on the hot functional testing and subsequent inspection of the Olkiluoto-3 internals to reclassify the U.S. EPR internals as non-prototype Category I, provided the other guidelines of RG 1.20 are fulfilled for the Olkiluoto-3 prototype design. The Olkiluoto-3 plant is currently under construction and is due to undergo hot functional testing prior to start of safety-related construction for the U.S. EPR, and qualifies as a valid prototype for the U.S. EPR RPV internals per RG 1.20. If design changes to the RPV internals are required as a result of the hot functional testing and subsequent inspection at Olkiluoto-3, the appropriate classification of the U.S EPR RPV internals will be determined in accordance with RG 1.20.

Non-prototype Category I designation is applicable to the U.S. EPR RPV internals because the RPV internals for Olkiluoto-3 are a similar design and are subjected to similar flow and thermal hydraulic conditions. The U.S. EPR RPV internals are analyzed for the effects of flow-induced vibration and are not expected to be subject to unacceptable flow induced vibrations. The analysis of the U.S. EPR RPV internals is described in Section 3.9.2.3. Either extensive measurements or a complete inspection program of the U.S. EPR RPV internals will be performed during hot functional testing in accordance to the guidelines delineated in RG 1.20 for the non-prototype Category I designation. A COL applicant that references the U.S. EPR design certification will submit the results from the vibration assessment program for the U.S. EPR RPV internals, in accordance with RG 1.20.

The comprehensive vibration assessment program confirms acceptable long-term, steady-state vibration response of the RPV internals for operating conditions. This program includes:

- A prediction of the vibrations of the RPV internals.
- A preoperational vibration test program of the internals of the reference plant.
- A correlation of the analysis and test results.

During the hot functional testing, the internals are subjected to a total operating time at normal full-flow conditions (or higher flow conditions) of at least 240 hours, including a cyclic loading of greater than 10^6 cycles on the structural elements of the RPV internals. The objective of these tests is to characterize the RPV internals vibratory behavior in steady-state conditions and in expected transient conditions. Considering that the vibratory behavior of the RPV internals is mainly a function of the flows, and large-scale temperature transients could influence the vibratory response, the relevant transients are:

- Plant heatup and cooldown.
- Reactor coolant pumps start or stop (one or more pumps operating simultaneously).

The plant heatup and cooldown transients are slow and can therefore be considered as a succession of steady-state conditions. Consequently, the relevant steady-state operating conditions are:

- Operation with all the reactor coolant pumps at several temperature levels from cold to hot conditions.
- Operation with at least one reactor coolant pump unavailable in cold and hot conditions.

Various conditions are tested in order to cover the potential situations for flow-induced vibration. The instrumentation is designed and installed to measure the vibration of the internals during hot functional testing. The instrumentation includes devices that are attached to the reactor vessel internals to measure component strains, displacements, accelerations, and dynamic pressures at those locations where the analytical predictions show the largest responses.

The field tests described below provide measurements of the vibratory response of the RPV internals for various primary flow rate conditions. The tests have the following objectives:

- Provide information that enables a direct assessment of the RPV internals vibratory behavior.
- Provide reference data to adjust the overall model and for the detailed assessment of the vibratory stresses.
- Provide reference data for the VMS calibration.

The field tests are focused on measuring the vibratory response to the modes likely to be excited by the flow turbulences, fluid-structure interactions, and acoustic sources (i.e., the low frequency modes) for which analytical predictions of the RPV responses have been determined. In addition, the field tests characterize the modes likely to be detected by the VMS.

Inspections before and after the hot functional test confirm that the RPV internals are functioning correctly. When no indications of harmful vibrations or signs of abnormal wear are detected, and no apparent structural changes are observed, the RPV internals are considered structurally adequate. If such indications are detected, further evaluation is required.

The testing and visual inspection plan to be used for the prototype RPV internals at Olkiluoto-3 involves visual inspections before and after the preoperational tests of the internals. These visual examinations are concerned with the accessible areas of the internals, and in particular the fastening devices, the bearings surfaces, the interfaces between the RPV internal parts that are likely to experience relative motions, and the inside of the RPV. Inspections of the lower and upper RPV internals are described in Tables 3.9.2-1—Visual Inspection of the Accessible Areas of the Upper Internals While on the Storage Stand, through 3.9.2-5—Visual Inspection of the Inside of the RPV Head While on the Storage Stand.

The RPV internals flow-induced vibration measurement program is conducted during preoperational tests of the Olkiluoto-3 and U.S. EPR reactors. The U.S. EPR RPV internals testing and inspection programs conform to RG 1.20.

RG 1.20, Revision 3, recommends that the potential adverse effects from pressure fluctuations and vibrations in piping systems should be considered for the steam generator (SG) internals for both PWRs and BWRs. The U.S. EPR SG upper internals (e.g., steam dryers, separators) are subject to secondary side steam flow. Although there are instances of these components in BWR plant designs experiencing excessive vibration resulting from plant power uprate, to date none have been reported for PWR SG designs both internationally or within the United States. This is further supported by a review of the INPO steam generator operating experience database which also does not have any events related to vibration problems for PWR SG upper internals. In response to public comments on the proposed revisions to RG 1.20 (i.e., DG-1163), the NRC states: “In addition to BWR plants, the pressurized-water reactor (PWR) at the Palo Verde plant experienced degradation from excess vibration that had characteristics similar to those of the phenomenon affecting the BWR plants.” However, AREVA NP understands that the excessive vibrations associated with the shut down coolant pipe at the Palo Verde plant did not lead to vibration problems with the SG upper internals.

The design of the U.S. EPR SG upper internals and the flow conditions for which they are subjected are similar to the existing and currently operating SGs in the United States and Europe. Based on operational experience, AREVA NP concludes that these non-safety-related components will not experience excessive vibration. Therefore, no flow-induced vibration analyses or startup testing is currently planned for these components.

The vibration of representative trains of piping attached to the RCS as well as main steam and main feedwater lines are measured during initial startup testing. These measurements will be taken at discrete piping locations and also at the other key components (e.g., valves and pumps) installed along the length of pipe. Accelerations will be measured using hand-held devices for both steady-state and transient flow conditions.

In the 1970s and 1980s, the above process was employed to address vibration concerns during startup testing for the current operating plants. Since there are very few instances of excessive pipe vibration while operating these plants at their design power level, this is a proven and reliable method of validation. The majority of the cases in which excessive pipe vibration have been observed have been at the stand off branch lines in the main steam piping system when the plant uprates to a higher power level. Therefore, greater scrutiny of these piping configurations will be stressed during the initial startup testing using the method outlined above. Under certain conditions, some of the stand off branch lines may be instrumented with permanent sensors to monitor their accelerations during the life of the plant.

Excessive vibration or instabilities in piping systems can be difficult to analytically predict for most flow-induced vibration mechanisms. This is due to subtle differences in the thermal hydraulic conditions in the piping that can have a significant effect on the vibration response of these piping components. For these reasons, the critical piping systems (e.g., the main steam and feedwater piping systems) will be instrumented with permanent sensors that will measure the accelerations in each translational direction during the operating life of the plant. The acceptance criteria for the piping and other key components installed along the length of the pipe will be based upon satisfying the appropriate displacement, acceleration, stress, and fatigue limits.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The dynamic model used in the analysis of the RPV internals, the RPV Isolated Model, includes the core, the RPV upper and lower internals, the RPV pressure boundary, the reactor coolant loops (piping and components), the RCS supports, and the Reactor Building internal structure. The dynamic analyses consider the effects of the gaps that exist between the vessel and the core barrel, between the vessel and the upper support assembly, between the vessel and the lower support plate, between fuel assemblies, and between the fuel assemblies and the heavy reflector. See Appendix 3C for a representative diagram of the RPV Isolated Model and additional information regarding the dynamic loading analysis of this model.

Analysis of the RPV internals for blowdown loads resulting from a guillotine break of the safety injection line nozzles on the hot and cold legs is performed using direct step-by-step integration methods. Note that breaks are not considered in the main coolant loop piping (hot and cold legs), pressurizer surge line, and main steam line piping (from the steam generators to the first anchor point location) due to the application of leak-before-break methodology to these lines (see Section 3.6.3). The forcing functions obtained from hydraulic analysis of the safety injection line breaks are defined at points in the RPV internals where changes in cross-section or direction of flow occur, such that differential loads are generated during the blowdown transient. Additional details of the structural analysis of the RPV Isolated Model for LOCA loading are given in Appendix 3C.

Analysis of the RPV internals for safe shutdown earthquake (SSE) loading uses direct step-by-step time-history analysis techniques. The SSE analysis of the RPV Isolated Model is described in Appendix 3C.

The response of the RPV internals to SSE loading are combined with their response to the safety injection line breaks by the square-root-of-the-sum-of-the-squares method. Section 3.9.3 provides the faulted load combinations considered in the stress and fatigue analyses of the RPV internals.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The results of the dynamic analysis of the RPV internals are compared to the results of preoperational tests, and this comparison verifies that the analytical model provides appropriate results. If the predicted responses differ significantly from the measured values, the vibration responses are determined with the measured forcing function as input.

3.9.2.7 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. ANP-10264NP, Revision 0, “U.S. EPR Piping Analysis and Pipe Support Design Topical Report,” AREVA NP Inc., September 2006.

3. ASME OM-S/G-2000, “Standards and Guides for Operation and Maintenance of Nuclear Power Plants,” The American Society of Mechanical Engineers, 2000.

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

Table 3.9.2-1—Visual Inspection of the Accessible Areas of the Upper Internals While on the Storage Stand

Component	Subcomponent	Inspection
Top surfaces of the upper support assembly	Control rod guide assembly (CRGA) housing	Presence and condition of the bolts and their locking cups
	Level measurement probe (LMP) thimble upper housing	Presence and condition of the bolts and their locking cups
	Head and vessel alignment pins	Presence and condition of the bolts and their locking bars
	Flange	Aspect of the bearing area
Bottom surfaces of the upper support assembly	Normal columns, LMP columns, and accessible CRGA columns	Presence and condition of the bolts and their locking cups
	Flange	Hold down spring contact area
CRGA columns	Accessible guide tubes for instrumentation lance finger	Bracket fastening; presence and condition of the bolts and their spot welds
Upper core plate (UCP) top surface	CRGA columns	Aspect of the flange / CRGA pin interface
	UCP guide pin inserts	Presence and condition of the bolts and their locking cups; aspect of the wear resistant alloy surfaces
UCP bottom surface	Normal columns and LMP columns	Presence and condition of the bolts and their locking bars
	CRGA pins	Presence and condition of the locking device
	Guide tubes for instrumentation lance finger	Bracket fasteners inside the UCP; presence and condition of the bolts and their spot welds
	Upper fuel pins	Presence and condition of the pins and their spot welds

Table 3.9.2-2—Visual Inspection of the Inside of the Lower Internals in the Reactor Pressure Vessel

Component	Subcomponent	Inspection
Core barrel flange top surface	Head and vessel alignment pins	Presence and condition of the bolts and their locking bars
	Hold down spring contact area	Surface aspect
Heavy reflector top	Upper core plate (UCP) guide pins	Presence and condition of the bolts and their locking cups; aspect of the wear resistant alloy surfaces
	Tie rods	Presence and condition of the nuts and their locking devices
Lower support plate	Access plug fasteners	Presence and condition of the bolts and their locking bars
	Lower fuel pins	Presence and condition of the pins and their spot welds

Table 3.9.2-3—Visual Inspection of the Outside of the Lower Internals While on the Storage Stand

Component	Subcomponent	Inspection
Irradiation baskets	Fasteners	Presence and condition of the bolts and their locking bars
Radial key inserts	Insert fasteners	Presence and condition of the bolts and their locking bars
	Wear resistant alloy surfaces of the inserts	Surface aspect

Table 3.9.2-4—Visual Inspection of the Inside of the RPV While on the Storage Stand

Component	Subcomponent	Inspection
Reactor pressure vessel (RPV) flange	Contact surface with the lower internal flange	Surface aspect
Outlet nozzles	Potential contact surface with the lower internal nozzles	Surface aspect
Radial keys	Insert fasteners	Presence and condition of the bolts and their locking bars
	Wear resistant alloy surfaces of the inserts	Surface aspect

Table 3.9.2-5—Visual Inspection of the Inside of the RPV Head While on the Storage Stand

Component	Subcomponent	Inspection
Reactor pressure vessel (RPV) head flange	Contact surface with the upper internal flange	Surface aspect
Adaptors	Thermal sleeves	Amplitude of vertical displacement