Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: T. R. QUAY

SUBJECT: RESPONSES TO NRC MECHANICAL ENGINEERING BRANCH QUESTIONS

Dear Mr. Quay:

Attached are responses to a number of items from the NRC Mechanical Engineering Branch discussed in a telephone call with the NRC staff on October 3, 1996. The synopsis of the NRC position comes from an NRC letter dated August 20, 1996. The questions are related to quality assurance, reactor vessel internal vibration, CRDMs, and equipment seismic qualification. The questions are identified by the numbers from the August 20, 1996 letter, DSER open item or RAI number, and Open Items Tracking System item number. This response completes our responses to questions related to subsections 3.2.1, 3.9.2, 3.9.4, 3.9.7, and Section 3.10.

This submittal will permit the completion of staff review for the subsections listed and preparation of the Final Safety Evaluation Report input.

Please contact Donald A. Lindgren on (412) 374-4856 if you have additional questions.

C L Haag for BAM.
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Attachments

cc: D. T. Jackson - NRC
N. J. Liparulo - Westinghouse (w/o attachments)

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1. Open Item 3.2.1-1 (562) - Appendix B for all Seismic Cat. II - Regulatory Guide (RG) 1.29, Position C.4

In the Open Item Tracking System Database (OITS), Westinghouse reports that this issue is closed based on a statement added to Seismic Category II requirements for Quality Assurance (QA). This information was added to SSAR Section 3.2.1.1.2 in Revision 7, and states that 10 CFR 50, Appendix B does not apply to Seismic Category II structures, systems, and components. The staff does not agree. As stated in the DSER for this open item, to satisfy Position C.4 in RG 1.29, the pertinent QA requirements of Appendix B should be applied to all Seismic Category II structures, systems, and components. This commitment should be added to SSAR Section 3.2.1.1.2 and Table 3.2-1. Therefore, Open Item 3.2.1-1 remains open.

Response

When the guidance in Regulatory Guide 1.29 position C.4. was developed, the concept of graded QA had not been developed. An Appendix B quality assurance program is not needed to provide that seismic Category II systems, structures, and components do not fail in a manner that would reduce the functioning of a safety-related component. The degree of quality assurance provided for AP600 equipment Class D provides an appropriate level of quality assurance for this function. Westinghouse has defined quality assurance requirements for the regulatory treatment of nonsafety systems, systems, and components. Those requirements also are sufficient to satisfy regulatory requirements for seismic Category II.

Revise the fourth paragraph of 3.2.1.1.2 as follows:

10 CFR, 50, Appendix B does not apply to seismic Category II structures, systems, and components. **Seismic Category II structures, systems, and components have QA requirements similar to those defined for structures, systems, and components covered by regulatory treatment of nonsafety systems (see Section 17.3).** The quality assurance requirements for Seismic Category II structures, systems, and components are sufficient to provide that these components will meet the requirement to not cause unacceptable structural failure of or interaction with seismic Category I items.

This item is Resolved pending formal SSAR revision.

2. Open Item 3.2.1-2 (563) - Appendix B for new and spent fuel storage racks

In the OITS, Westinghouse reports that this issue is closed because SSAR Table 3.2-3 lists the new and spent fuel storage racks as Seismic Category I, and as such are required to meet applicable portions of Appendix B. The staff agrees that to meet RG 1.29, Appendix B should be applied to these components. However, since the new and spent fuel storage racks are classified as AP600 Class D, it is possible that this commitment might be misinterpreted when one consults SSAR Table 3.2-1. According to this table, AP600 Class D components do not have to meet either RG 1.29 seismic design requirements or Appendix B. Table 3.2-1 should be clarified by adding a note to state that although the new and spent fuel storage racks are Class D, they are designed as Seismic Category I, and meet the applicable QA requirements of Appendix B.

Response

The text in subsection 3.2.2 will be revised to clarify that Appendix B applies to Class D systems and components that are seismic Category I.

Revise the seventh paragraph of 3.2.2.6 as follows:

Standard industrial quality assurance standards are applied to Class D structures, systems, and components to provide appropriate integrity and function although 10 CFR 50. Appendix B and 10 CFR 21 do not apply. **10 CFR 50, Appendix B and 10 CFR 21 do apply to Class D structures, systems, and components that are seismic Category I.** These **industrial quality assurance** standards are consistent with the guidelines for NRC Quality Group D. The industry standards used for Class D structures, systems and components are widely used industry standards. Typical industrial standards used for Class D systems and components are provided as follows:

This item is Resolved pending formal SSAR revision.

21 Open Item 3.9.2.3-2 (783) - Flow-induced vibration prediction analysis
Action Westinghouse

The staff has received and reviewed Revision 1 (May 1, 1995) of the draft report of "AP600 Reactor Internals Flow-Induced Vibration Assessment Program" during and after the May 10, 1995, meeting. The staff's evaluation of the revised draft report concluded that Westinghouse should finalize the report by incorporating the following: (1) add a summary table of vibration prediction analysis results as included in Westinghouse letter dated June 1, 1995, (2) revise the "Introduction" section and other parts of the report for consistency with the SSAR revision, such as including statements designating the reactor internals of the first AP600 plant as a prototype, and (3) to show additional sensors at the guide tubes in Table 8.1 for monitoring their vibrations, which will be consistent with the revised SSAR Table 3.9-4 of the SSAR. The final report was submitted with changes but was not included in the references list in a recent revision of the SSAR Section 3.9.9 as discussed under DSER Open Item 3.9.2.3-1 above. Westinghouse has agreed to implement the above staff requests. Therefore, DSER Open Item 3.9.2.3-2 is technically resolved, pending acceptable completion of Westinghouse actions relative to the above requests.

Response

The report on vibration assessment will be included in the SSAR. Revise the first paragraph of subsection 3.9.2.3 of the SSAR as follows

The vibration characteristics and behavior due to flow-induced excitation are complex and not readily ascertained by analytical means alone. Assessment of vibrational response is done using a combination of analysis and testing. Comparisons of results obtained from reference plant vibration measurement programs have been used to confirm the validity of scale model tests and other prediction methods as well to confirm the adequacy of reference plant internals regarding flow induced vibration. In the following discussion the term "reference plant" is

equivalent to the term prototype as used and defined in Standard Review Plan 3.9.2 and Regulatory Guide 1.20 for vibration assessment of reactor internals. The flow-induced vibration assessment is documented in WCAP-14761 (Reference 18)

Add the following to section 3.9.9.

18. "AP600 Reactor Internals Flow-Induced Vibration Assessment Program," WCAP-14761, March, 1996.

This item is Resolved pending formal SSAR revision.

22. Open Item 3.9.2.4-1 (785) - Japanese CRDM seismic input tests - RAI# 210.94
Action Westinghouse

The staff requested more detailed information regarding production tests of the CRDM and acceptance standards for ensuring operational adequacy under loss-of-coolant-accident (LOCA) and safe shutdown earthquake (SSE) events. In the June 27, 1994 response to Q210.94, and during a subsequent meeting, Westinghouse indicated that laboratory seismic testing with a combination of a fuel assembly, CRDM, and rod cluster control assembly has been performed in Japan to demonstrate the ability of rod insertion under Japanese standard earthquake levels. A copy of the reference regarding the testing was provided to the staff (Reference 14 in the response to Q210.94 and Reference 17 in SSAR Section 3.9.9, Revision 4). The staff's review of this reference determined that Westinghouse should verify whether the Japanese test input meets the seismic qualification level of the AP600 design. This was DSER Open Item 3.9.2.4-1. In addition, the staff was told that other tests of CRDMs to ensure functioning under LOCA loads were performed and documented in WCAP-8446. This report has been reviewed and accepted by the staff as a Topical Report, and since 1976, has been referenced in most pressurized water reactor (PWR) license applications whose CRDMs were designed by Westinghouse. In the response to Q210.94, Westinghouse also proposed a revision of Section 3.9.4.4 of the SSAR to provide more descriptive information about the CRDM tests. This issue was resolved pending receipt of the SSAR revision. This was DSER Confirmatory Item 3.9.2.4-4.

However, during the May 1995 meeting, Westinghouse could not establish the basis for using the foreign test results for seismic qualification of the AP600 CRDM. Thus the referenced foreign test (Reference 17 in SSAR Section 3.9.9) was not suitable to be used for the CRDM seismic qualification of the AP600 plant. Westinghouse indicated that it will delete the reference to the test in SSAR Section 3.9.4.4 in a future revision of the SSAR. Westinghouse also indicated that demonstration of CRDM operability during a seismic event is impractical and insertion of control rods is not required as long as operability of CRDM is ensured immediately following the earthquake. The staff's subsequent evaluation concurs that demonstration of CRDM operability during a seismic event is not a regulatory requirement as long as its operability can be verified after the seismic event. However, Westinghouse should demonstrate adequacy of seismic qualification to ensure post-SSE operability of the control rod drive system in the AP600 design. In conclusion, further SSAR revision is needed to provide additional information to clarify the method and verify the adequacy of seismic qualification of the CRDM in the AP600 design. Thus Open Item 3.9.2.4-1 in conjunction with the Confirmatory Item 3.9.2.4-4 remains open, pending further Westinghouse actions.

Response

Operability of the control rod drive mechanisms is provided by limiting the allowable bending moments during seismic and other dynamic events. These limits have been validated in several test programs used to qualify the control rod drive mechanisms. The results of these tests have been provided to the NRC in support of license applications for operating nuclear power plants. There is no substantial design differences between the control rod drive mechanisms in these operating plants and the AP600 design.

Information on the operability limits for the control rod drive mechanism will be added to the AP600 SSAR.

Add the following paragraph to SSAR subsection 3.9.4.3.

To assure functional capability of the control rod drive mechanism following a seismic event or a pipe break, the bending moments on the control rod drive mechanisms are limited to those that produce stress levels in the pressure boundary of the control rod drive mechanism less than ASME Code limits during anticipated transient conditions. This limit provides that the rod travel housing does not bend to the extent that the drive rod binds during insertion of the control rods. The analysis evaluates the load combinations that include safe shutdown earthquake and pipe break. The pipe break considered is at least as large as the largest pipe in or connected to the reactor coolant system that is not qualified as a leak before break line. See subsection 3.6.3 for information on the lines qualified for leak-before-break. See subsection 3.9.7 for information on the control rod drive mechanism deflection limit requirements for the integrated head package

This item is Resolved pending formal SSAR revision.

24. Open Item 3.9.3.1-5 (790) - ISLOCA criteria - RAI# 210.61
Action Westinghouse

In the response to RAI# 210.61, the proposed revision to the SSAR Sections 1.9.5.1 and 5.4.7.2.2 did not include a commitment to design the low pressure side of the applicable piping systems to 40% of the RCS design pressure. In a telephone conference on April 11, 1995, Westinghouse agreed to revise the SSAR to include this commitment. SSAR Revision 5 provided this acceptable commitment in Section 5.4.7.2.2. However, SSAR Section 1.9.5.1 has not yet been revised.

Response

Section 1.9.5.1 references the criteria in 5.4.7. This approach provides sufficient information and no change to the SSAR is required.

This item is Closed.

31. Potential thinning of incore neutron monitoring thimble tubes - RAI# 210.214 (OITS #3374)
Action Westinghouse

The incore neutron monitoring thimble tubes have experienced thinning as a result of flow-induced vibration in operating PWRs of Westinghouse design. NRC Bulletin 88-09 had requested all licensees of these plants to establish and implement an inspection program to periodically confirm incore thimble tube integrity. Westinghouse is requested to provide information to verify that either such a concern does not exist in AP600 due to an improved thimble design, or an inspection program in conformance with guidelines stated in NRC Bulletin 88-09 was established and will be implemented as a COL action item in all AP600 plants. In the latter case, provide a description of the inspection program in the SSAR.

Response

The AP600 incore thimble is an improved design using Inconel (Ni-Cr-Fe) Alloy 690 material for wear resistance, has a larger diameter, and is stiffer compared to thimbles in conventional Westinghouse PWR plants to minimize vibration. In addition, the gap between the thimble tube and the guide tube above the reactor vessel and through the upper reactor vessel internals is reduced compared to the gap in operating plants. This feature results in more points of contact to also minimize vibration. The incore thimble design has an outer tube around fixed detectors and a central tube or solid mandrel. There is no provision for use of movable incore detectors. Based on the design enhancements and experience at an operating plant, the thimbles are not expected to experience significant wear.

The outer tube of the thimble and the seal at the end of the guide tube near the top of the integrated head package are reactor coolant pressure boundary. The connecting leads for the incore detectors exit the thimble tube assembly through a pressure tight seal. A leak through the outer tube into the gap between outer tube and the central tube or mandrel would be retained within the thimble tube assembly and does not represent an non-isoable leak.

These features preclude the need for inservice inspection.

Revise the final paragraph of SSAR subsection 3.9.7.2 as follows:

In-core instrumentation support structure (IISS) - The in-core instrumentation support structure is used during refueling operations. This support structure is used for withdrawing the in-core instrumentation thimble assemblies into the integrated head package. It protects and supports the thimble assemblies when they are in the fully withdrawn position. The in-core instrumentation system consists of thermocouples to measure fuel assembly coolant outlet temperature, and in-core flux thimbles containing fixed detectors for measurement of the neutron flux distribution within the reactor core. The incore thimble tubes are designed for resistance to fluid-induced vibration and wear. The thimble is stiffer than the design in previous operating plants and the gap between the thimble tube and the tubes used to guide and protect the thimble inside the reactor vessel is smaller to minimize vibration. The potential for wear is also addressed by the material selection for the tube. The design of the thimble tube assembly also precludes an non-isoable leak of reactor coolant. The

thermocouples and neutron detectors are routed through the integrated head package. These are inserted into the core through the reactor vessel head and upper internals assembly. Also, the in-core instrumentation support structure includes a platform which provides access to the in-core instrumentation during maintenance and refueling and to attach the lifting system to the crane hook.

This item is Resolved pending formal SSAR revision.

32. Open Item 3.9.7-1 (812) - Deflection limits for integrated head package RAI# 210.97
Action Westinghouse

SSAR Section 3.9.7.3 states that because of the application of LBB, breaks are not postulated in reactor coolant loop (RCL) pipes down to 4-inch diameter. Therefore, these loads are not considered in the design of the integrated head package. It should be noted that the application of LBB in small lines has not yet been endorsed by the staff. Subsequent to the resolution of Open Items 3.6.3.4-1 and 3.6.3.6-4, loads from postulated breaks in some small diameter pipe may have to be included in this design.

In DSER Open Item 3.9.7-1, the staff requested a description of the analyses and or test data that was used to establish the deflection limit of the top of the control rod drive mechanism rod travel housing. The response to this open item in Revision 5 to Section 3.9.7.3 of the SSAR does not provide sufficient detail for the staff to prepare a safety evaluation of this issue. As a part of a future design review meeting with the staff, W is requested to identify the documents that contain a description of these analyses and tests, and be prepared to discuss the design basis loads and methodology used to establish the deflection limits, and the test procedures and results obtained to demonstrate that the drive rod will not bind during insertion while being subjected to these loads. If these analyses and tests were not AP600 specific, discuss the basis for applying the results to the AP600 design.

Pending (1) resolution of Open Items 3.6.3.4-1 and 3.6.3.6-4, and (2) the staff's review and acceptance of the analysis and testing discussed above, Item 3.9.7-1 remains open.

Response

Please see the response to 3.9.2.4-1. The deflection limit for the integrated head package is based on limiting the deflection of the CRDM. Deflection of the CRDM is evaluated analytically. RAI 210.228 in NRC Letter dated August 20, 1996 identifies that the staff has concerns with application of leak-before-break to "...small diameter (4-inch) piping...". The differences in dynamic loads between a 4 and 6 inch LOCA is not limiting for the design and analysis of the control rod drive mechanism.

This item is Closed

33. Open Item 3.10-1 (813) - Use of seismic experience data
Action Westinghouse

Revision 5 to the SSAR revised Section 10.2 to respond to this issue. This revision is identical to the response to Q210.81. In Section 3.10 of the DSER, dated November, 1994, the staff stated that this response was not completely acceptable, and identified this issue as a DSER Open Item. The SSAR should be revised to state that the COL applicant will submit all of the information described in the DSER to the staff for review and approval prior to including this information in the equipment qualification file. In addition, WCAP 13054 should be revised to delete the exception to the applicable portion of SRP 3.10.

Response

Section 3.10.6 of the SSAR will be revised as follows

The Combined License applicant will address, as part of the Combined License application, identification of the equipment qualified based on experience and include details of the methodology and the corresponding experience data. The corresponding experience data for each piece of equipment will be included in the equipment qualification file.

The applicable portion of WCAP-13054 was revised in Revision 2.

This item is Resolved pending formal SSAR revision.

34. Open Item 3.10-2 (814) - Dynamic analysis of valve disks
Action Westinghouse

Revision 5 to the SSAR revised Section 3.10.2.2 to respond to this item. The revision states: "Valve disks are evaluated for maximum design line pressure and maximum differential pressure resulting from plant operating, transient, and accident conditions." This does not appear to address the issue as described in the DSER, i.e., the SSAR should be revised to describe the methodology used in the AP600 design to analyze the feedwater line valve disks when they are subjected to dynamic loads due to a LOCA. In addition, as requested in the DSER, WCAP 13054 should be revised to delete an exception to SRP, Section 3.10.II.1.a(14)(b).

Response

Dynamic loads in valves from pipe breaks are evaluated by considering the effect of an equivalent differential pressure on the valve disk.

The fourth paragraph of SSAR subsection 3.10.2.2 will be revised as follows:

The safety-related valves are subjected to a series of tests before service and during the plant life. Before installation, the following tests are performed: body hydrostatic test to ASME Code, Section III, requirements, back-seat and main seat leakage tests, disc hydrostatic tests, and operational tests to verify that the valve opens and closes. For the qualification of motor

operators for environmental conditions, see Section 3.11. After installation, the valves undergo system level hydrostatic tests, construction acceptance tests, and preoperational tests. Where applicable, periodic in-service inspections and operations are performed in situ to verify the functional capability of the valve. On active valves, an analysis of the extended structure is performed for static equivalent seismic safe shutdown earthquake loads applied at the center of gravity of the extended structure. The maximum stress limits used for active Class 1, 2, and 3 valves are compared to acceptable standards in the ASME Code. Valve discs are evaluated for maximum design line pressure and maximum differential pressure resulting from plant operating, transient, and accident conditions. Feedwater line valve discs are evaluated for the effect of dynamic loads of pipe breaks by considering the effect of an equivalent differential pressure. Valve operating conditions are included as part of the valve design specification and are used to evaluate the valve disc. Additional information is provided on the controlled-closure, feedwater check valve in subsection 10.4.7.2.2.

The exception has been deleted from WCAP-13054, in Revision 2.

This item is Resolved pending formal SSAR revision.

35. Open Item 3.10-3 (815) - Reactor Coolant Pressure Boundary (RCPB) valve leakage per SRP 3.10 should be in SSAR Action Westinghouse

Revision 5 to the SSAR revised Section 3.10.2.2 to provide an acceptable response to this item. However, the exception to SRP, Section 3.10.II.4 in WCAP-13054 has not yet been deleted. Therefore, this issue remains open.

Response

The exception was removed from WCAP-13054, Revision 2

This item is Closed.

36. Open Item 3.10-4 (816) - Aging by analysis - IEEE 323-1983 vs 1974
Action Westinghouse

Revision 5 to the SSAR revised Appendix 3D to commit to the staff's position to use IEEE 323-1974 rather than the 1983 edition. However, SSAR, Section 3.11 has not yet been revised to provide the same commitment (Ref. Open Item 3.11.3.2-1), and WCAP 10354 has not yet been revised to delete the exception to SRP Section 3.10.II.1.c.

Response

SSAR Revision 8 removed the reference to IEEE 323-1974 in Section 3.11. The exception was removed in WCAP-13054, Revision 2

This item is Closed.