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ASEA BROWN BOVERI

December 23, 1992
LD-92-123

Docket No. 52-002

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Closure of System 80+™ Draft Safety Evaluation Report Issues

Dear Sirs:

Enclosed with this letter are responses to 79 of the issues identified in the Draft Safety Evaluation Report (DSER) for System 80+. The enclosed responses are sorted by review branch, including a listing of the issues for that branch with two check marks in the left column to indicate which responses are included. The number of responses submitted is now 619.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

Stanley E. Ritterbusch for

C. B. Brinkman
Acting Director
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CBB/ser
cc: J. Trotter (EPRI)
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Civil Engineering and Geosciences Branch

NUMBER	TYPE	TITLE	BRANCH
02.1.1-1	COL ITEM	The COL applicant must provide site-specific information on site and location.	ECGB
02.1.1-1	COL ITEM	The COL applicant must provide site-specific information on the exclusion area authority and control.	ECGB
02.1.3-1	COL ITEM	The COL applicant must provide site-specific information on the population distribution.	ECGB
02.2.2-1	COL ITEM	The COL applicant must provide site-specific information on transportation.	ECGB
02.2.3-1	COL ITEM	The COL applicant must provide site-specific information on industrial hazards.	ECGB
02.3-1	COL ITEM	The COL applicant must provide detailed site characteristics on meteorology.	ECGB
02.3-2	COL ITEM	The COL applicant will document verification that site-specific assumptions are within the values specified in the CESSAR.	ECGB
02.4.01-1	COL ITEM	The COL applicant must provide site-specific information on external floods.	ECGB
02.4.03-1	COL ITEM	The COL applicant must provide site-specific information on the probable maximum flood on streams and rivers.	ECGB
02.4.04-1	COL ITEM	The COL applicant must provide site-specific information on potential dam failures.	ECGB
02.4.05-1	COL ITEM	The COL applicant must provide site-specific information on probable maximum surge and seiche flooding.	ECGB
02.4.06-1	COL ITEM	The COL applicant must provide site-specific information on probable maximum tsunami loading.	ECGB
02.4.07-1	COL ITEM	The COL applicant must provide site-specific information on ice effect.	ECGB
02.4.08-1	COL ITEM	The COL applicant must provide site-specific information on cooling water canals and reservoirs.	ECGB
02.4.09-1	COL ITEM	The COL applicant must provide site-specific information on channel diversions.	ECGB
02.4.10-1	COL ITEM	The COL applicant must provide site-specific information on flood protection requirements.	ECGB
02.4.11-1	COL ITEM	The COL applicant must provide site-specific information on the cooling water supply.	ECGB
02.4.12-1	COL ITEM	The COL applicant must provide site-specific information on groundwater.	ECGB
02.4.13-1	COL ITEM	The COL applicant must provide site-specific information on accidental release of liquid effluents in g und and surface water.	ECGB
02.4.14-1	COL ITEM	The COL applicant must provide site-specific information on technical specifications and emergency operation requirements.	ECGB
02.5-1	COL ITEM	The COL applicant will perform a site-specific soil column analysis to calculate response spectra.	ECGB
02.5.1-1	COL ITEM	The COL applicant will provide site-specific information on physiography, geomorphology, stratigraphy, lithography and tectonics.	ECGB
02.5.2.5.1-1	COL ITEM	The COL applicant must show that peak ground accelerations and site design response spectra meet site envelope parameters.	ECGB
02.5.2.5.2-1	COL ITEM	The COL applicant will develop site specific geological, seismological and geotechnical data.	ECGB
02.5.2.5.2-2	COL ITEM	The COL applicant must ensure site-specific conditions at a shallow soil site conform to the standard design.	ECGB
02.5.3-1	COL ITEM	The COL applicant must demonstrate that no potential exists for surface faulting affecting the site.	ECGB
02.5.4.01-1	COL ITEM	The COL applicant must provide site-specific information on the geologic features underlying the site.	ECGB
02.5.4.02-1	COL ITEM	The COL applicant must provide the state-of-the-art methods to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area.	ECGB
02.5.4.02-2	COL ITEM	The COL applicant will verify that minimum shear wave velocity of foundation soils is not less than those stated in CESSAR.	ECGB
02.5.4.03-1	COL ITEM	The COL applicant will provide all data pertaining to site-specific soil layers between the basemat and the underlying rock stratum.	ECGB
02.5.4.04-1	COL ITEM	The COL applicant will perform geophysical and geotechnical investigations at the site.	ECGB
02.5.4.05-1	COL ITEM	The COL applicant will provide data concerning the extent of all seismic Category I excavations, fills, and slopes.	ECGB
02.5.4.06-1	COL ITEM	The COL applicant will discuss groundwater conditions relative to foundation stability of safety related structures.	ECGB
02.5.4.07-1	COL ITEM	The COL applicant must show that the CESSAR assumptions regarding variation of shear wave velocity are applicable to specific site.	ECGB
02.5.4.08-1	COL ITEM	The COL applicant must show that no liquefaction potential exists at SSE level for soils under seismic Category I structures.	ECGB
02.5.4.10-1	COL ITEM	The COL applicant must show that site soil bearing capacity is equal to or exceeds the value in CESSAR Table 2.0-1.	ECGB
02.5.5-1	COL ITEM	The COL applicant must provide site-specific information on stability of slope.	ECGB
02.5.6-1	COL ITEM	The COL applicant must provide site-specific information on embankments and dams.	ECGB
03-1	COL ITEM	The COL applicant should idenfify the applicable ISI and IST code editions in accordance with 10 CFR 50.55a(g).	ECGB
03.03.1-1	COL ITEM	The COL applicant must ensure that the velocity of wind stated in the CESSAR is not exceeded by the site specific design basis wind.	ECGB
03.03.2-1	COL ITEM	The COL applicant must ensure that the CESSAR tornado loadings are bounding for the site specific location.	ECGB
03.04.1-1	COL ITEM	The maximum site-specific flood levels and other safety-related structures where flood protection measures are required for the site will be addressed by the COL applicant.	ECGB
03.04.2-2	COL ITEM	The COL applicant shall ensure that all seismic Category I structures are protected against flood damage.	ECGB
03.05.1.3-1	COL ITEM	The COL applicant should submit a summary of the turbine maintenance and inspection program and results of probabilistic evaluation.	ECGB
03.05.1.5-1	COL ITEM	The missiles generated near the site will be addressed in the site specific SAR.	ECGB
03.05.1.6-1	COL ITEM	The aircraft hazards will be considered on a site specific basis.	ECGB

NUMBER	TYPE	TITLE	BRANCH
03.06.2-1	COL ITEM	The COL applicant must provide final designs of high and moderate energy fluid systems.	ECGB
03.06.3-1	COL ITEM	The COL applicant should verify that the actual material properties and final piping analyses are within the bounding LBB analyses.	ECGB
03.08.5-1	COL ITEM	The COL applicant should submit the site-specific foundation mat construction procedures.	ECGB
03.09.3.1-1	COL ITEM	The COL applicant should verify that the edition of the ASME code used in the site-specific design is in accordance with DSER Section 3.0.	ECGB
03.09.3.4-1	COL ITEM	The COL applicant must provide a listing of all safety-related components which utilize snubbers per SRP 3.9.3.	ECGB
05.2.1.1-1	COL ITEM	The COL applicant must specify the ASME Code edition that will be used in the construction of the reactor coolant pressure boundary components.	ECGB
05.2.2.2-1	COL ITEM	The COL applicant should determine the LTOP enable temperature based on the plant specific material properties and pressure-temperature limit curves.	ECGB
05.2.2.3-1	COL ITEM	The COL applicant should verify that the material properties and end-of-life fluence (60 years) are within the limits assumed in the CESSAR.	ECGB
05.2.4-1	COL ITEM	The COL applicant should submit its PSI and ISI program plans for staff review.	ECGB
05.3.1-01	COL ITEM	The COL applicant should verify that the assumptions of material properties and 60-year fluence apply to the actual plant specific values.	ECGB
05.3.2-1	COL ITEM	The COL applicant should submit plant-specific material fracture toughness data and the resulting pressure-temperature curves.	ECGB
20.1-01	COL ITEM	The COL applicant should submit its steam generator tube inservice inspection program for staff review.	ECGB
20.2-01	COL ITEM	The COL applicant should verify that the CESSAR assumptions regarding the reactor vessel supports' material properties and 60-year neutron fluence are met.	ECGB
02.4.03-1	CONF ITEM	The applicable contents of ABB-CE letter LD-92-045 should be incorporated into the CESSAR.	ECGB
03.06.2-1	CONF ITEM	The staff will confirm that the applicant revises CESSAR Section 3.6.2.1 as previously committed.	ECGB
03.06.2-2	CONF ITEM	The staff will confirm that the applicant will revise the CESSAR to reference the 1988 edition of ANSI/ANS-58.2.	ECGB
03.07-1	CONF ITEM	The Applicant must modify or update CESSAR as discussed in DSER Section 3.7.	ECGB
03.07.2-1	CONF ITEM	The applicant must incorporate responses to RAIs Q220.5, 220.11, 220.18, 220.20 and 220.21 into the CESSAR.	ECGB
03.07.2-2	CONF ITEM	The applicant committed to revise the note in CESSAR Table 3.7-1 to commit to all conditions of RG 1.84 on the use of ASME N-411-1.	ECGB
03.07.2-3	CONF ITEM	The applicant should clarify CESSAR Section 3.7.2.11 to state how the additional eccentricity of 5 percent of the maximum building dimension will be applied.	ECGB
03.07.2-4	CONF ITEM	The applicant committed to clarify CESSAR Section 3.7.2.13 statements associated with the seismic analysis of the safety-related dams.	ECGB
03.07.3-01	CONF ITEM	The staff will confirm that the applicant uses the modeling acceptance criteria of SRP Section 3.7.2.	ECGB
03.07.3-02	CONF ITEM	The staff will confirm that the responses to RAIs Q210.36 and 210.37 are incorporated into the CESSAR.	ECGB
03.09.1-1	CONF ITEM	The staff will confirm that the applicant adds a description of the SASSI program to the CESSAR.	ECGB
03.09.3.1-1	CONF ITEM	The staff will confirm that the applicant revises CESSAR Section 3.9.3 as previously proposed.	ECGB
03.09.3.3-1	CONF ITEM	The staff will confirm that the applicant revises CESSAR Section 3.9.3.3 as previously proposed.	ECGB
02.4.14-1	OPEN ITEM	The applicant should revise letter LD-92-045 to remove reference to subjects not addressed by the CESSAR.	ECGB
02.5-1	OPEN ITEM	The applicant should use envelope response spectra for design analysis of seismic Category I structures.	ECGB
02.5.2.5.1-1	OPEN ITEM	The time histories associated with CMS2 do not satisfy SRP 3.7.1 acceptance criteria for 7 percent damping.	ECGB
02.5.2.5.1-2	OPEN ITEM	The problem of significant "valley" occurring in the foundation spectra presented in a previous meeting must be addressed.	ECGB
02.5.2.5.1-3	OPEN ITEM	The CESSAR should be revised to include CMS1 and CMS3.	ECGB
02.5.2.5.1-4	OPEN ITEM	The staff must review the applicant's formal discussion in CESSAR on how CMS1 will be used.	ECGB
02.5.2.8-1	OPEN ITEM	The applicant should address soil properties associated with compression waves.	ECGB
02.5.3-1	OPEN ITEM	The applicant should clearly state in CESSAR that plant will not be designed to withstand surface faulting.	ECGB
02.6-1	OPEN ITEM	The applicant should clarify how emergency cooling water and condenser cooling water inlet temperatures will be used in the design.	ECGB
02.6-2	OPEN ITEM	The applicant should include the additional site parameters the staff has listed (see DSER Section 2.6).	ECGB
03-1	OPEN ITEM	The applicant should verify that specific editions of all national codes and standards referenced have been identified (except ISI and IST).	ECGB
03.05.3-1	OPEN ITEM	The applicant should incorporate Table 1 of SRP 3.5.3 into the CESSAR.	ECGB
03.06.2-1	OPEN ITEM	CESSAR Section 3.6.2 refers to descriptions of the results of a determination of break locations and dynamic effects of ruptured piping. No such descriptions were found.	ECGB
03.06.2-2	OPEN ITEM	The applicant should revise CESSAR Section 3.6.2 in accordance with the staff position that CESSAR Section 3.6.2 criteria apply to high and moderate energy system piping.	ECGB
03.06.2-3	OPEN ITEM	The applicant should revise the stress criteria in CESSAR Section 3.6.2.1.4.1 F.	ECGB
03.06.2-4	OPEN ITEM	The applicant should not revise break location criteria as proposed in its response to RAI Q210.14.	ECGB
03.06.2-5	OPEN ITEM	The staff does not have enough information to review the alternative approach to the design detail regarding pipe rupture.	ECGB
03.06.2-6	OPEN ITEM	The acceptability of the methods of the dynamic analysis of pipe whip cannot be determined at this time.	ECGB

NUMBER	TYPE	TITLE	BRANCH
03.06.3-1	OPEN ITEM	The applicant should submit its bounding LBB analyses.	ECGB
03.07-1	OPEN ITEM	The applicant must complete the seismic analyses of all Category I structures and update the CESSAR to include Tier 1 and 2 information.	ECGB
03.07.1-1	OPEN ITEM	The time histories of CMS2 do not satisfy the SRP 3.7.1 acceptance criteria for 7 percent damping.	ECGB
03.07.1-2	OPEN ITEM	The applicant should submit the time histories and their corresponding response spectra associated with CMS1 and CMS3.	ECGB
03.07.1-3	OPEN ITEM	The applicant should clarify whether or not ASME Code Case N-411 damping values will be used as discussed in RG 1.84.	ECGB
03.07.2-1	OPEN ITEM	The applicant's seismic analyses for all seismic Category I structures are not complete.	ECGB
03.07.2-2	OPEN ITEM	The staff requires that the detailed process of developing dynamic models of nuclear island structures, including the fine-tuning, be documented in an auditable form.	ECGB
03.07.2-3	OPEN ITEM	The applicant should demonstrate that the 13 generic soil conditions provide a conservative envelope.	ECGB
03.07.2-4	OPEN ITEM	The applicant should define criteria to ensure that the stick models developed are equivatlent to the 3-D finite element models.	ECGB
03.07.2-5	OPEN ITEM	The applicant should demonstrate that issues addressed in SRP Section 3.7.3 Paragraph II.1.a.(iii) on reducing large static models, have been satisfactorily considered.	ECGB
03.07.2-6	OPEN ITEM	The applicant should describe analysis methods and design criteria that will be used to ensure structural integrity of non-safety related structures.	ECGB
03.07.2-7	OPEN ITEM	The applicant should clarify statement in CESSAR Section 3.7.2.9 to clearly describe the procedures employed.	ECGB
03.07.2-8	OPEN ITEM	The applicant should provide definitions of the damping terms used and guidance for estimating proportional damping ratio for the time history method.	ECGB
03.07.3-07	OPEN ITEM	The applicant should provide generic approaches used in the evaluation of the intake structure as well as the acceptance criteria that will be used to evaluate that structure.	ECGB
03.07.3-09	OPEN ITEM	The applicant should provide generic approaches to, and acceptance criteria for evaluation of buried or above-ground tanks.	ECGB
03.07.3-10	OPEN ITEM	The applicant should provide generic approaches and acceptance criteria for evaluation of buried piping, conduits and tunnels.	ECGB
03.07.3-12	OPEN ITEM	The applicant should present a complete set of information as discussed in DSER Section 3.7.3.	ECGB
03.07.4-1	OPEN ITEM	The applicant should clarify CESSAR Section 3.7.4.4 by requiring the plant operating procedures to define "significant exceedance" of design earthquake level of interest.	ECGB
03.08.2-01	OPEN ITEM	The applicant should address the uncertainty of the mechanical properties, environmental qualification, and aging effects on the self-expanding cork in the transition region.	ECGB
03.08.2-02	OPEN ITEM	The applicant should address the measures to be implemented to prevent the collection of moisture in the transition region.	ECGB
03.08.2-03	OPEN ITEM	The applicant should address the containment shell seismic fragility and containment performance in PRA evaluation for a beyond design basis event.	ECGB
03.08.2-04	OPEN ITEM	The applicant should provide the stress analysis results for the most highly stressed meridian as previously discussed with the staff during a April 29, 1992 meeting.	ECGB
03.08.2-05	OPEN ITEM	The applicant should describe the method used to verify that designs of penetrations and reinforcements satisfy stress limits of SRP Section 3.8.2.	ECGB
03.08.2-06	OPEN ITEM	The applicant should verify that the finite element mesh size is small enough to have achieved convergence of the ANYSYS bifurcation buckling load.	ECGB
03.08.2-07	OPEN ITEM	The applicant should substantiate the buckling shape resulting from the previous analysis or perform an additional analysis to eliminate anomalies.	ECGB
03.08.2-08	OPEN ITEM	The applicant should justify acceptability of the factor of safety of 2 for stability with Level C loading condition.	ECGB
03.08.2-09	OPEN ITEM	The applicant should submit prebuckling stresses for the most highly stressed meridian and verify that stresses at buckling are in the elastic range.	ECGB
03.08.2-10	OPEN ITEM	The applicant should verify that Sandia strain criteria have been satisfied for all strains in axisymmetric analysis model.	ECGB
03.08.2-11	OPEN ITEM	The applicant should describe the method to be used to verify that all strains at the discontinuities satisfy Sandia strain criteria.	ECGB
03.08.2-12	OPEN ITEM	The applicant should provide a corrosion analysis of the containment for a 60-year plant design life.	ECGB
03.08.3-1	OPEN ITEM	The applicant should explicitly address the effects of concrete cracking in the seismic analysis of all Category I structures.	ECGB
03.08.4-1	OPEN ITEM	The applicant should provide design descriptions, assumptions and criteria for all seismic Category I structures.	ECGB
03.08.4-2	OPEN ITEM	The applicant should clarify its commitment to design all subcompartments for global pressure/temperature effects.	ECGB
03.08.5-1	OPEN ITEM	The applicant should provide a description in the CESSAR design description, assumptions and criteria for the foundations of all seismic Category I structures.	ECGB
03.08.5-2	OPEN ITEM	The applicant should complete a design analysis of the foundation mats for nuclear annex and containment and their respective internal structures.	ECGB
03.08.5-3	OPEN ITEM	The applicant should provide acceptance criteria regarding the factors of safety against overturning, sliding and floating of the spherical containment.	ECGB
03.09.3-1	OPEN ITEM	The applicant should submit the entire final version of the DSDG for staff review.	ECGB
03.09.3.1-1	OPEN ITEM	The applicant should correct or clarify all of the loading combination tables in the CESSAR.	ECGB
03.09.3.1-2	OPEN ITEM	The applicant should identify the level C and D service condition transients in load combination tables in CESSAR Section 3.9.3.	ECGB
03.09.3.1-3	OPEN ITEM	The applicant should revise the loading combination information in CESSAR Section 3.9.3 and elsewhere to include pipe rupture.	ECGB

NUMBER	TYPE	TITLE	BRANCH
03.09.3.1-4	OPEN ITEM	The applicant should commit to perform an ASME Section III, Class 1 fatigue analysis on Class 2 and 3 components.	ECGB
03.09.3.1-5	OPEN ITEM	The applicant should conform to NRC Bulletins 88-08 and 88-11 for piping connected to the reactor coolant system.	ECGB
03.09.3.1-6	OPEN ITEM	The applicant should provide functional capability criteria for piping products and piping material in piping systems.	ECGB
03.09.3.1-7	OPEN ITEM	The applicant should commit to provide explicit design criteria in accordance with SRP 3.9.3, Section II.1 for internal parts of components.	ECGB
03.09.3.1-8	OPEN ITEM	The applicant should prepare procedures for generating design specifications for procurement of ASME Section III components.	ECGB
03.09.3.1-9	OPEN ITEM	The applicant should submit explicit information regarding proposed design criteria to be used for duct support construction.	ECGB
03.09.3.2-1	OPEN ITEM	The applicant should revise CESSAR Sections 3.9.3.2 and 3.10.	ECGB
03.09.3.4-1	OPEN ITEM	The applicant should commit to "construct" pipe supports to ASME Section III, Subsection NF requirements.	ECGB
03.09.3.4-2	OPEN ITEM	The applicant should revise CESSAR Section 3.9.3 to commit to a jurisdictional boundary between ASME Code, Subsection NF and building structures.	ECGB
03.09.3.4-3	OPEN ITEM	The applicant's commitment to ACI-349 is unacceptable because ACI-349, Appendix B has not been endorsed by the staff.	ECGB
03.10-1	OPEN ITEM	The staff cannot complete its review until the applicant has submitted a revised CESSAR Section 3.10.	ECGB
04.5.1-1	OPEN ITEM	Inconel 600 may be used in the CEDM motor housing assembly. Inconel 600 is susceptible to cracking. The applicant should consider alternate materials.	ECGB
04.5.1-2	OPEN ITEM	The applicant should consider the use of ASTM A262 in the CEDM instead of A708 for verifying the non-sensitization of austenitic stainless steel materials.	ECGB
04.5.1-3	OPEN ITEM	The applicant should explain fully why they are using materials that contain cobalt for the pins and latches in the CEDM.	ECGB
04.5.1-4	OPEN ITEM	The applicant is using Type 304 and 316 stainless steels in the CEDM. These steels are susceptible to stress corrosion cracking.	ECGB
04.5.1-5	OPEN ITEM	The applicant's ferrite content limits for austenitic steel castings and weld metal do not conform to the industry or staff guidelines.	ECGB
04.5.1-6	OPEN ITEM	CESSAR Section 4.5.1.1 indicates that martensitic stainless steel will be used. The applicant should specify the heat treatment for these materials.	ECGB
04.5.1-7	OPEN ITEM	CESSAR Section 4.5.1.1 indicates that Inconel X-750 will be used. The applicant should verify that this is an acceptable material and specify the heat treatment.	ECGB
04.5.1-8	OPEN ITEM	CESSAR Section 4.5.1.3.3 indicates a carbon content limit for austenitic stainless steel. The applicant should consider a 0.02% limit.	ECGB
04.5.1-9	OPEN ITEM	CESSAR Section 4.5.1.1 indicates that CEDM materials were tested and exceed lifetime requirements. The applicant should verify that a 60-year life was used.	ECGB
04.5.2-1	OPEN ITEM	The applicant is proposing to use a cobalt based alloy as a hardfacing material. The applicant should demonstrate why an alternative material is unacceptable.	ECGB
04.5.2-2	OPEN ITEM	The applicant should consider the use of ASTM A262 instead of A708 for verifying non-sensitization of austenitic stainless steel reactor internal materials.	ECGB
04.5.2-3	OPEN ITEM	The applicant should consider using low carbon wrought austenitic stainless steel instead of Type 304 stainless steel.	ECGB
04.5.2-4	OPEN ITEM	CESSAR Section 4.5.2.1 indicates that Inconel will be used for the flow skirt. The applicant should specify the type of Inconel to be used and consider an alternate material.	ECGB
04.5.2-5	OPEN ITEM	The applicant's ferrite content limits for austenitic steel castings and weld metal is not in conformance with industry and staff guidance.	ECGB
04.5.2-6	OPEN ITEM	CESSAR Section 4.5.2.1 indicates that precipitation hardened stainless steel will be used. The applicant should specify the heat treatment.	ECGB
04.5.2-7	OPEN ITEM	CESSAR Section 4.5.2.3.1.4 indicates a carbon content limit for austenitic stainless steel. The applicant should consider a 0.02% limit.	ECGB
05.2.1.2-1	OPEN ITEM	The applicant should provide a complete list of all ASME Code case interpretations referenced in the CESSAR.	ECGB
05.2.2.2-1	OPEN ITEM	CESSAR Section 5.2.2.4.4.4 states that Stellite will be used in relief valve discs. The applicant should demonstrate why an alternative material is unacceptable.	ECGB
05.2.2.3-1	OPEN ITEM	The applicant did not use the 10 CFR 50.61 margin in its pressurized thermal shock calculation.	ECGB
05.2.3-01	OPEN ITEM	The applicant should conform to RG 1.50 recommendations or propose an acceptable alternative.	ECGB
05.2.3-02	OPEN ITEM	The applicant should comply with the guidance in RG 1.71 for welding under conditions of limited accessibility.	ECGB
05.2.3-03	OPEN ITEM	The applicant should identify the Inconel materials in the CESSAR and not wait until the procurement phase to identify the Inconel materials.	ECGB
05.2.3-04	OPEN ITEM	The applicant should use ASTM A262 instead of A708 for verifying the non-sensitization of austenitic stainless steel reactor coolant pressure boundary materials.	ECGB
05.2.3-05	OPEN ITEM	The applicant should provide a complete list of the materials used for reactor coolant pressure boundary components in CESSAR Table 5.2-2.	ECGB
05.2.3-06	OPEN ITEM	The applicant should explicitly account for the effects of the environment in the fatigue analysis of components.	ECGB
05.2.3-07	OPEN ITEM	The applicant should revise the primary water chemistry to be consistent with the EPRI Guidelines and EPRI Utility Requirement Document.	ECGB
05.2.3-08	OPEN ITEM	The applicant should consider alternatives to cast austenitic stainless steel materials due to thermal aging and inspection concerns.	ECGB
05.2.3-09	OPEN ITEM	The applicant should be consistent with industry or staff guidelines for ferrite content limits of austenitic stainless steels.	ECGB

NUMBER	TYPE	TITLE	BRANCH
05.2.3-10	OPEN ITEM	The applicant should provide a discussion relating to the lubricants used for threaded fasteners within the RCPB. The use of molybdenum disulfide should be justified.	ECGB
05.2.3-11	OPEN ITEM	The applicant should impose controls on grinding austenitic stainless steel materials to avoid introducing a susceptibility to stress corrosion cracking.	ECGB
05.2.3-12	OPEN ITEM	The applicant should provide justification for the use of SA 540 Grade B23 or B24 bolting materials in the RCPB.	ECGB
05.2.3-13	OPEN ITEM	The applicant should consider the effects of dynamic strain aging of carbon steel materials.	ECGB
05.2.3-14	OPEN ITEM	The applicant states that Type 304 and 316 stainless steel will be used in the RCPB. The applicant should consider using low carbon wrought austenitic stainless steel.	ECGB
05.2.3-15	OPEN ITEM	The applicant's reference to RG 1.2 should be deleted since it has been withdrawn by the NRC.	ECGB
05.2.3-16	OPEN ITEM	CESSAR Section 5.2.3.4.1.1.1 indicates a carbon content for austenitic stainless steel. The applicant should consider a 0.02 percent limit.	ECGB
05.2.4-1	OPEN ITEM	The applicant should state that all Class 1 components will be designed to be accessible for ASME Section XI inspections.	ECGB
05.2.4-2	OPEN ITEM	The CESSAR should state that PSI will meet the construction edition of ASME Section XI and ISI Section XI will be in accordance with 10 CFR 50.55a(g).	ECGB
05.2.4-3	OPEN ITEM	The applicant should state that all PSI requirements, of ASME Section XI of same edition of ASME code used for construction, will be met.	ECGB
05.2.4-4	OPEN ITEM	The applicant should revise CESSAR to define the division of responsibility between ABB-CE and the COL applicant regarding PSI and ISI.	ECGB
05.2.4-5	OPEN ITEM	The applicant should state that PSI and subsequent ISI will be conducted with equivalent equipment and techniques.	ECGB
05.2.4-6	OPEN ITEM	The staff is recommending the use of ASME Section XI Appendices VII and VIII for CESSAR.	ECGB
05.2.4-7	OPEN ITEM	The staff is recommending the use of ASME Section XI Subsection IWH for CESSAR.	ECGB
05.2.4-8	OPEN ITEM	The applicant should confirm that the value of the cumulative usage factor (CUF) will correspond to a 60-year plant design life.	ECGB
05.3.1-01	OPEN ITEM	The applicant should consider lowering the nickel content in the reactor vessel forging and the phosphorous content in the reactor vessel forging and weld.	ECGB
05.3.1-02	OPEN ITEM	The applicant should revise the estimate of the shift in the reference temperature for its reactor vessel surveillance program.	ECGB
05.3.1-03	OPEN ITEM	The applicant should revise its capsule withdrawal schedule.	ECGB
05.3.1-04	OPEN ITEM	The applicant should clarify whether or not there will be welds in the beltline region.	ECGB
05.3.1-05	OPEN ITEM	The applicant should provide a technical justification for the use of molybdenum disulfide lubricants on the reactor vessel studs.	ECGB
05.3.1-06	OPEN ITEM	The applicant should address the environmental effects on fatigue of the reactor vessel materials.	ECGB
05.3.1-07	OPEN ITEM	The applicant should comply with the guidance in RG 1.50 in order to provide reasonable assurance that cracking of components will not occur due to residual weldment stresses.	ECGB
05.3.1-09	OPEN ITEM	The applicant should demonstrate that the Charpy upper-shelf energy of the reactor vessel beltline materials is acceptable in accordance with 10 CFR 50 Appendix G.	ECGB
05.3.1-10	OPEN ITEM	The applicant should use ASTM A262 instead of A708 for verifying non-sensitization of austenitic stainless steel.	ECGB
05.3.1-11	OPEN ITEM	The ferrite content in austenitic steel castings and weld material should conform to industry or staff guidelines.	ECGB
05.3.2-1	OPEN ITEM	The applicant should revise the predicted shift in reference temperature.	ECGB
05.3.2-2	OPEN ITEM	The applicant should clarify its intent relating to preservice hydrostatic test limits.	ECGB
05.3.2-3	OPEN ITEM	There is a factor of 2 missing from the RG 1.99, Revision 2 equation in the applicant's response to RAI Q252.5.	ECGB
05.3.2-4	OPEN ITEM	The applicant should use the same value for RT-NDT for all the times it is applied.	ECGB
05.4.1.1-1	OPEN ITEM	The applicant should justify the use of SA-508 Class 3 material in the reactor coolant pump flywheel.	ECGB
05.4.1.1-2	OPEN ITEM	The actual flywheel material should be tested for fracture toughness.	ECGB
05.4.1.1-3	OPEN ITEM	The applicant should commit to maintaining the normal operating temperature of the flywheel 56 deg C (100 F) above the RT-NDT.	ECGB
05.4.1.1-4	OPEN ITEM	The applicant should perform surface examination on all finished machined bores, keyways, splines, and drilled holes in the flywheel.	ECGB
05.4.1.1-5	OPEN ITEM	The applicant should state that the design overspeed of the flywheel is at least 10 percent above the highest anticipated overspeed in accordance with SRP 5.4.1.1.	ECGB
05.4.1.1-6	OPEN ITEM	The applicant should revise CESSAR to indicate that a surface examination will be performed on all exposed surfaces.	ECGB
05.4.1.1-7	OPEN ITEM	The applicant should perform a preservice baseline inspection on the flywheel that incorporates all of the procedures for ISI using ASME Section III acceptance criteria.	ECGB
05.4.1.1-8	OPEN ITEM	The applicant should clearly state that it will meet RG 1.14.	ECGB
05.4.2-01	OPEN ITEM	The applicant should discuss the welding qualification, fabrication processes, and inspection during fabrication and assembly, for the entire steam generator.	ECGB
05.4.2-02	OPEN ITEM	The applicant should revise the CESSAR to describe the ISI program for steam generator tubes.	ECGB
05.4.2-03	OPEN ITEM	The applicant should provide clarification for 9 items listed in DSER Section 5.4.2 that are not consistent with the EPRI secondary water chemistry guidelines.	ECGB
05.4.2-04	OPEN ITEM	The applicant should clarify 2 statements on secondary water chemistry that are listed in DSER Section 5.4.2.	ECGB
05.4.2-05	OPEN ITEM	The secondary water chemistry guidelines should contain the recently published EPRI guidelines for makeup water to the steam generators.	ECGB
05.4.2-06	OPEN ITEM	The applicant should clarify the intent and provide justification for using high strength bolting material that may be susceptable to stress corrosion cracking.	ECGB
05.4.2-07	OPEN ITEM	The applicant should provide a corrosion allowance for the 60-year plant design life and its technical justification.	ECGB

NUMBER	TYPE	TITLE	BRANCH
05.4.2-08	OPEN ITEM	The environmental effects on fatigue of steam generator materials should be addressed by the applicant.	ECGB
05.4.2-09	OPEN ITEM	The applicant should revise the CESSAR to be consistent with industry or staff guidelines on the ferrite content in austenitic stainless steels.	ECGB
05.4.2-10	OPEN ITEM	The applicant should limit the carbon content in the austenitic stainless steel steam generator materials to 0.02 percent.	ECGB
05.4.2-11	OPEN ITEM	The applicant should describe the plant design provisions that will facilitate steam generator replacement.	ECGB
06.1-01	OPEN ITEM	The applicant should follow the guidance in RG 1.50 to minimize the chance of cracking from residual stress.	ECGB
06.1-02	OPEN ITEM	INCONEL 600 is susceptible to stress corrosion cracking. The applicant should consider the use of alternate materials.	ECGB
06.1-03	OPEN ITEM	Type 304 and 316 austenitic stainless steels are susceptible to intergranular stress corrosion cracking. The applicant should consider the use of alternate materials.	ECGB
06.1-04	OPEN ITEM	The ferrite content in austenitic stainless steels should be consistent with industry or staff guidelines.	ECGB
06.1-05	OPEN ITEM	The applicant should consider dynamic strain aging on carbon steel materials.	ECGB
06.1-06	OPEN ITEM	The applicant should use ASTM A262 instead of A708 materials for ESF components.	ECGB
06.1-07	OPEN ITEM	The applicant should consider limiting the carbon content in austenitic stainless steels to 0.02 percent.	ECGB
06.1-08	OPEN ITEM	The applicant should provide a corrosion allowance for a 60-year plant design life and its technical basis.	ECGB
06.1-09	OPEN ITEM	The applicant should explicitly account for the effects of the environment in the fatigue analysis of the ESF materials.	ECGB
06.6-7	OPEN ITEM	The staff is recommending the use of ASME Section XI Subsection IWH for the CESSAR.	ECGB
09.3.2-1	OPEN ITEM	CESSAR Section 9.3.2 does not adequately describe how the post-accident sampling system will meet all of the regulatory requirements.	ECGB
09.3.2-2	OPEN ITEM	The applicant should provide additional information if it chooses to adopt the NUREG-0737 alternate requirements.	ECGB
09.3.4-1	OPEN ITEM	The applicant should provide the basis for classifying the reactor coolant pump seal injection function of the chemical and volume contro system as non-safety related.	ECGB
20.1-02	OPEN ITEM	If the applicant cannot obtain staff approval for LBB, the applicant should provide details of its analysis on assessing the effects of asymetric blowdown loads.	ECGB
20.2-02	OPEN ITEM	The applicant should describe the materials, limits on residual elements, limits on reference temp. and upper shelf impact energy and inspection requirements for RV supports.	ECGB
20.2-03	OPEN ITEM	The applicant should provide the estimated 60-year neutron fluence level at the reactor vessel supports.	ECGB
20.2-04	OPEN ITEM	The applicant should describe its procedures for estimating the extent of irradiation embrittlement of the reactor vessel supports and provide the results.	ECGB
20.2-05	OPEN ITEM	The applicant should provide additional information on its fracture mechanics analysis of the reactor vessel supports, including assumptions and acceptance criteria.	ECGB
20.2-06	OPEN ITEM	If the fracture mechanics analysis for the reactor vessel supports is based on LBB assumptions, the applicant should provide technical justifications.	ECGB
20.2-08	OPEN ITEM	The applicant should reference GL 91-17 when responding to Generic Issue 29.	ECGB
20.2-09	OPEN ITEM	The pertinent information and requirements in NRC bulletins, generic letters, and information notices issued regarding Generic Issue 29 should be factored into the CESSAR.	ECGB
20.2-11	OPEN ITEM	The staff requires additional information regarding the applicant's response to Generic Issue 79.	ECGB
20.2-16	OPEN ITEM	The applicant must specifically address the issues covered in GSI-113.	ECGB
20.2-18	OPEN ITEM	The applicant should state in the CESSAR that the inservice inspection of the steam generator tubes will be based on improved eddy current testing techniques.	ECGB
02.6-1	SITE PARAM	The COL applicant must verify that site specific date is bounded by CESSAR Table 2.0-1.	ECGB

ECGB

Confirmatory Item 3.7.2-3

The applicant should clarify CESSAR Section 3.7.2-11 to state how the additional eccentricity of 5 percent of the maximum building dimension will be applied.

Response:

The additional eccentricity of 5 percent of the maximum building dimension will be accounted for by increasing the forces and moments of the static finite element models of structures by appropriate factors representing the 5 percent eccentricity. Section 3.7.2-11 of the CESSAR will be revised, as attached, to reflect this.

An additional eccentricity of 5% of the maximum building dimension is input as torsional factors applied to the static finite element structural model to calculate element forces & moment

3.7.2.11 Methods Used To Account for Torsional Effects

The mathematical models used in analysis of Seismic Category I systems, components, and piping systems include sufficient mass points and corresponding dynamic degrees-of-freedom to provide a three-dimensional representation of the dynamic characteristics of the system. The distribution of mass and the selected location of mass points account for torsional effects of valves and other eccentric masses.

The structural models used for Seismic Category I systems are constructed with elements containing 6 degrees of freedom per node, incorporating torsional effects into the models. Torsional effects are also accounted for in the building models used to generate floor response spectra. ~~An additional eccentricity of 5% of the maximum building dimension was included to account for accidental torsion.~~

3.7.2.12 Comparison of Responses

With the exception of the surge line, the time-history method is used for structural analysis of the NSSS and the associated building structures. Therefore, responses obtained from the response spectrum and time-history methods are not compared. I

3.7.2.13 Methods for Seismic Analysis of Dams

If applicable for the site, analyses of safety-related dams will be performed taking into account appropriate factors such as the behavior of dam material under horizontal and vertical seismic loadings. Dam soil-structure-interaction effects will be considered.

3.7.2.14 Determination of Safety-Related Structure Overturning Moments

The overturning moments and base shears due to seismic forces for Category I structures are determined using the response spectrum method of analysis. The seismic motion is input to the structural models in three independent orthogonal directions. The overturning moments for shell structures are automatically included in the analysis of this type of structure.

3.7.2.15 Analysis Procedure for Damping

For modal superposition method, composite modal damping values are used for structures with components of different damping characteristics. The composite modal damping values are based on weighting the damping factors according to the mass or the stiffness of each element. For the mass proportional damping, formulation is as follows:

ECGB

Open Item 2.5-1

The applicant should use envelope response spectra for design analysis of seismic Category I structures.

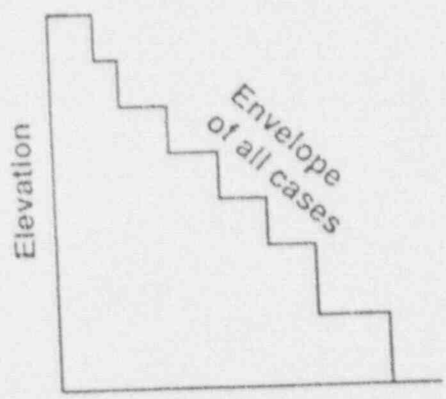
Response:

The following design criteria will be applied to the design of the System 80+ structures, piping and components. The design criteria were developed with the purpose of including adequate conservatism in the standard design.

Design Criteria for System 80+ Category I Structures and Components:

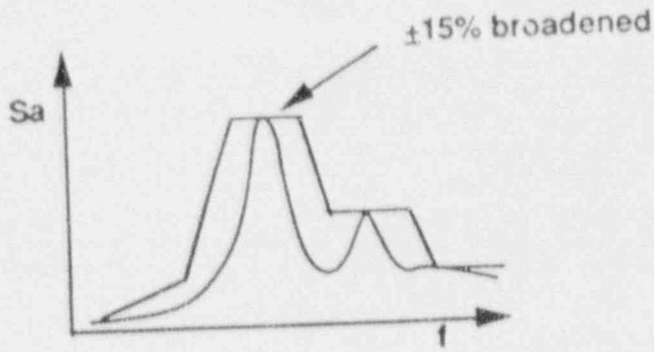
1. Design of Seismic Category I Structures

   The envelope of floor shears, axial forces and moments from all soil cases will be used to design the Category I structures (except of the Steel Containment Vessel). The shears, axial forces and moments will be applied statically to a detailed model of each structure in order to obtain local stresses for the design of all walls, columns, beams, slabs and other structural elements.



   Because the Steel Containment Vessel was modeled in the seismic SSI analyses with a simpler model (compared to the finite element model which is used for its detailed design), a special procedure will be followed to compute the stresses on the SCV due to the seismic loads. A dynamic analysis of the detailed finite element model will be performed using ±15% broadened spectra (translational and rotational) corresponding to elevation +91.75 of Interior Structure (SCV support location) for all the governing cases.

ECGB

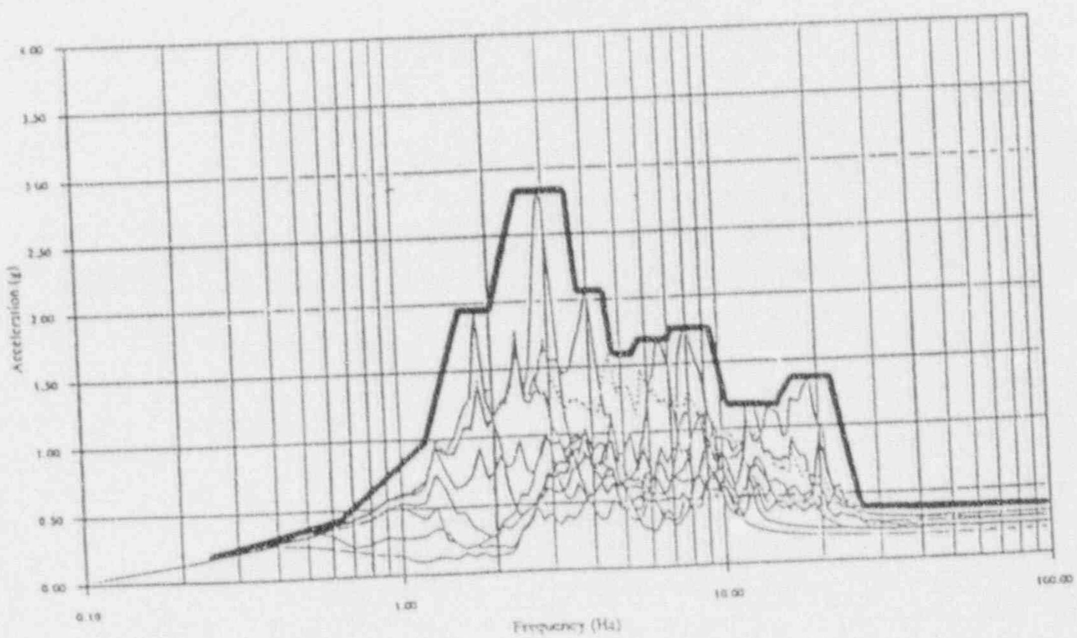


2. Design of components and piping

The design of the System 80+ piping and in-structure components will be performed using one of the three options presented below.

Option 1:

This is the first option that will be used in the design process. According to this option, broadening of the raw response spectra by ±15% will be initially performed for all soil cases. The envelope of the broadened spectra of all soil cases will then be directly used in the design. The objective is to exercise this option for as many of the piping and components as possible. However, it is recognized that excessive conservatism may be introduced in the design of some piping or components, in which case Options 2 or 3 will be applied.



ECGB

Option 2:

According to this option, broadening of the raw response spectra by ±15% will be performed for all soil cases. Grouping of the sites will then be performed according to site categories (a maximum of 2 or 3 categories will be selected, e.g. soft sites, medium sites, hard sites). Following the site grouping, and envelope of the broadened spectra for each category of sites will be developed. The envelope of spectra of each category will then be used in the design process.

Option 3:

According to this option, frequency shifting of the raw response spectra by ±15% will be performed for all soil cases. The resulting spectra from the frequency shifting of each individual soil case will then be used in the design precess. Since the soil cases cover a wide range of sites, it is judged that the design process using Option 3 contains adequate conservatism.

ECGB

Option 2:

According to this option, broadening of the raw response spectra by ±15% will be performed for all soil cases. Grouping of the sites will then be performed according to site categories (a maximum of 2 or 3 categories will be selected, e.g. soft sites, medium sites, hard sites). Following the site grouping, and envelope of the broadened spectra for each category of sites will be developed. The envelope of spectra of each category will then be used in the design process.

Option 3:

According to this option, frequency shifting of the raw response spectra by ±15% will be performed for all soil cases. The resulting spectra from the frequency shifting of each individual soil case will then be used in the design precess. Since the soil cases cover a wide range of sites, it is judged that the design process using Option 3 contains adequate conservatism.

ECGB

Open Item 2.5.2.5.1-1

The time histories associated with CMS2 do not satisfy SRP 3.7.1 acceptance criteria for 7-percent damping.

Response:

The vertical CMS2 rock outcrop time history is below the 7% target spectrum in the 3-10 Hz range and the 20 Hz range. However, for the vertical seismic analysis, the rock outcrop motion was convoluted to the soil surface in a very conservative manner (as evidenced by the high amplitudes of the vertical free-field surface spectra). Therefore, the surface and foundation level spectra contain adequate conservatism.

In addition, the CMS1 vertical motion (0.3g PGA) envelops the CMS2 vertical rock outcrop motion (0.2g PGA) at almost all frequencies, as shown in the attached Figure 3.7.1-1.1.

Based on the above, it is judged that, although the CMS2 vertical spectra strictly do not meet the SRP 3.7.1 criteria for 7% damping, the analysis results are not affected.

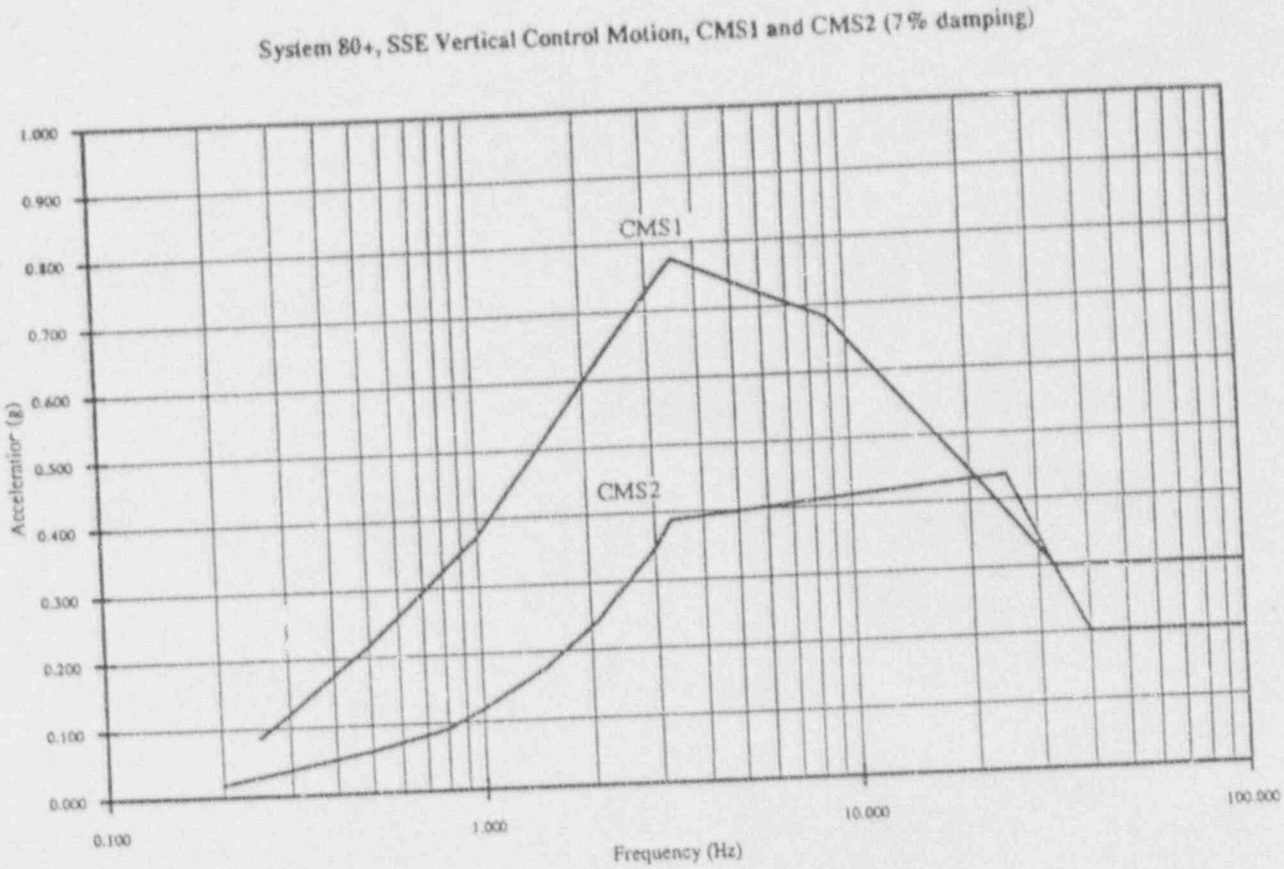


Figure 3.7.1-1.1 - Comparison between CMS1 and CMS2 vertical rock spectra (7% Damping)

ECGB

Open Item 2.5.2.5.1-2

The problem of significant "valley" occurring in the foundation spectra presented in a previous meeting must be addressed.

Response:

The reduction of spectra at the free-field foundation level occurs at a frequency corresponding approximately to that of the sublayer between the ground surface (where the CMS1 motion is specified) and the foundation level. To overcome this reduction, the deconvolution is done considering a wide range of modulus values of the soil comprising this layer.

ECGB

Open Item 2.5.2.5.1-3

The CESSAR should be revised to include CMS1 and CMS3.

Response:

The CESSAR is being revised to include the CMS1 and CMS3 time histories, the corresponding spectra match at 1, 2, 5 and 7% damping and the CMS1 and CMS3 Power Spectral Densities. Figures 2.5.2.5.1-4.1 through 2.5.2.5.1-4.11 show the information that will be included in the CESSAR. All time histories meet the SRP criteria for development of artificial time histories. CMS1 meets the SRP 3.7.1, App. A requirements for Power Spectral Density (PSD) for the horizontal motions, as shown in Figure 2.5.2.5.1-4.4. In addition, it is shown in Figures 2.5.2.5.1-4.5 and 2.5.2.5.1-4.8 through -11 that the PSD of the vertical CMS1 time history and all three time histories of the CMS3 motion have adequate PSD's with no deficiency in power at any frequency range.

CJ 2.5.2.5.1-3

ECGG

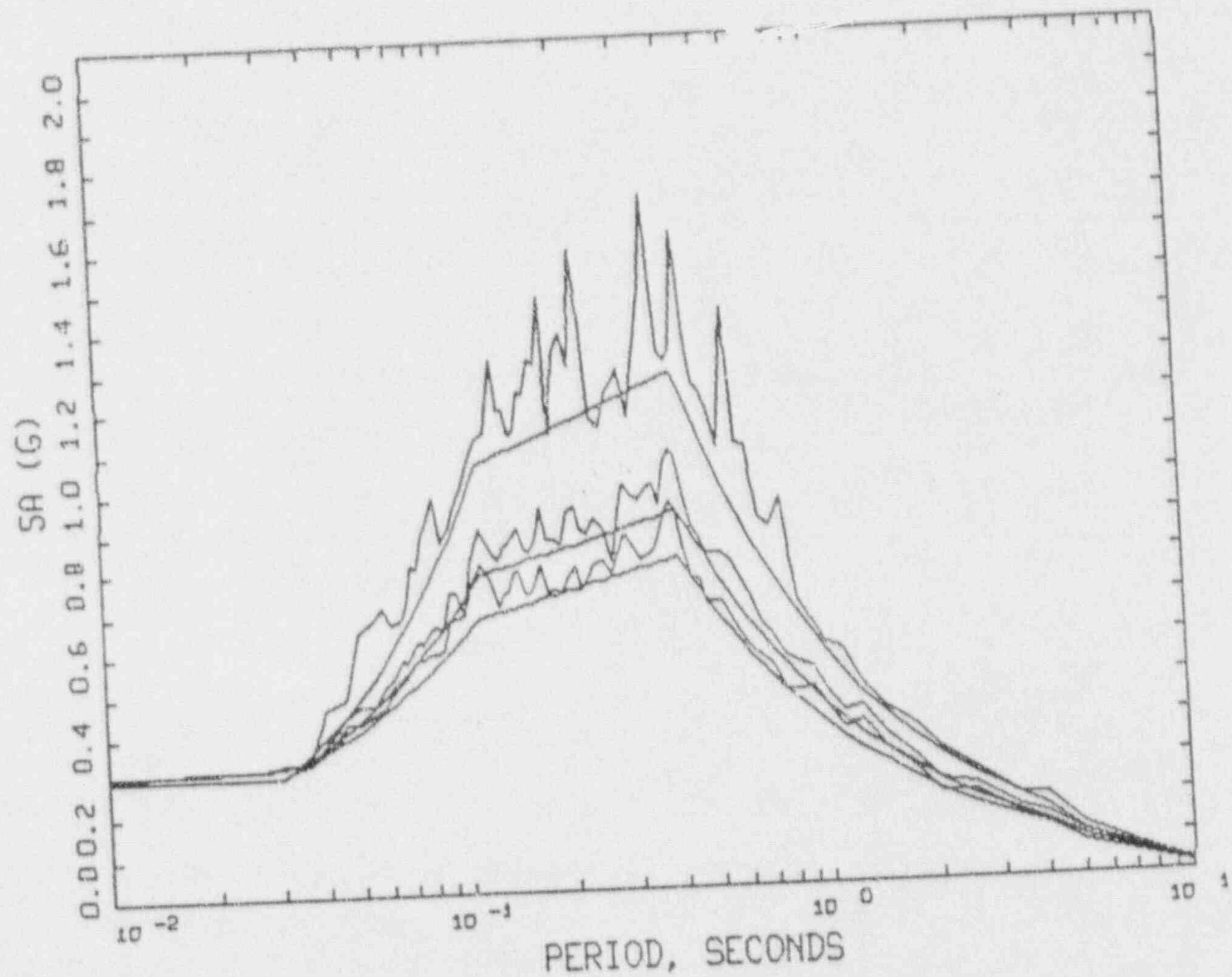


Figure 2.5.2.5.1-4.1 - Spectral match of time history and CMS1 target spectrum (Horizontal Direction, Time History 1, Damping 1, 2, 5, 7%)

OI 2.5.2.5.1-3

ECGB

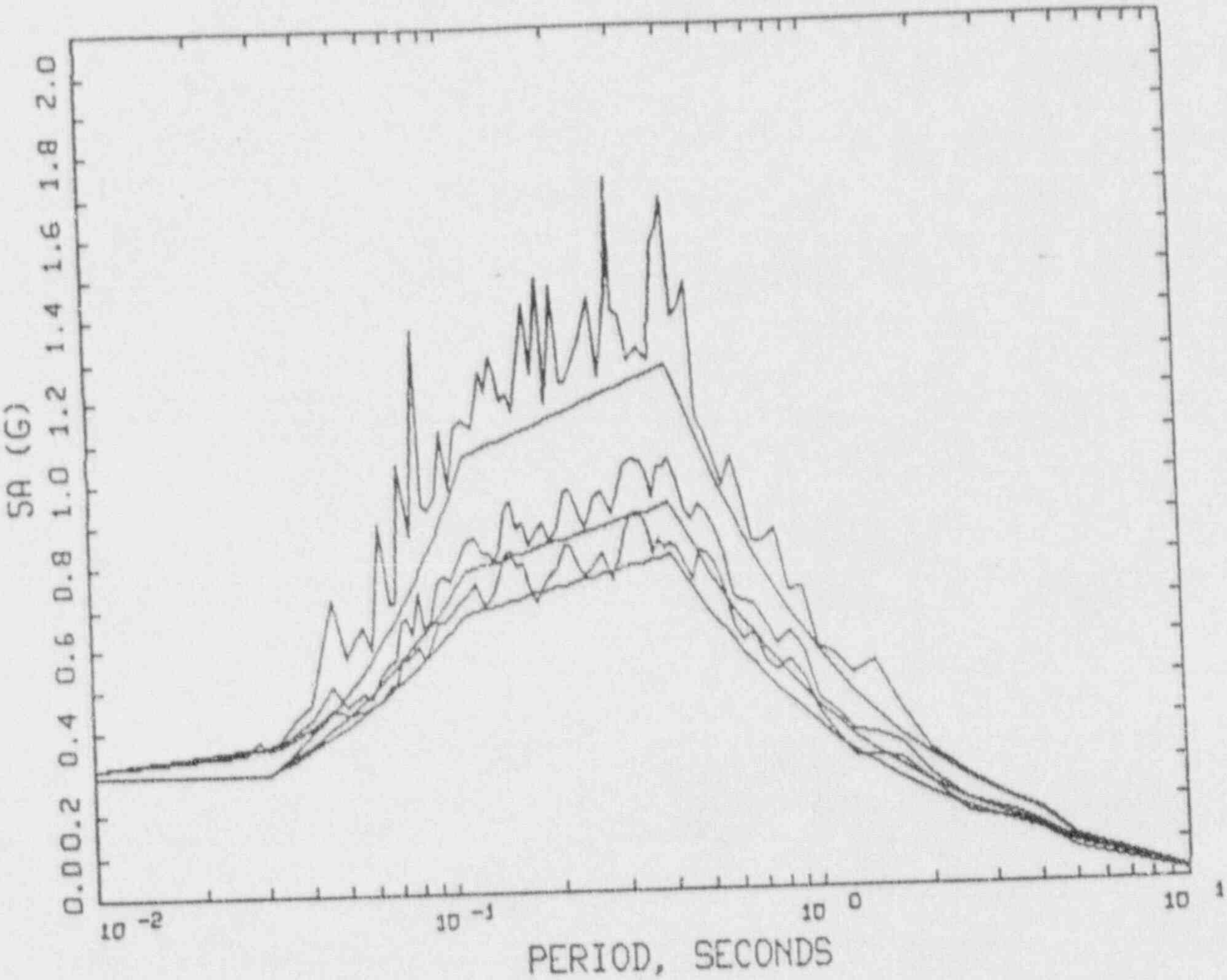


Figure 2.5.2.5.1-4.2 - Spectral match of time history and CMS1 target spectrum (Horizontal Direction, Time History 2, Damping 1, 2, 5, 7%)

OI 2.5.2.5.1-3

ECGB

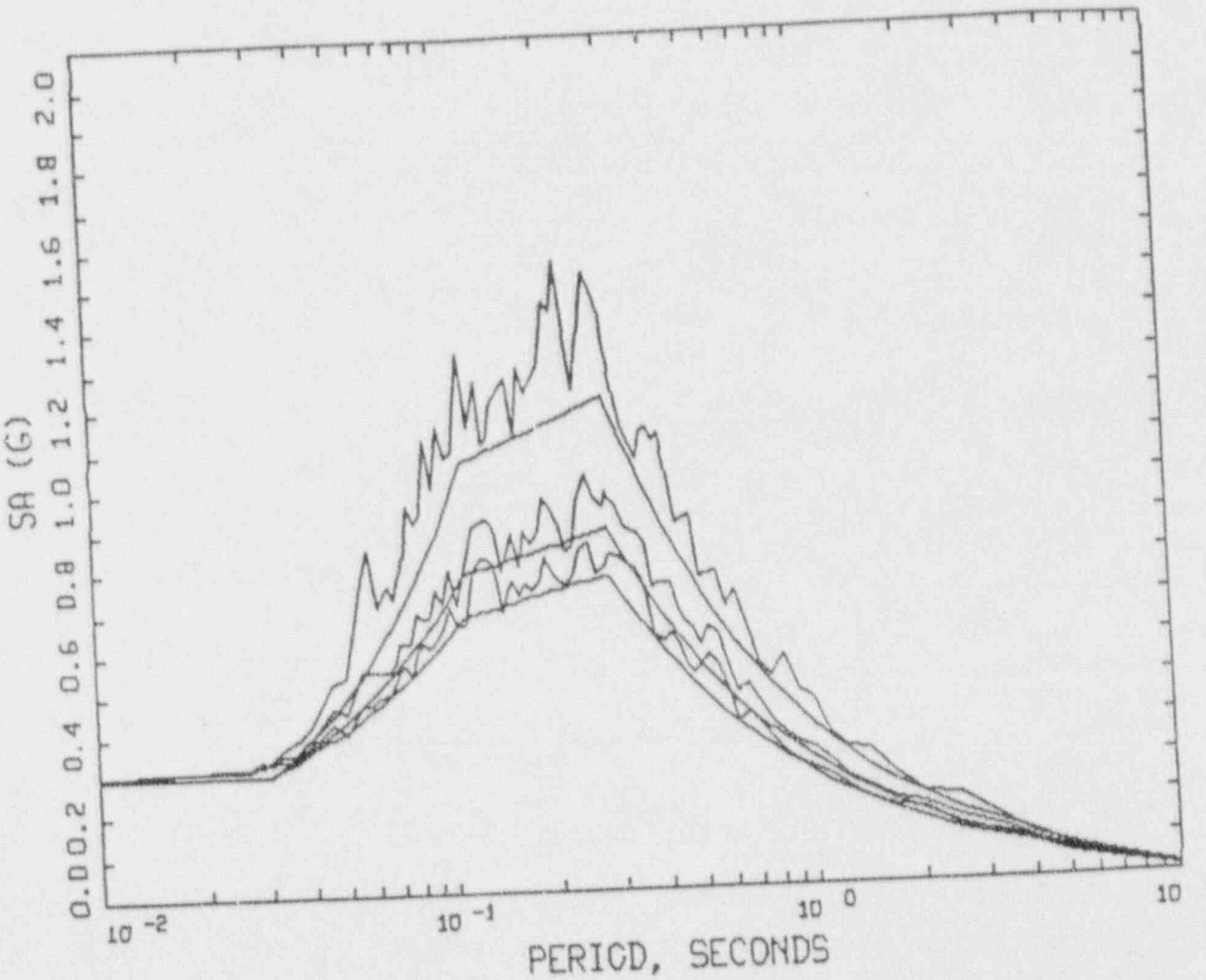


Figure 2.5.2.5.1-4.3 - Spectral match of time history and CMS1 target spectrum (Vertical Direction, Damping 1, 2, 5, 7%)

ECGB

OI 2.5.2.5.1-3

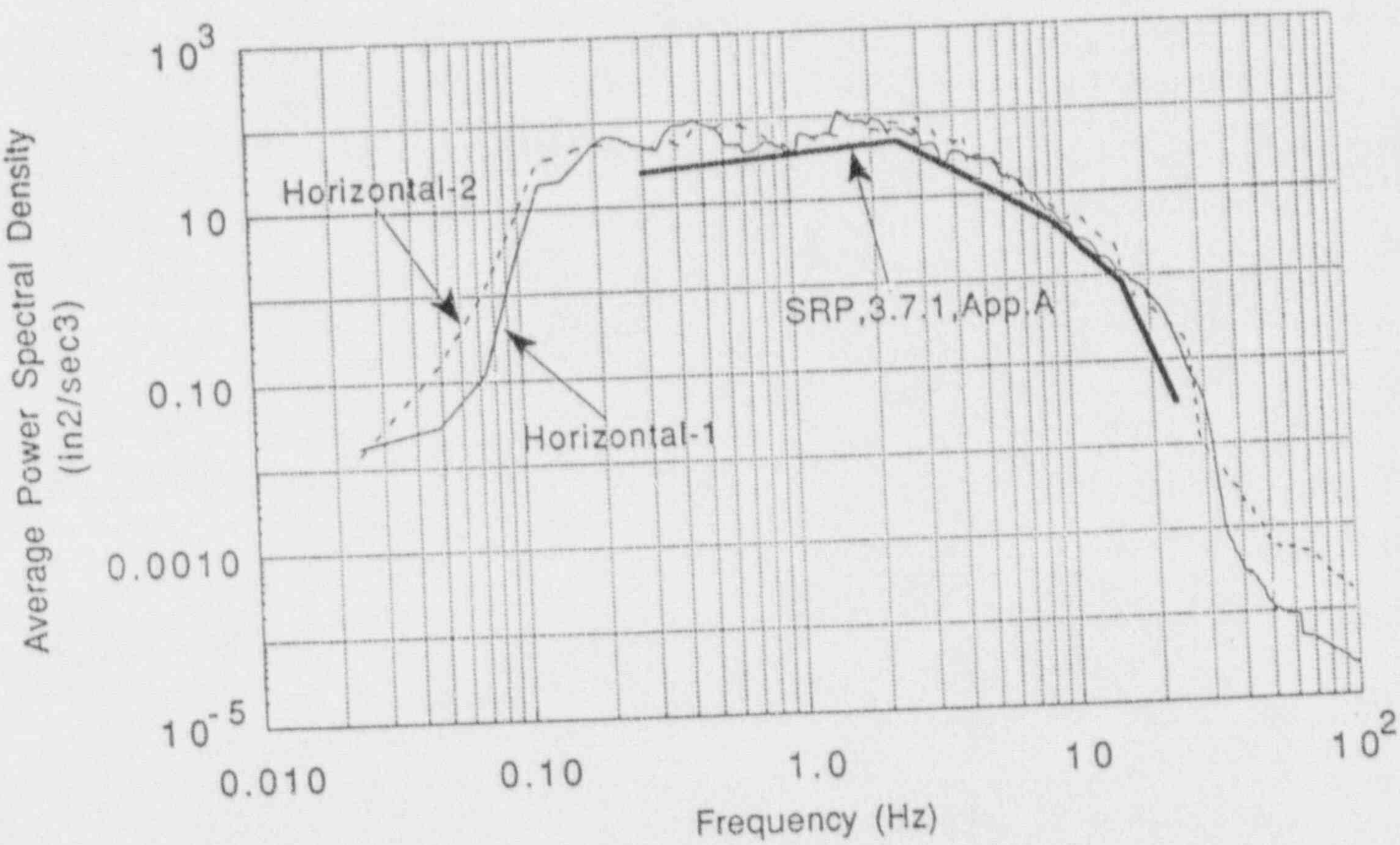


Figure 2.5.2.5.1-4.4 - Power Spectral Density of CMS1 Motion (Horizontal Direction, Time Histories 1 and 2)

OI 2.5.2.5.1-3

ECGB

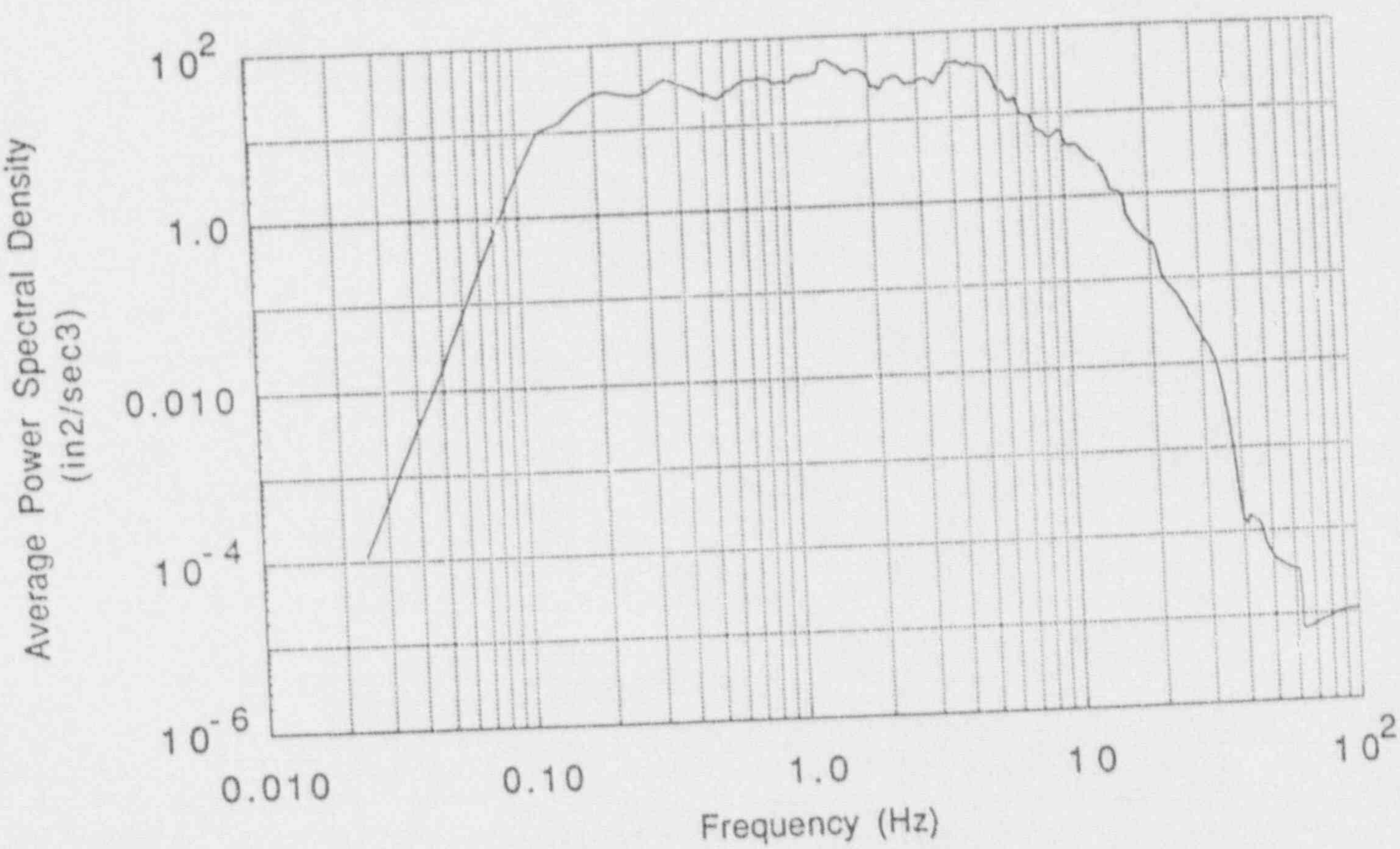


Figure 2.5.2.5.1-4.5 - Power Spectral Density of CMS1 Motion
(Vertical Direction)

OI 2.52.5.1-3

ECGB

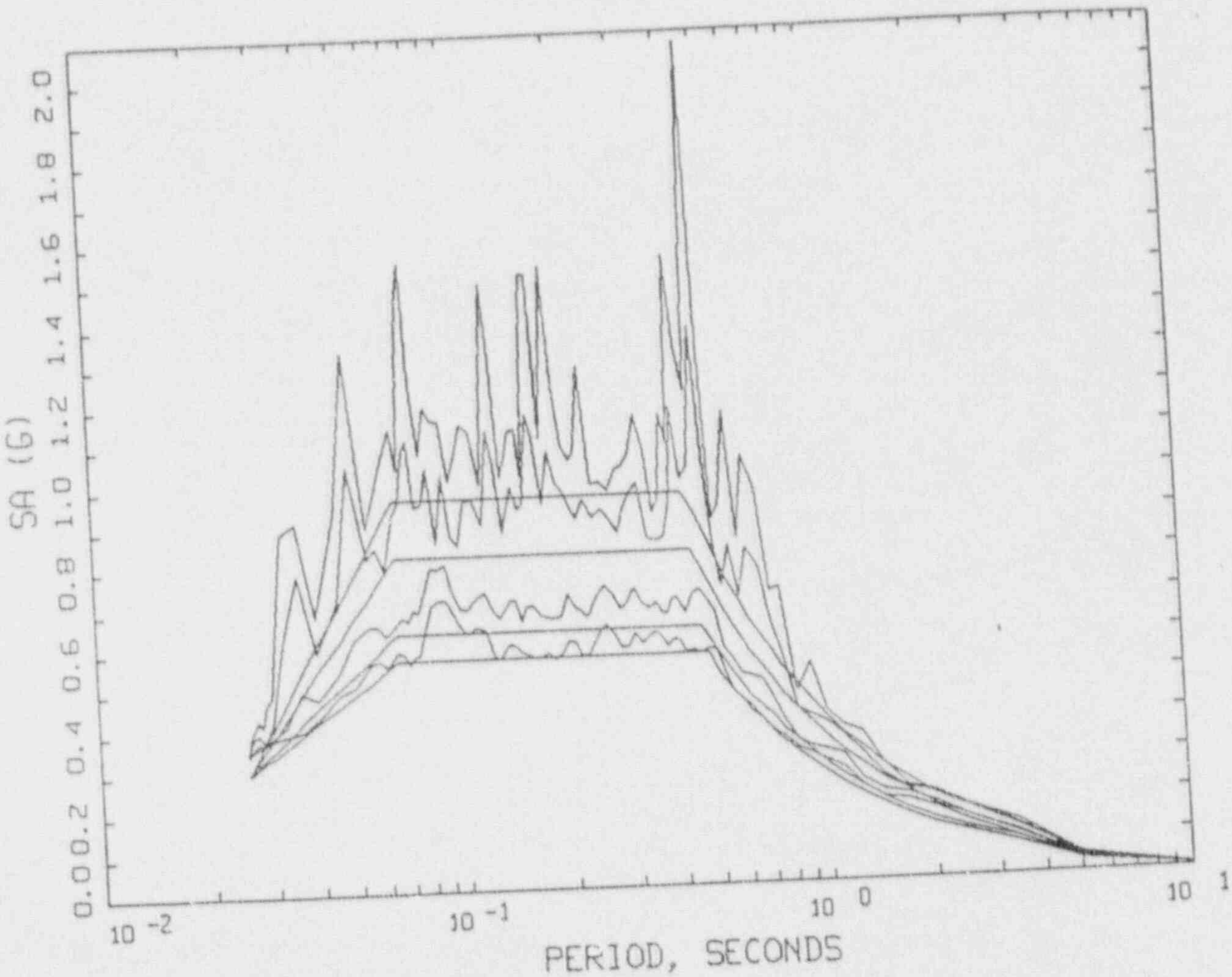


Figure 2.5.2.5.1-4.6 - Spectral match of time history and CMS3 target spectrum (Horizontal Direction, Time History 1, Damping 1, 2, 5, 7%)

OI 2.5.2.5.1-3

ECGG

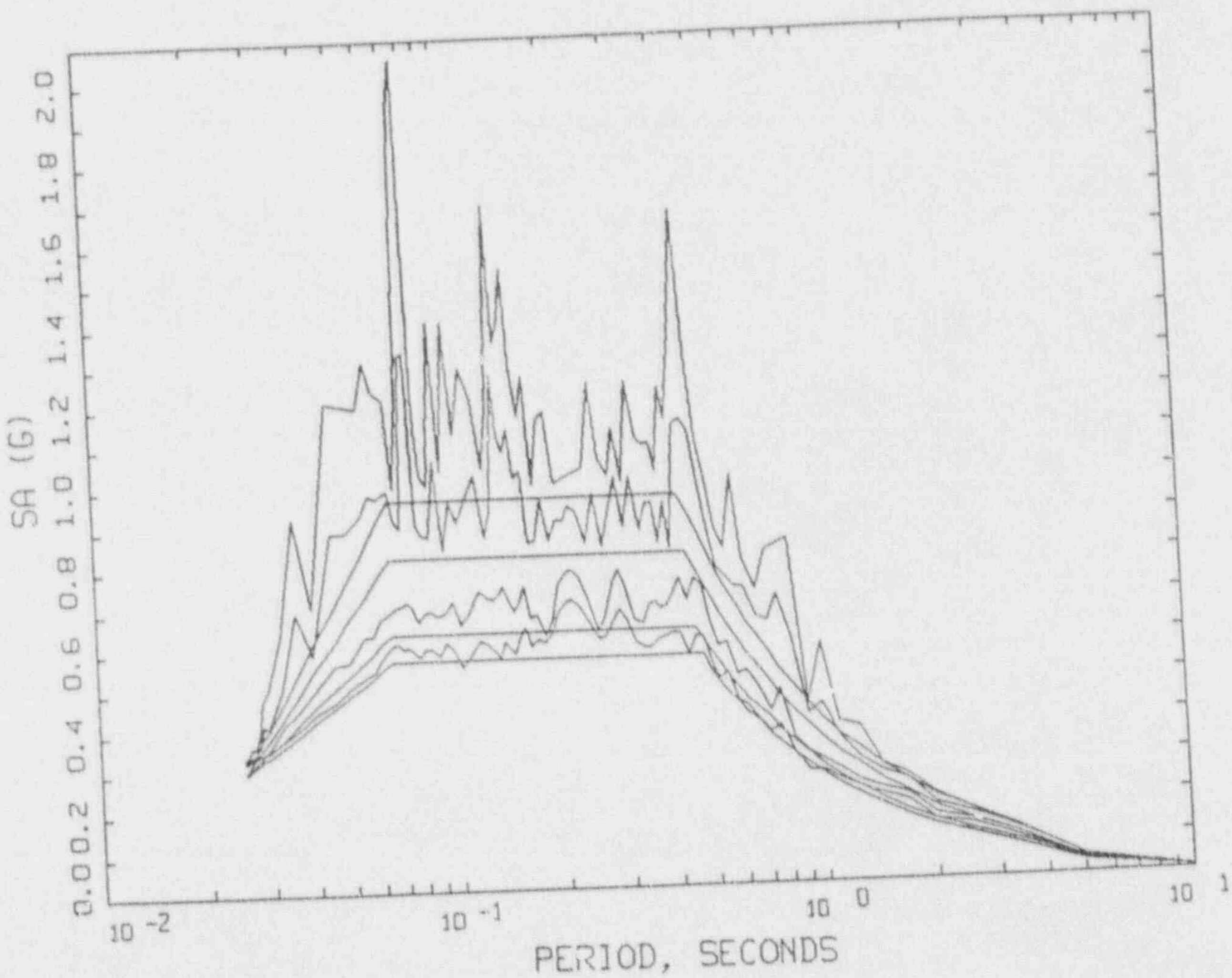


Figure 2.5.2.5.1-4.7 - Spectral match of time history and CMS3 target spectrum (Horizontal Direction, Time History 2, Damping 1, 2, 5, 7%)

OI 2.5 2.5.1-3

ECGS

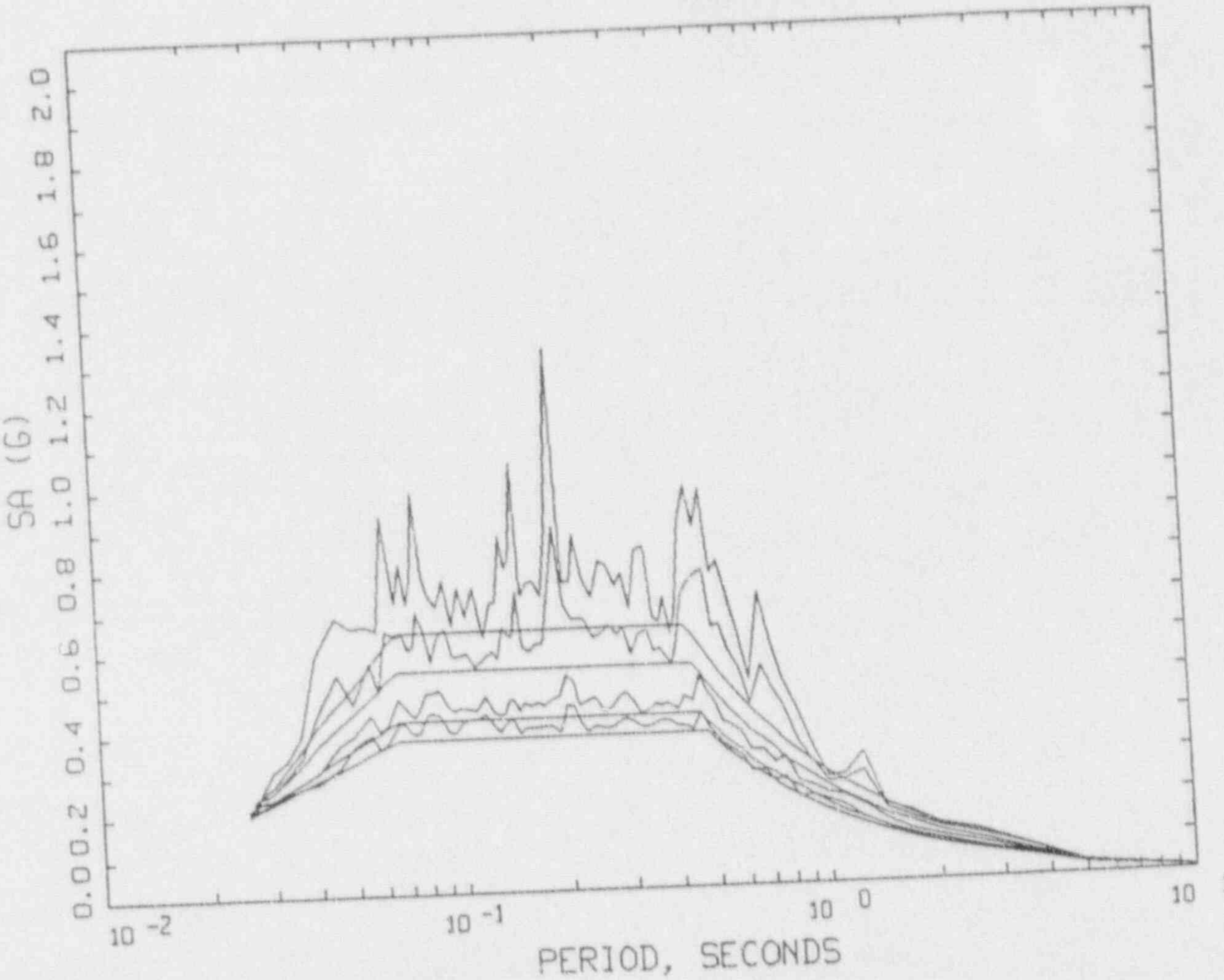


Figure 2.5.2.5.1-4.8 - Spectral match of time history and CMS3 target spectrum (Vertical Direction, Damping 1, 2, 5, 7%)

UI 2.5 2.5.1-3

ECGB

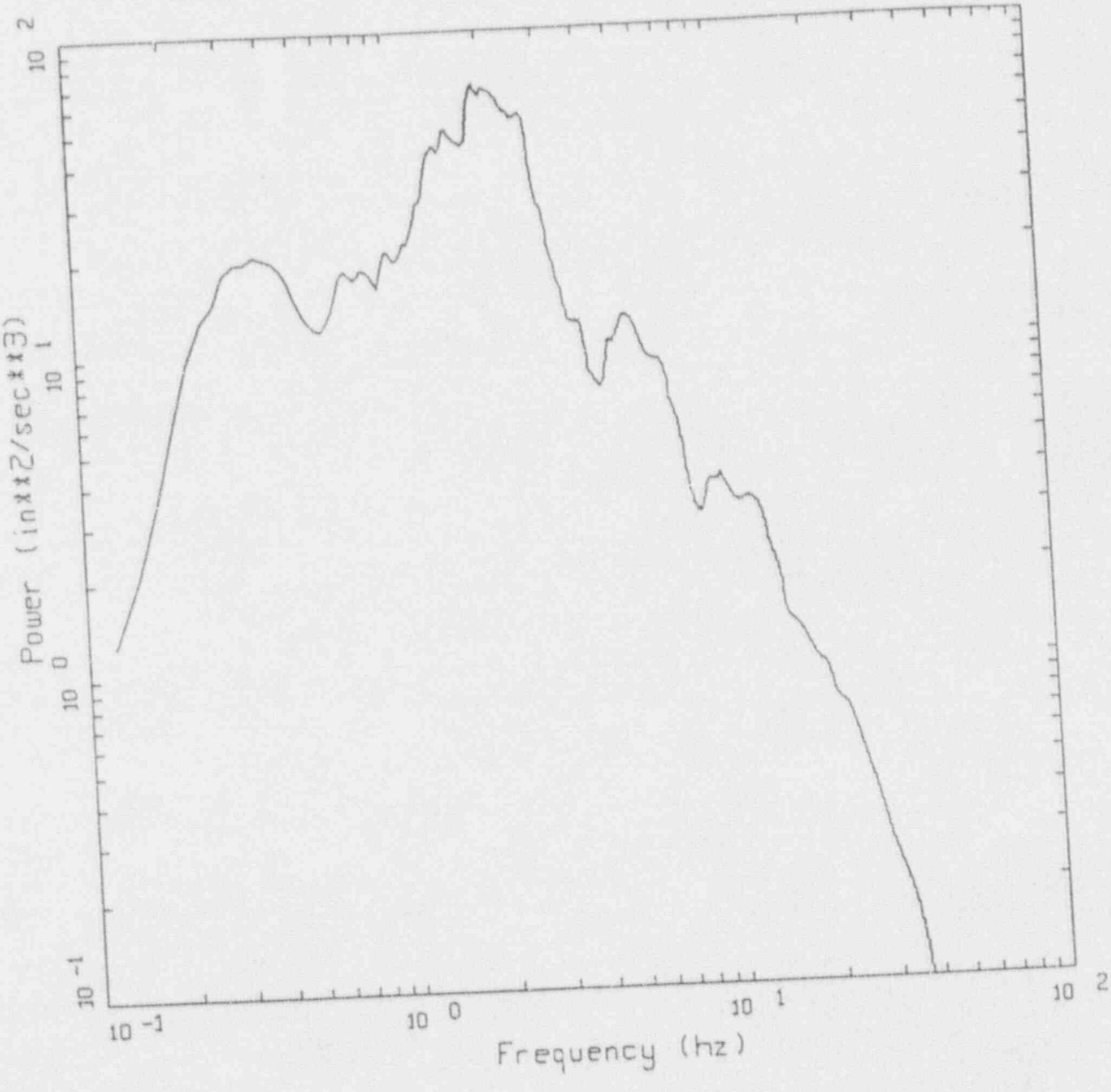


Figure 2.5.2.5.1-4.9 - Power Spectral Density of CMS3 Motion (Horizontal Direction, Time History 1)

OI 2.5.2.5.1-3

ECGB

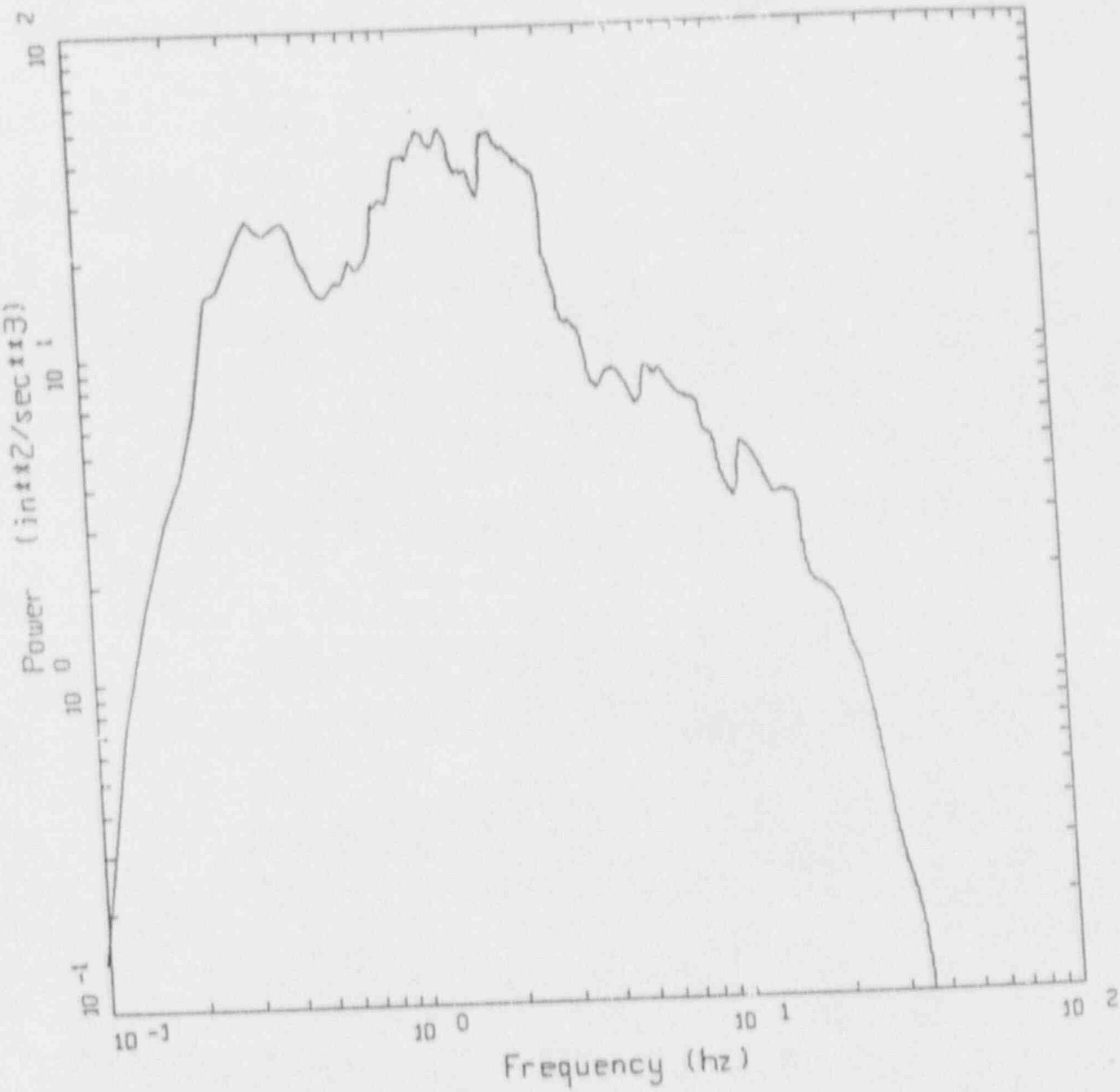


Figure 2.5.2.5.1-4.10 - Power Spectral Density of CMS3 Motion (Horizontal Direction, Time History 2)

OI 2.5.2.5.1-3 ECGB

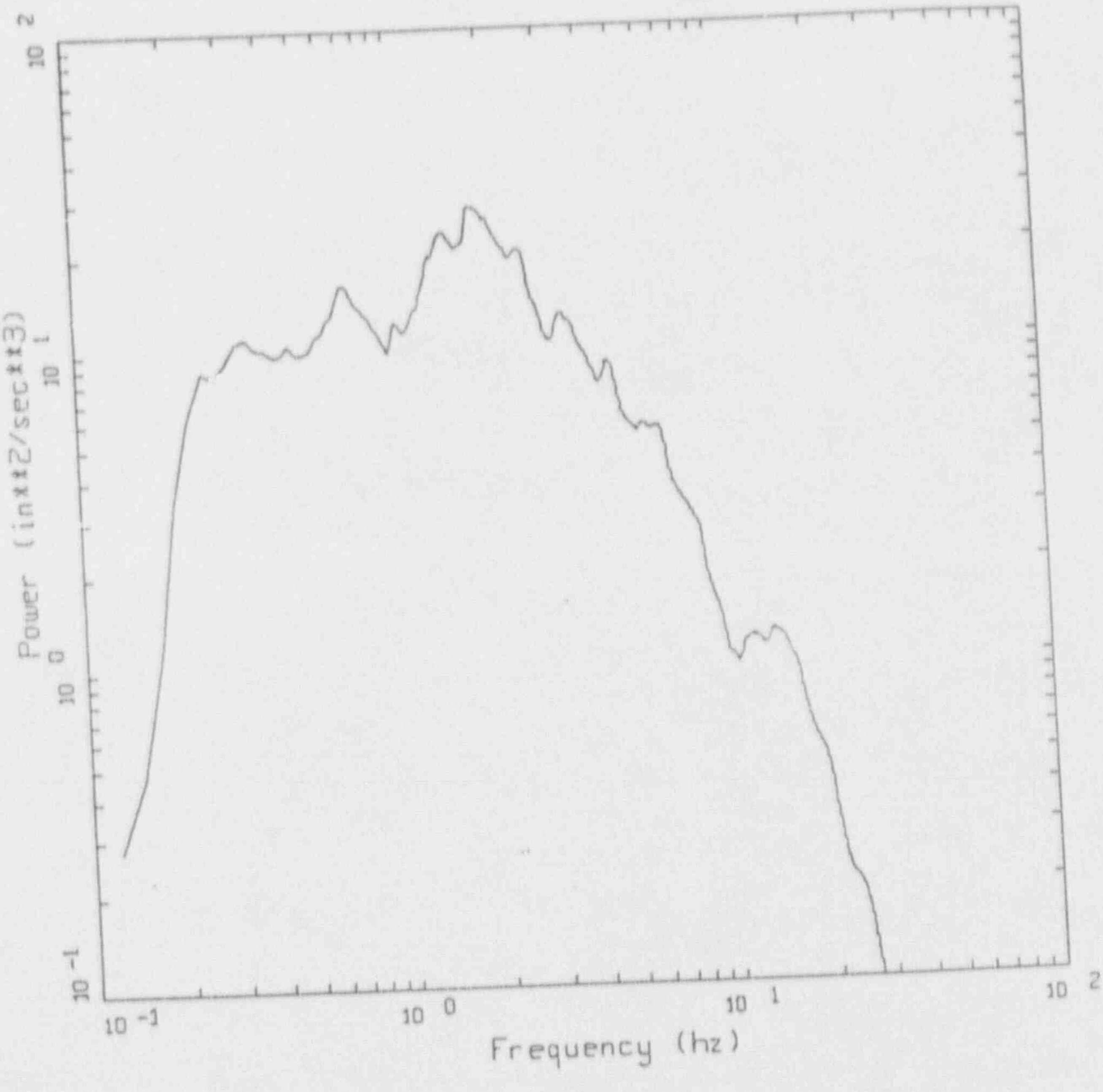


Figure 2.5.2.5.1-4.11 - Power Spectral Density of CMS3 Motion (Vertical Direction)

ECGB

Open Item 2.5.2.5.1-4

The staff must review the applicant's formal discussion in CESSAR on how CMS1 will be used.

Response:

The analytical procedure that will be followed to generate in-structure response spectra and design loads for the System 80+ structures and components using CMS1 is shown in the schematic of the following page. CMS1 is applied at the free-field ground surface of each generic soil profile. The strain-iterated soil properties of each profile where CMS1 is applied are derived from the soil analysis using the CMS2 motion. Therefore, the foundation impedances and the transfer functions at the structures are the same for the analyses of all three motions, CMS1, CMS2 and CMS3.

The usage of the in-structure response spectra for the design of the System 80+ structures and components is discussed in the response to Open Item 2.5-1. The procedures outlined in the response to Open Item 2.5-1 are applicable to all three motions, CMS1, CMS2 and CMS3.

OI 2.5.35.1-4

FCGB

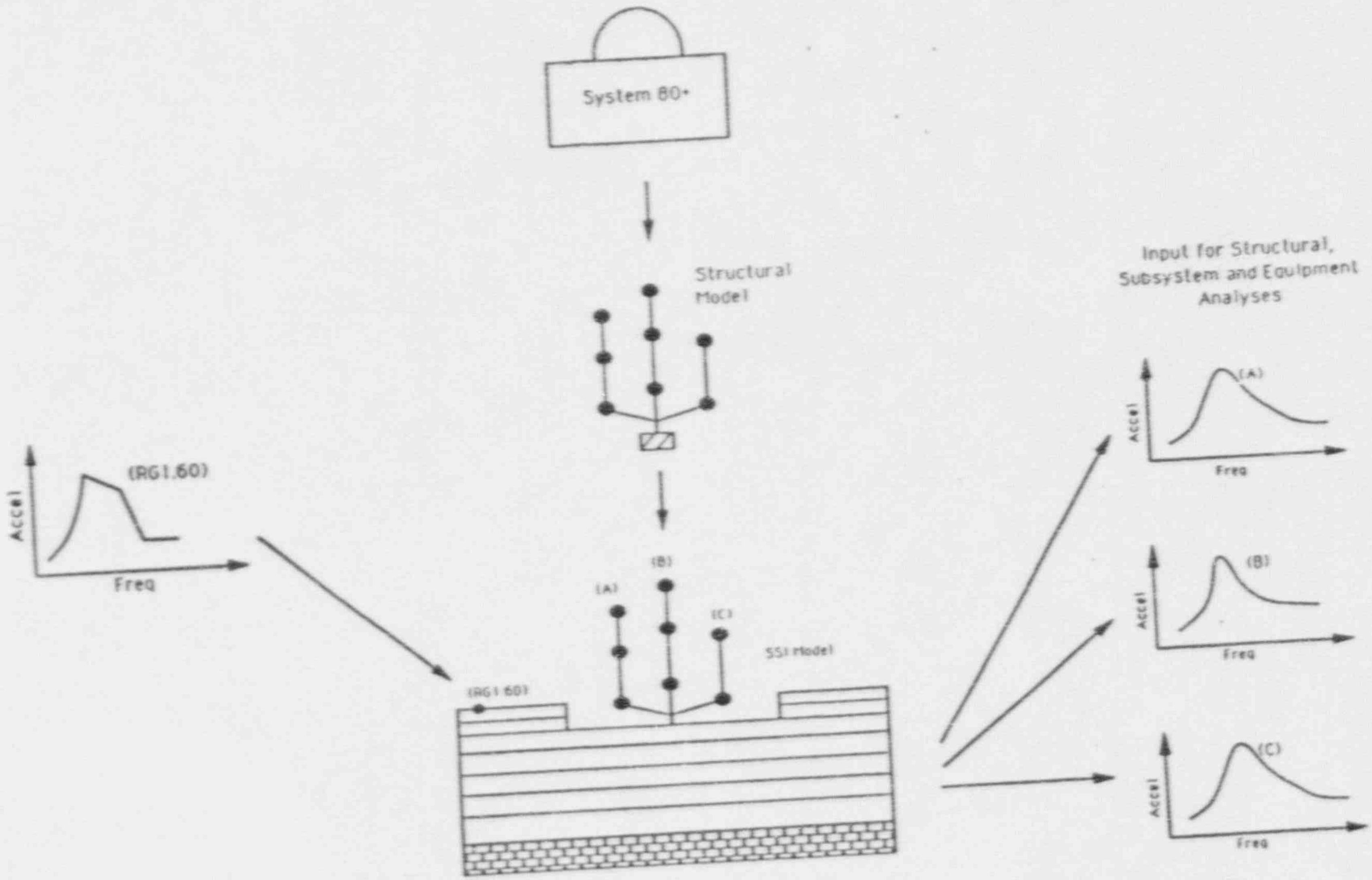


Figure 3.2 - Outline of Application of Control Motion CMS1 in SSI Analyses

ECGB

Open Item 2.5.2.8-1

The applicant should address soil properties associated with compression waves.

Response:

For the analyses involving vertical motions, the constrained modulus for each sublayer was used. This constrained modulus was calculated using the strain-compatible shear modulus (obtained from the analyses involving the horizontal ground motions) and an assigned Poisson's ration of 0.4.

Initially, a constant compression wave velocity (equal to that of water) was used throughout the soil profile. The results were not significantly different from those calculated using the procedure outlined above.

ECGB

Open Item 3.7.1-1

The time histories of CMS2 do not satisfy the SRP 3.7.1 acceptance criteria for 7-percent damping.

Response:

See the response to Open Item 2.5.2.5.1-1.

ECGB

Open Item 3.7.1-2

The applicant should submit the time histories and their corresponding response spectra associated with CMS1 and CMS3.

Response:

See the response to Open Item 2.5.2.5.1-3.

ECGB

Open Item 3.7.2-5

The applicant should demonstrate that issues addressed in ARP section 3.7.3 Paragraph II.1.a.(iii) on reducing large static models, have been satisfactorily considered.

Response:

The dynamic modeling of the hot leg piping was based on matching significant frequencies as determined by a multi-mass model for each hot leg. Each hot leg was modeled using 9 mass points with 27 dynamic degrees of freedom (DDOF). The fundamental frequencies are 178.4 Hz and 183.1 Hz in the lateral directions and 483.6 Hz in the axial direction. Use of only one mass point with 2 DDOF is sufficient to match the fundamental frequencies in the lateral direction. Matching higher order frequencies would serve no purpose (the second modes in the lateral directions are at 500 Hz) since inclusion of these modes will not change the response of the hot leg piping to seismic excitation. It is worth noting that referring to the hot leg as a 'piping run' is a misnomer, since the hot leg is 167 inches long and has a 49 inch O.D., for a length of diameter ratio of only 3.4.

ECGB

Open Item 3.7.4-1

The applicant should clarify CESSAR Section 3.7.4.4 by requiring the plant operating procedures to define "significant exceedance" of design earthquake level of interest.

Response:

CESSAR DC subsection 3.7.4.4 will be revised as attached, and Reference 13 will be added to the reference list for Section 3.7.

INSERT 3.7.4.3 here

DELETE

Observed values which exceed OBE acceleration threshold on the time-history accelerograph are indicated by an alarm light during playback. Further analysis is needed to authenticate structural loads and to evaluate observations via the structural response-seismic model. An observation that exceeds the SSE acceleration threshold is validated in a similar manner with the structural response-seismic model. When evaluated accelerations exceed OBE threshold values, the reactor is manually shutdown. The alarm lights and the recorder data are available simultaneously with the seismic event.

3.7.4.4 Comparison of Measured and Predicted Responses I

DELETE

The computer program which evaluates the time-history data computes the maximum response accelerations at various points of the model. The observed response spectra are compared with the computed response spectra. Agreement between the observed response spectra and the computed response spectra from the time-history inputs demonstrates the adequacy of the analytical model. The magnitude of actual forces at various structural locations is then compared to design values to authenticate the capability of the plant to continue operation without undue risk to the health and safety of the public.

add INSERT 3.7.4.4 here

INSERT FOR SECT 3.7.4.4 SHEET 1 OF 2

In the event of an earthquake, a two phase comparison of measured to predicted response is made to determine whether or not OBE exceedance has occurred. This procedure is followed by an evaluation of the operability of the instrumentation used to collect data for the seismic event. The above evaluations will be made within 4 hours of the seismic event, even if the plant automatically trips off-line during the earthquake.

The data used to determine whether OBE ground motion has been exceeded is available from the instrumentation located at the ground surface in the free field. Other instrumentation provides data at the foundation level of the containment structure and at other Seismic Category I structures and equipment. As described below, this data is used in potential damage assessment of those components and structures, provided that additional evaluation is warranted by the results of the OBE exceedance assessment.

The first comparison made is between the measured response spectra, as determined by the accelerographs and response spectrum recorders receiving data from the appropriate sensors, and a criterion value, defined as the greater of the design response spectra or .2 g's. These comparisons are made at selected frequencies in the 2 to 10 Hz range (eg, 8 frequency points evenly spaced on a logarithmic scale, per Reference 13) for all earthquake directions. The response spectrum check is performed at the 5% damping level, and OBE exceedance is considered to have possibly occurred if, in accordance with Reference 13, one measured spectral ordinate for any one of the earthquake directions exceeds the criterion value, and one additional measured spectral ordinate exceeds 2/3 of the

INSERT FOR SECT 3.7.4.4 (SHEET 2 OF 2)

criterion value. If this response spectrum check indicates that OBE exceedance has not occurred, an additional measured versus predicted response spectrum check in the 1 to 2 Hz range is performed based on OBE spectral velocities. The criteria values for this check are the greater of the predicted spectral velocity at the given frequencies or a value of 6 inches per second. If both types of response spectrum checks fail to indicate OBE exceedance, then no further comparisons need to be made.

If either spectrum comparison indicates possible OBE exceedance, a second comparison is made involving computation of the cumulative absolute velocity (CAV). This calculation is performed by dedicated software and hardware at the site. If the calculated CAV is greater than 0.16 g-sec, the CAV limit has been exceeded and OBE exceedance has occurred.

If OBE exceedance occurs, potential plant damage assessments are made using the reactor building dynamic analysis model. The measured free field seismic data is used to create time-history input for the analytical model, and predicted maximum response accelerations are computed at locations in the model corresponding to the remaining locations at which measured response spectra data has been obtained during the earthquake. These locations correspond to the locations of the major Seismic Category I structures and equipment. The measured response spectra are then compared to the computed response spectra. Agreement between the measured and the computed response spectra demonstrates adequacy of the analytical model. The magnitudes of the actual forces at various structural locations are then computed and compared to the design values to authenticate the capability of the plant to either continue or resume operation without undue risk to the health and safety of the public.

REFERENCES FOR SECTION 3.7 (continued)

12. NUREG-1061, VOLUME 4, U.S. Nuclear Regulatory Commission, Report of the U.S. Nuclear Regulatory Commission Piping Review Team, April 1985.

13. EPRI Report No. NP-5930, "A Criterion for Determining Exceedance of the OBE", July 1988.

ECGB

Open Item 4.5.1-1

In CESSAR Section 4.5.1.1, the applicant states that Inconel 600 materials may be used in the fabrication of the (CEDM) motor housing assembly. Operating experience indicates that Inconel 600 is susceptible to cracking. The applicant should consider alternate materials that are resistant to cracking.

Response:

The area surrounding the CEDM motor housing operates at a calculated maximum temperature of less than 450 °F, which is considerably less than the temperature at which Inconel 600 would be expected to crack during the design life. ABB-CE intends to use ASME SB-166, which allows the use of Inconel 690 or Inconel 600, in the CEDM motor housing to be consistent with the reactor vessel head CEDM nozzles.

ECGB

Open Item 4.5.1-2

The applicant is proposing to use ASTM A708 in lieu of ASTM A262 (recommended in Regulatory Guide 1.44) for verifying non-sensitization of austenitic stainless steel materials. The proposed alternative (ASTM A708) is not equivalent to ASTM A262 and is unacceptable; the applicant should consider the using ASTM A262.

Response:

ABB-CE uses ASTM A262 for verifying non-sensitization of austenitic stainless steel product forms. This is consistent with the guidance provided in Regulatory Guide 1.44.

ABB-CE uses the Modified Strauss Test (ASTM A708) to identify whether fabricated (following welding or heat treating) austenitic stainless steel is sensitized and susceptible to intergranular stress corrosion cracking (IGSCC) or stress assisted intergranular attack (IGA) under Pressurized Water Reactor (PWR) coolant conditions (as opposed to BWR operating conditions). In addition, venting of the RCS reduces the oxygen content and thus the susceptibility to intergranular stress corrosion cracking in an ABB-CE PWR. (See also Open Item 4.5.1-4)

ASTM A708 (Modified Strauss Test) has been accepted by the Materials Engineering Branch as indicated in Standard Review Plan Section 4.5.1, III.2., as an alternative test that determines whether controls on the processing of austenitic stainless steel will be adequate to ensure that PWR components will not become susceptible to localized corrosion associated with sensitization. Regulatory Guide 1.44 page 1.44-2, paragraph 2 allows " Alternate test methods that can be qualified are also acceptable."

Therefore ABB-CE considers the use of ASTM A708 acceptable for verifying that fabricated austenitic stainless steel PWR components will not become susceptible to localized corrosion associated with sensitization.

Moreover, ABB-CE PWR operating experience to date demonstrates that these controls have been succesful in preventing any instance of localized corrosion associated with sensitized austenitic stainless steel.

ECGB

Open Item 4.5.1-3

The applicant is proposing to use Stellite, which is a cobalt-based alloy, for pins and latches in the CEDM. Activation of cobalt is a concern relating to the radioactivity in current nuclear plants. Therefore, cobalt application should be avoided in the CESSAR for as low as reasonably achievable considerations. In CESSAR 5.2.3.2.2, "Materials of Construction Compatibility with Reactor Coolant," the applicant states that cobalt-based alloys will be avoided except in cases where no proven alternative exists. The applicant should provide a discussion that it evaluated other alternatives to cobalt-based alloy and found them unacceptable for CESSAR applications.

Response:

Haynes Stellite No. 36, a cobalt-based alloy, is used for the CEDM latches and their pins. This alloy was selected due to its excellent wear resistance.

ABB-CE is reviewing work being conducted within ABB, by EPRI, and by others to evaluate replacement non-cobalt alloys. It is ABB-CE's opinion that while these alloys offer the potential for possessing equivalent wear resistance, limited full scale test data and the lack of operating experience makes it premature to commit to the use of these materials at this time. This situation may change with the completion of EPRI sponsored valve test programs and additional evaluation and testing of alternate materials defined in the ABB-CE First of a Kind Engineering Program.

Should alternative materials exhibit desirable characteristics, ABB-CE will consider the use of substitutes for Stellite.

CESSAR-DC will be revised in a future amendment to allow for the use of a material demonstrated to be functionally equivalent to Haynes Stellite No. 36 for the CEDM latches and pins.

3. Latch and magnet housing

ASTM A276, Type 316 (austenitic stainless steel)

QQ-C-320, Class 2B (chrome plating) D

ASTM A276, Type 440C (martensitic stainless steel)

4. Spacer

ASTM A240, Type 304 (austenitic stainless steel)

5. Alignment Tab

ASTM A276, Type 410 (martensitic stainless steel)

6. Spring

AMS 5698B, Inconel X-750 (nickel base alloy)

7. Pin

Haynes Stellite No. 6B (cobalt base alloy) or an alternate material demonstrated to be functionally equivalent

8. Dowel pin

300 Series SST

~~ASTM A314, Type 410 (martensitic stainless steel)~~

9. Spacer and screw

ASTM A276, Type 321 (austenitic stainless steel)

10. Stop

ASTM A276, Type 304 (austenitic stainless steel)

11. Latch and pin

Haynes Stellite No. 36 (cobalt base alloy) or an alternate material demonstrated to be functionally equivalent

12. Locking cup and screws

Type 300 Series austenitic stainless steel

13. Steel Ball

ASTM A276, Type 440C F

The functions of the CEDM motor assembly components are described in Section 3.9.4.1.

ECGB

Open Item 4.5.1-4

The applicant is proposing to use Types 304 and 316 austentitic stainless steel. However, these materials are susceptible to intergranular stress corrosion cracking. The applicant should consider using low-carbon, wrought austenitic stainless steel, which includes Types 304L and 316L, 304NG, 316NG, and modified Type 347.

Response:

Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel, indicates that for systems where the oxygen content is kept below 0.1 ppm when the temperature is above 200 °F, unstabilized grades of stainless steel are acceptable. During the start-up and operation of the ABB-CE NSSS, these conditions are maintained through specified chemistry control. ABB-CE specifically prohibits the use of unstabilized or normal carbon content material from being exposed to the sensitizing range of 800-1500 °F with the exception of the short duration during welding. Furthermore, all welded austenitic stainless steel is limited by additional ordering requirements to 0.065% maximum carbon content. Controls on welding ensures that this material will not become sensitized and susceptible to localized corrosion under PWR operating conditions. Low carbon or stabilized grades would be used if exposure to the sensitizing temperature range was required.

Experience to date indicates that these controls have been successful in preventing any instance of localized corrosion associated with sensitized austenitic stainless steel.

Therefore, ABB-CE considers the use of Type 304 and 316 stainless steels, in conjunction with the material chemistry and welding controls noted above, acceptable for resistance to stress corrosion cracking.

ECGB

Open Item 4.5.1-5

The ferrite content limits for austenitic steel castings and weld metal in CESSAR are broader than those in industry guidelines and staff guidance. The applicant should revise the CESSAR to be consistent with industry guidelines and staff guidance.

Response:

The ferrite limits on austenitic stainless steels are as follows:

Undiluted Weld Metal
for Stainless Steel Weldments: 5 FN - 15 FN

These limits comply with the ASME Boiler and Pressure Vessel Code, Regulatory Guides 1.31 and 1.44 and also represent those used (successfully) in most of the Pressurized Water Reactor (PWR) components fabricated by ABB-CE.

The above specified range of delta ferrite for weld material of 5 to 15 FN provides adequate control of stainless steel for System 80+. This range of ferrite in stainless steel weld metals has been shown to be sufficient to avoid microfissuring during welding. The 5 to 15 FN ferrite range, combined with other controls on materials, heat treatments and welding parameters, has also been demonstrated to effectively avoid sensitization and intergranular stress corrosion cracking (IGSCC). The effectiveness of these controls has been demonstrated through successful in-reactor service.

The additional controls recommended in the EPRI ALWR document, including the 8 FN average value may be appropriate for some BWR applications because of the more agressive nature of the BWR environment. However, the existing controls on stainless steel welds are sufficient to avoid IGSCC resulting from weld metal sensitization in the PWR enivronment.

Moreover, in many cases, even wider limits had been utilized without any detrimental consequences. It is ABB-CE's experience that these limits provide more than adequate resistance to hot fissuring and resistance to intergranular corrosion particularly in the PWR environment.

CESSAR-DC will be revised in a future amendment to reflect the ferrite content limits given above.

Weld heat affected zone sensitized austenitic stainless steel (which will fail in the Strauss Test, ASTM A708) is avoided in control element drive mechanism structural components by careful control of:

A. Weld heat input to less than 60 kJ/in

B. Interpass temperature to 350°F maximum

C. Carbon content to ≤ 0.065% D

4.5.1.4 Control of Delta Ferrite in Austenitic Stainless Steel Welds

The austenitic stainless steel, primary pressure retaining welds in the control element drive mechanism structural components are consistent with the recommendations of Regulatory Guide 1.31 as follows:

The delta ferrite content of A-No.8 (Table QW-442 of the ASME Code, Section IX) austenitic stainless steel welding materials is controlled to 5FN-~~20FN~~ 15. D

The delta ferrite determination is carried out using methods specified in the ASME Code, Section III, for each heat, lot or heat/lot combination of weld filler material. For the submerged arc process, the delta ferrite determination for each wire/flux combination may be made on a production or simulated (qualification) production weld.

4.5.1.5 Cleaning and Contamination Protection Procedures

The procedure and practices followed for cleaning and contamination protection of the control element drive mechanism structural components are in compliance with the recommendations of Regulatory Guide 1.37 (including ANSI/ASME NQA-2-1983) and are described below: F

Specific requirements for cleanliness and contamination protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described below indicate the type of procedures utilized for components to provide contamination control during fabrication, shipment, and storage.

Contamination of austenitic stainless steels of the Type 300 series by compounds that can alter the physical or metallurgical structure and/or properties of the material is avoided during all stages of fabrication. Painting of Type 300 series stainless steels is prohibited. Grinding is accomplished with resin or

ECGB

Open Item 4.5.1-6

CESSAR Section 4.5.1.1 indicates that martensitic stainless steel will be used. The applicant should specify the heat treatment for these materials.

Response:

Components in the Control Element Drive Mechanism (CEDM) using martensitic stainless steels include the motor housing assembly, magnet and spacer, latch and magnet housing, alignment tab, and steel ball. The heat treatment for these components is as follows:

Motor Housing Assembly - ASME Code Case N-4-11 (modified Type 403 martensitic stainless steel), and additional requirements of ASME SA-182.

Heat treat - Heat to 1800 °F +/- 25 °F, air cool and temper at 1125 °F minimum for 4 hours per Code Case N-4-11.

Magnet and Spacer - ASTM A276, Type 410.

Magnet & Latch Spacer - Condition A - annealed

Center Spacer - Condition T - hardened and tempered at a relatively high temperature, Brinell hardness - 300 max.

Latch and Magnet Housing Inserts - ASTM A276, Type 440C.

Heat treat- Heat at 1850 °F to 1950 °F for 1 hour at temperature. Quench in oil to room temperature. Immediately after quenching subzero cool to -100 °F +/- 25 °F for 2 hours. Temper at 600 °F +/- 25 °F for 2 hours & air cool to room temperature. Repeat temper procedure at 600 °F +/- 25 °F for 2 hours & air cool to room temperature to achieve Rockwell C-54 +/- 5.

Alignment Tab - ASTM A276, Type 440C.

Heat treat- Heat at 1850 °F to 1950 °F for 1 hour at temperature. Quench in oil to room temperature. Immediatly after quenching subzero cool to -100 °F +/- 25 °F for 2 hours. Temper at 600 °F +/- 25 °F for 2 hours & air cool to room temperature. Repeat temper procedure at 600 °F +/- 25 °F for 2 hours & air cool to room temperature to achieve Rockwell C-54 +/- 5.

Open Item 4.5.1-6

Response: (cont.)

Dowel Pin - This item is listed as ASTM A314, Type 410, but actually is 300 series Stainless Steel. The dowel pin material will be revised in a future amendment of CESSAR.

Steel Ball - ASTM A276, Type 440C, Condition T - hardened and tempered at a relatively high temperature, then case hardened.

It should be noted that only the Motor Housing Assembly is a primary pressure boundary component. The heat treatment for the Motor Housing Assembly only will be added to CESSAR-DC in a future amendment. Heat treatments for all other materials are provided in this response for information only.

ECGB

Open Item 4.5.1-7

In CESSAR, Section 4.5.1.1 indicates that Inconel X-750 (AMS 5698B and AMS 5699B) will be used. The applicant should confirm that these materials are listed as acceptable in Section III of the ASME Code or RG 1.85. Further the applicant should specify the heat treatment.

Response:

AMS 5698 and 5699 forms of Inconel X-750 are used for springs to be used at elevated temperatures and requiring resistance against relaxation. These materials have demonstrated to be acceptable for their intended use by prototype testing of the CEDM. The springs, not part of the primary pressure boundary, are not required to be listed as accepted in Section III of the ASME Code or RG 1.85.

AMS 5698 and 5699 are drawn from hot finished wire rod which has been previously ground or has had surface preparation (other than by pickling) for removal of seams or other injurious surface imperfections. The wire is heat treated at 2100 °F before reducing to size.

As these CEDM springs are not pressure boundary components, this heat treatment is provided in this response for information only. CESSAR will not be modified to include this heat teatment.

ECGB

Open Item 4.5.1-8

In CESSAR Section 4.5.1.3.3, the applicant indicates a carbon content limit for austenitic stainless steel. The applicant should consider limiting the carbon content to less than 0.02-percent.

Response:

ABB-CE has considered limiting the carbon content for austenitic stainless steel to less than 0.02% and concluded that for PWR conditions, a carbon content of less than 0.065%, combined with rigorous welding process control, is sufficient to assure that the material will not become sensitized. See response to Open Item 4.5.1-4.

ECGB

Open Item 4.5.1-9

In CESSAR Section 4.5.1.1, the applicant indicated that CEDM materials were used in an extensively tested CEDM assembly that exceeded lifetime requirements. The applicant should verify that the test results are applicable to a 60-year plant life.

Response:

CEDM materials were used in an extensively tested CEDM assembly that exceeded lifetime requirements, as described in Section 3.9.4.4.1. The design duty or lifetime requirement as defined in CESSAR Section 3.9.4.1 is a total cumulative CEA travel of 100,000 feet of operation without loss of function and not the 60-year plant life. As indicated in CESSAR Section 3.9.4.1, the CEDM is designed to operate without maintenance for a minimum of 1-1/2 years and without replacing components for a minimum of 3 years. Therefore the test results of the extensively tested CEDM do not need to be verified to the 60-year plant life.

The operational requirement for the System 80+ CEA's, with the possible exception of the lead regulating CEA group, is expected to be less than the 100,000 feet of travel (the tested life) over the 60 year plant life. If plants institute daily load cycle operation on a regular basis, the lead regulating CEA group may exceed 100,000 feet of travel.

The regulating CEA's are much lighter than the CEA weight used during accelerated CEDM motor life tests, and it is expected that, when operating a regulating CEA, the System 80+ CEDM motors are capable of operation in excess of 100,000 feet of cumulative travel. Depending on the extent that the lead regulating CEA group is utilized, a one time CEDM motor replacement for this bank of CEA's may be required during the 60- year plant life.

As indicated in CESSAR Section 3.9.4.1, all CEDM pressure boundary components have a design life of 60 years.

ECGB

Open Item 4.5.2-1

The applicant is proposing to use Stellite, which is a cobalt-based alloy, as a hardfacing material. As discussed in Section 4.5.1 of this DSER, the applicant states that cobalt-based alloys, will be avoided except if no proven alternative exists. The applicant should state that no other alternatives to the cobalt-based alloy have been evaluated and found acceptable for CESSAR applications.

Response:

ABB-CE is reviewing work being conducted within ABB, by EPRI, and by others to evaluate replacement non-cobalt alloys. It is ABB-CE's opinion that while these alloys offer the potential for possessing equivalent wear resistance, limited full scale test data and the lack of operating experience makes it premature to commit to the use of these materials at this time. This situation may change with the completion of EPRI sponsored valve test programs and additional evaluation and testing of alternate materials defined in the ABB-CE First of a Kind Engineering Program.

Should alternative materials exhibit desirable characteristics, ABB-CE will consider the use of substitutes for Stellite.

CESSAR-DC will be revised in a future amendment to allow for the use of a material demonstrated to be functionally equivalent to Stellite.

Open Item 4.5.2-1

E. Bolt and pin material

ASTM-A-453 and ASTM-A-638, Grade 660 material (trade name A-286) is used for bolting and pin applications. This alloy is heat treated in accordance with the ASTM specifications by precipitation hardening at 1300-1400°F for 16 hours to a minimum yield strength of 85,000 psi. Its corrosion properties are similar to those of the Type 300 series austenitic stainless steels. It is austenitic in all conditions of fabrication and heat treatment. This alloy was used for bolting in previous reactor systems and test facilities in contact with primary coolant and has proven completely satisfactory. D

F. Chrome plating and hardfacing

Chrome plating or hardfacing are employed on reactor internals components or portions thereof where required by function. Chrome plating complies with Federal Specification No. QQ-C-320. The hardfacing material employed is Stellite 25, or an alternate material demonstrated to be functionally equivalent D

All of the materials employed in the reactor internals and in-core instrument support system have performed satisfactorily in operating reactors such as Palisades (Docket-50-255), Fort Calhoun (Docket-50-285) and Maine Yankee (Docket-50-309).

4.5.2.2 Welding Acceptance Standards

Welds employed on reactor internals and core support structures are fabricated in accordance with Article NG-4000 in Section III, and meet the acceptance standards delineated in article NG-5000, Section III, Division I, and control of welding is performed in accordance with Section III, Division I, and Section IX of the ASME Code. In addition, consistency with the recommendations of Regulatory Guides 1.31 and 1.44 is described in Section 4.5.2.3. F

4.5.2.3 Fabrication and Processing of Austenitic Stainless Steel

The following information applies to unstabilized austenitic stainless steel as used in the reactor internals.

4.5.2.3.1 Control of the Use of Sensitized Austenitic Stainless Steel

The recommendations of Regulatory Guide 1.44, as described in Sections 4.5.2.3.1.1 through 4.5.2.3.2.5, are followed except for the criterion used to demonstrate freedom from sensitization. The ASTM A708 Strauss Test is used in lieu of the ASTM A262 Method E, Modified Strauss Test, to demonstrate freedom from

ECGB

Open Item 4.5.2-2

As discussed in Section 4.5.1, the applicant is proposing to use American Society for Testing and Materials (ASTM) A708 in lieu of ASTM A262; ASTM A262 is recommended in RG 1.44 for sensitization of austenitic stainless steel materials. The proposed alternative (ASTM A708) is not equivalent to ASTM A262 and is unacceptable; the applicant should consider using ASTM A262.

Response:

ABB-CE uses ASTM A262 for verifying non-sensitization of austenitic stainless steel product forms. This is consistent with the guidance provided in Regulatory Guide 1.44.

ABB-CE uses the Modified Strauss Test (ASTM A708) to identify whether fabricated (following welding or heat treating) austenitic stainless steel is sensitized and susceptible to intergranular stress corrosion cracking (IGSCC) or stress assisted intergranular attack (IGA) under Pressurized Water Reactor (PWR) coolant conditions (as opposed to BWR operating conditions).

ASTM A708 (Modified Strauss Test) has been accepted by the Materials Engineering Branch as indicated in Standard Review Plan Section 4.5.1, III.,2. as an alternative test that determines whether controls on the processing of austenitic stainless steel will be adequate to ensure that PWR components will not become susceptible to localized corrosion associated with sensitization. Regulatory Guide 1.44 page 1.44-2, paragraph 2 allows " Alternate test methods that can be qualified are also acceptable."

Therefore ABB-CE considers the use of ASTM A708 acceptable for verifying that fabricated austenitic stainless steel PWR components will not become susceptible to localized corrosion associated with sensitization.

Moreover, ABB-CE PWR operating experience to date demonstrates that these controls have been succesful in preventing any instance of localized corrosion associated with sensitized austenitic stainless steel.

ECGB

Open Item 4.5.2-3

The applicant is proposing to use Type 304 austenitic stainless steel. However, these materials are susceptible to intergranular stress corrosion cracking. The applicant should consider using low-carbon, wrought austenitic stainless steel, which includes Types 304L, 316, 304NG, and modified Type 347.

Response:

Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel, indicates that for systems where the oxygen content is kept below 0.1 ppm when the temperature is above 200 °F, unstabilized grades of stainless steel are acceptable. During the start-up and operation of the ABB-CE NSSS, these conditions are maintained through specified chemistry control. ABB-CE specifically prohibits the use of unstabilized or normal carbon content material from being exposed to the sensitizing range of 800-1500 °F with the exception of the short duration during welding. Furthermore, all welded austenitic stainless steel is limited by additional ordering requirements to 0.065% maximum carbon content. Controls on welding ensures that this material will not become sensitized and susceptible to localized corrosion under PWR operating conditions. Low carbon or stabilized grades would be used if exposure to the sensitizing temperature range was required.

Experience to date indicates these controls have been successful in preventing any instance of localized corrosion associated with sensitized austenitic stainless steel.

Therefore, ABB-CE considers the use of Type 304 and 316 stainless steels, in conjunction with the material chemistry and welding controls noted above, acceptable for resistance to stress corrosion cracking.

ECGB

Open Item 4.5.2-4

In CESSAR Section 4.5.2.1, the applicant indicates that Inconel will be used to fabricate the flow skirt. The applicant should clarify whether Inconel 600 will be used. As discussed in Section 4.5.1 of this DSER, the applicant should consider alternate materials that are resistant to cracking.

Response:

The flow skirt is fabricated from Inconel 600 due to its specific design requirements. Inasmuch as the skirt is one of the coolest regions of the primary systems, the potential for primary water stress corrosion cracking (PWSCC) is minimal. The suitability of using Inconel 600 for the flow skirt has been demonstrated by many years of successful operating experience. The material form required for the size and configuration of the flow skirt is not available in an alternate material with comparable properties (such as the PWSCC resistant Inconel 690).

Therefore, this precludes using Inconel 690 at the present time.

Should suitable product forms of an alternate material become available, ABB-CE will consider incorporating them into the design.

ECGB

Open Item 4.5.2-5
As discussed in Section 4.5.1 of this DSER, the ferrite content limits for austenitic steel castings and weld metal given in CESSAR are broader than those in industry guidelines and staff guidance. The applicant should revise the CESSAR to be consistent with industry guidelines and staff guidance, whichever is more limiting.

Response:
The ferrite limits on austenitic stainless steels are as follows:

Castings	5 FN - 30 FN
Undiluted Weld Metal for Stainless Steel Weldments:	5 FN - 15 FN
Stainless Steel Overlay Cladding	5 FN - 15 FN

These limits comply with the ASME Boiler and Pressure Vessel Code, Regulatory Guides 1.31 and 1.44 and also represent those used (successfully) in most of the Pressurized Water Reactor (PWR) components fabricated by ABB-CE.

The above specified range of delta ferrite for weld material and overlay cladding of 5 to 15 FN provides adequate control of stainless steel for System 80+. This range of ferrite in stainless steel weld metals and overlay cladding has been shown to be sufficient to avoid microfissuring during welding. The 5 to 15 FN ferrite range, combined with other controls on materials, heat treatments and welding parameters, has also been demonstrated to effectively avoid sensitization and intergranular stress corrosion cracking (IGSCC). The effectiveness of these controls has been demonstrated through successful in-reactor service.

The additional controls recommended in the EPRI ALWR document, including the 8 FN average value may be appropriate for some BWR applications because of the more agressive nature of the BWR environment. However, the existing controls on stainless steel welds are sufficient to avoid IGSCC resulting from weld metal sensitization in the PWR environment.

The range of ferrite content from 5 - 30 FN for stainless steel castings is also sufficient to avoid IGSCC from sensitization.

Moreover, in many cases, even wider limits had been utilized without any detrimental consequences. It is ABB-CE's experience that these limits provide more than adequate resistance to hot fissuring and resistance to intergranular corrosion particularly in the PWR environment.

Open Item 4.5.2-5

sensitization in fabricated unstabilized austenitic stainless steel, since the former test has shown, through experimentation, excellent correlation with the type of corrosion observed in severely sensitized austenitic stainless steel.

4.5.2.3.1.1 Solution Heat Treatment Requirements

All raw austenitic stainless steel material, both wrought and cast, employed in the fabrication of the reactor internals is supplied in the solution annealed condition, as specified in the pertinent ASTM or ASME B&PV Code material specification (i.e., 1900 to 2050°F for 0.5 to 1.0 hour per inch of thickness and rapidly cooled to below 700°F). The time at temperature is determined by the size and the type of component.

Solution heat treatment is not performed on completed or partially fabricated components. Rather, the extent of chromium carbide precipitation is controlled during all stages of fabrication as described in Section 4.5.2.3.1.4.

4.5.2.3.1.2 Material Inspection Program

Extensive testing of stainless steel mockups, fabricated using production techniques, was conducted to determine the effect of various welding procedures on the susceptibility of unstabilized Type 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure are used in the fabrication of reactor internals components. The ASTM Standard A708 (Strauss Test) is the criterion used to determine susceptibility to intergranular corrosion. This test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steel. As such, ASTM A708 is utilized as a go/no-go standard for acceptability.

As a result of the above tests, a relationship was established between the carbon content of Type 304 stainless steel and weld heat input. This relationship is used to avoid weld heat affected zone sensitization as described in Section 4.5.2.3.1.4.

4.5.2.3.1.3 Unstabilized Austenitic Stainless Steels

The unstabilized grade of austenitic stainless steel with a carbon content greater than 0.03% used for components of the reactor internals is Type 304. This material is furnished in the solution annealed condition. The acceptance criterion used for this material, as furnished from the steel supplier, is ASTM A262, Method E.

Exposure of completed or partially fabricated components to temperatures ranging from 800 to 1500°F is prohibited except as described in Section 4.5.2.3.1.5.

Duplex, austenitic stainless steels containing more than 5FN delta ferrite (weld metal, cast metal, weld deposit overlay) are not considered unstabilized since these alloys do not sensitize, i.e., form a continuous network of chromium-iron carbides. Specifically, alloys in this category are:

CF8M CF8	Cast stainless steel (delta ferrite controlled to 5FN-~~33~~FN) 30
308, 309 312, 316	Singly and combined stainless steel weld filler metals (delta ferrite controlled to 5FN-~~20~~FN 15 as deposited) D

In duplex austenitic/ferritic alloys, chromium-iron carbides are precipitated preferentially at the ferrite/austenite interfaces during exposure to temperatures ranging from 800-1500°F. This precipitate morphology precludes intergranular penetrations associated with sensitized Type 300 series stainless steels exposed to oxygenated or otherwise faulted environments.

4.5.2.3.1.4 Avoidance of Sensitization

Exposure of unstabilized austenitic Type 300 series stainless steels to temperatures ranging from 800 to 1500°F will result in carbide precipitation. The degree of carbide precipitation or sensitization depends on the temperature, the time at that temperature, and the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing halides. Such a metallurgical structure will readily fail the Strauss Test, ASTM A708. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

Weld heat affected zone sensitized austenitic stainless steels (which will fail the Strauss Test, ASTM A708) are avoided by careful control of:

A. Weld heat input to less than 60 kJ/in

B. Interpass temperature to 350°F maximum

C. Carbon content to ≤ 0.065 D

ECGB

Open Item 4.5.2-6

In CESSAR Section 4.5.2.1, the applicant indicates that precipitation hardened stainless steel will be used. The applicant should specify the heat treatment for these materials.

Response:

The precipitation hardened stainless steel used in the reactor internals is SA 453 Grade 660 or SA 638 Grade 660. The heat treatment of either of these specifications is as follows:

Solution Treatment:	1650 +/- 25 °F, for 2 hours minimum oil or water quench
Hardening Treatment:	1350 +/- 25 °F, for 16 hours air cool

These materials are used for very limited applications, and ABB-CE has had very good experience with these materials in operating plants.

ECGB

Open Item 4.5.2-7

In CESSAR Section 4.5.2.3.1.4 indicates a carbon content limit for austenitic stainless steel. As discussed in Section 4.5.1 of this DSER, the applicant should consider limiting the carbon content to less than 0.02 percent.

Response:

ABB-CE has considered limiting the carbon content for austenitic stainless steel to less than 0.02% and concluded that for PWR conditions, a carbon content of less than 0.065%, combined with rigorous welding process control, is sufficient to assure that the material will not become sensitized. See the response to Open Item 4.5.2-3.

Open Item 5.2.1.2-1:

ECGB

The applicant should provide a complete list of all ASME code case interpretations referenced in the CESSAR.

Response to Open Item 5.2.1.2-1:

ASME code case interpretations is incorrect nomenclature. The ASME code presents both code cases and paragraph interpretations. CESSAR identifies only ASME Code Cases which are intended to be utilized during design and manufacture.

CESSAR Table 5.2-3 will be modified to add the ASME Code Cases below:

1. N-411-1 Alternative Damping Values for Response Spectra Analysis for Class 1, 2 and 3 Piping Section III, Divison 1

2. N-71-15 Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III, Division 1.

3. N-60-3 Material for Core Support Structures, Section III, Division 1.

4. N-474-1 Design Stress Intensities and Yield Strength Values for UNS N06690 with a minimum specific Yield Strength of 35 KSI, Class 1 Components, Section III, Division 1.

Note: 1) Code Cases intended for use are in accordance with Regulatory Guide 1.85, Revision 28 - Materials Code Case Acceptability ASME Section III, Division 1 and Regulatory Guide 1.84 - Design and Fabrication Code Case Acceptability ASME Section III, Division 1.

Open Item 5.2.3-5: ECGB

The applicant should provide a complete list of the materials used for reactor coolant pressure boundary components in CESSAR Table 5.2-2.

Response to Open Item 5.2.3-5:

The following will be added to CESSAR Table 5.2-2:

Surge Line	SA-312 TP347 (Piping); SA-403 WP347 (Elbows) SA-182 F347 (Safe Ends)
Accumulator Line	SA-312 TP316, TP304 or SA-376 TP316, TP304

In addition, Table 5.2-2 will be revised as noted on the attached. These changes will be included in the next CESSAR revision.

Ref: Open Item 5.2.3-5

TABLE 5.2-2

(Sheet 1 of 5)

REACTOR COOLANT SYSTEM MATERIALS

Component	Material Specification	
Reactor Vessel		
Forgings	SA-508 Class 2 and 3	I
Cladding (a)	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite or NiCrFe alloy (equivalent to SB-~~168~~ 166)	D
Nozzle Safe Ends	SA-508 Class 1	I
Reactor vessel head (a) CEDM Nozzles	SB-166	
Vessel internals[a]	Austenitic Stainless Steel and NiCrFe alloy	
Fuel cladding[a]	Zircaloy-4	
Instrument nozzles (a)	SB-166	
Control element drive mechanism housings		
Lower	Type 403 stainless steel according to Code Case N-4-11 with end fittings to be SB-166 and/or SA-182 Type 348 stainless steel	D
Upper	SA-479 and SA-213 Type 316 stainless steel with end fitting of SA 479 Type 316 and vent valve seal of Type 316 and vent valve seal of Type 440 stainless steel seat	
Closure head bolts	SA-540 B24 or B23	
Pressurizer		
Shell	SA-533 Grade A or B Class 1 or SA-508 Class 3	D
Cladding[a]	Weld deposited austenitic stainless steel with 5 FN-18FN delta ferrite or NiCrFe alloy (equivalent to SB-166)	D

Ref: Open Item 5.2.3-5

TABLE 5.2-2 (Cont'd)

(Sheet 2 of 5)

REACTOR COOLANT SYSTEM MATERIALS

Component	Material Specification	
Forged nozzles	~~SA-508 Class 1, 2 or 3~~ SA-541, Class 3	D
Instrument nozzles(a)	SB-166	
~~Surge and safety valve~~ Nozzle safe ends(a)	SA-182, F347	
Safety Valve Nozzle Flange(a)	SA-182	
Studs and nuts	~~SA-540 B24 or B23~~ SB-637	
Steam generator		
Primary Head	SA-533 Grade B, Class 1 or SA-508 Class 3	I
Primary Nozzles	SA-508 Class 2 or 3	
Primary head cladding(a)	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite	D
Tubesheet	SA-508 Class 2 or 3	
Tubesheet stay	SA-508 Class 2 or 3	
Tubesheet cladding(a)	Weld deposited NiCrFe alloy (equivalent to SB-168)	
Tube(a)	NiCrFe Alloy 690 (SB-163)	
Tube supports ~~(a)~~	A-176, Type 409 (ASTM)	
Secondary shell	SA-533 Grade A or B, Class 1, or SA-508, Class 3	I
Secondary head	SA-516 Grade 70 or SA-508, Class 1A	
Secondary nozzles	SA-508 Class 1, 2 or 3	
Secondary nozzle safe ends	SA-508 Class 1A	
Secondary instrument nozzles	SA-106 Grade B	
Secondary studs and nuts	SA-540 Grade B24, or SA-193 Grade B7	

Ref: Open Item 5.2.3-5

TABLE 5.2-2 (Cont'd)

(Sheet 3 of 5)

REACTOR COOLANT SYSTEM MATERIALS

Component	Material Specification	
Primary studs and nuts	SA-564, Type 530H1100 or SB-637, No. 771B	I
Reactor Coolant Pumps		
Casing[(a)]	SA-508 Class 2 or 3 or austenitic stainless steel	B
Cladding	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite	D
Internals	SA-487 CA6NM, SA 336 Grade F8 or austenitic stainless steel	B
Reactor Coolant Piping		
Pipe (30 in. and 42 in.)	SA-516 Grade 70 or SA-508 Class 1(a)	
Cladding[(a)]	Weld deposited austenitic stainless steel with 5FN-18FN delta ferrite	D
Piping nozzles and safe ends		D
Nozzle forgings	SA-508 Class 1(a) or SA-182 ~~SA-105, SA-541 Class 1, 2 or 3, or SB-166~~	D
Instrument Nozzles	SB-166	
Nozzle safe ends	SA-182 ~~or SB-166~~	
Valves	SA-351 CF8M or SA-182	

Ref: Open Item 5.2.3-5

TABLE 5.2-2 (Cont'd)

(Sheet 4 of 5)

REACTOR COOLANT SYSTEM MATERIALS

WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

Base Material Type ~~Specification~~	Base Material Type	Type of Weld Material	
1. SA-533 Gr. B Cl.1	SA-533 Gr. B Cl.1	a. SFA 5.5,[(b)] E-8018-C3, E-8018-G b. MIL-E-18193, B-4	
2. SA-508 Cl.2	SA-533 Gr. B Cl.1	a. SFA 5.5, E-8018-C3, E-8018-G b. MIL-E-18193, B-4	
3. SA-508 Cl.1	SA-508 Cl.2	a. SFA 5.5, E-8018-C3, E-8018-G	
4. SA-516 Gr. 70	SA-516 Gr. 70	a. SFA 5.1, E-7018	
5. SA-182 F1	SA-516 Gr. 70	a. SFA 5.1, E-7018	
6. SA-105 Gr. 11	SA-~~351~~ 182 ~~CF8M~~ F347	a. SFA 5.14, ERNiCr-3 SFA-5.11, ENiCrFe-3	
7. SA-182 F1	SA-~~351~~ 182 ~~CF8M~~ F347	a. SFA 5.11, ENiCrFe-3 SFA-5.14, ERNiCr-3	
8. SA-105 Gr. 11	SA-182 F316	a. SFA 5.14, ERNiCr-3 SFA-5.11, ENiCrFe-3	
9. SB-166	SA-182 F316	a. Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3	D
10. SB-167	SA-182 F304	a. Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3	
11. SA-516 Gr. 70	SA-351 CF8M	a. SFA ~~5.1, E-7018~~ 5.11, ENiCrFe-3 ~~b. MIL-E-18193, B-4~~	D
12. SA-182 F1	SA-182 F316	a. SFA ~~5.1, E-7018~~ 5.11 ENiCrFe-3 SFA 5.14, ERNiCr-3	
13. SB-166	SA-533 Gr. B Cl.1	a. SFA 5.14, ERNiCr-3 SFA-5.11, ENiCrFe-3	

Ref: Open Item 5.2.3-5

TABLE 5.2-2 (Cont'd)

(Sheet 5 of 5)

REACTOR COOLANT SYSTEM MATERIALS

WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

Base Material Type ~~Specification~~	Base material Type		Type of Weld Material
14. SA-182	SB-167	a.	SFA 5.14, ERNiCr-3 SFA 5.11, ENiCrFe-3
15. SA-516 Gr. 70	SA-508 Cl.2	a.	SFA 5.5,(b) E-~~8016~~ 8018-C3
16. Austenitic stainless steel cladding		a.	SFA 5.9, ER-308 S' \ 5.9, ER-309 ~~SFA 5.9, ER-312~~ SFA 5.4, E-309, E-308
17. Inconel Cladding	~~Inconel~~	a.	SFA-5.11 ENiCrFe-3 SFA-5.14 ERNiCr-3
18. SA-508 Cl. 3	SA-508 Cl. 3	a. b.	SFA 5.5,(b) E-8018-C3, E-8018-G MIL-E-18193, B-4
19. SA-508 Cl. 3	SA-533 Gr. B Cl.1	a. b.	SFA 5.5, E-8018-C3, E-8018-G MIL-E-18193, B-4
20. SA-508 Cl. 3	SA-508 Cl. 2	a.	SFA 5.5, E-8018-C3, E-8018-G
21. SA-508 Cl. 3	SA-516 Gr. 70	a.	SFA 5.5,(b) E-8018-C3
22 SB-166, 167, 168	SB-166, 167, 168	a.	SFA 5.11, ENiCrFe-3 SFA 5.14, ERNiCr-3

D

Notes: a. Materials exposed to reactor coolant.

b. Special weld wire with low residual elements of copper, nickel and phosphorous as specified for the reactor vessel core beltline region. D

ECGB

Open Item 5.3.1-1

The applicant should consider lowering the nickel content in the reactor vessel forging and the phosphorus content in the reactor vessel forging and weld.

Response to Open Item 5.3.1-1

The nickel content of the reactor vessel beltline forgings is allowed to vary the full range (0.4% to 1.00%) permitted by the SA 508 Class 3 material specification. The available data indicate that the influence of nickel on susceptibility to irradiation damage is limited when other impurity elements, in particular copper, are controlled to very low levels. Weld metals show slightly more sensitivity to nickel at low copper contents than base metals. Therefore, the nickel content for beltline weld metal is controlled to lower levels than the forging material.

Phosphorus content in the reactor vessel forgings and weld metal is controlled to a maximum of 0.012%. The possible effect of phosphorus on predicted shifts in low copper RPV beltline materials was evaluated by ORNL and NRC using the Power Reactor-Embrittlement Database (PR-EDB). Only a marginal correlation was observed between phosphorus content and predicted shift for low copper materials. The conclusion that there was any significant correlation was judged to be weak. Even in those cases where a possible trend with phosphorus content was suggested by the data, the differences between the observed and predicted shifts were within the margins applied by Regulatory Guide 1.99, Rev. 2. In addition, in the cases where a trend with phosphorus was suggested, it was generally the high phosphorus (>0.012%) materials which indicated the possible trend. Based on this previous assessment of the effects of impurity elements on the irradiation response of materials, the existing controls on residual elements are concluded to be sufficient. The limits on copper, phosphorus and other residual elements will minimize the extent of radiation damage to the RPV beltline materials. The radiation induced shifts in reference temperatures for these materials can be predicted with reasonable accuracy and conservatism using the methodology of Regulatory Guide 1.99, Rev. 2. The revised phosphorus content for the reactor vessel forgings and weld metal will be included in a future amendment of CESSAR-DC as shown on the attached page.

Ref: Open Item 5.3.1-1

B. The adjustment in the reference temperature caused by irradiation (ΔRT_{NDT}) is +53°. This calculated value assumes a forging with $\boxed{0.06}$ 0.03 wt-% maximum copper content, and a 1.00 wt-% maximum nickel content. J

C. The margin added for uncertainties is +34°F.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specification

A list of specifications for the principal ferritic materials, austenitic stainless steels, bolting and weld materials, which are part of the reactor coolant pressure boundary is given in Table 5.2-2.

Studies have shown that the irradiation induced mechanical property changes of SA-533B and SA-508 materials can depend significantly upon the amount of residual elements present in the compositions, namely; copper, nickel, phosphorous, and vanadium. D It has also been found that residual sulfur affects the initial toughness of SA-533B and SA-508 materials. Specific controls are placed on the residual chemistry of reactor vessel materials and the as-deposited welds used to join these materials to limit the B maximum predicted increase in the reference temperature (RT_{NDT}, which is discussed in Sections 5.3.1.6 and 5.3.2.1.1) and to limit the extent of the reactor vessel beltline. The beltline is defined by Appendix C of 10 CFR 50.

Materials used in the reactor vessel beltline and the as-deposited welds contain no greater than the following low percentages of residual elements: B

Copper (in welds)	0.03		
Copper (in forgings)	$\boxed{0.06}$ 0.03	Phosphorous	~~0.015~~ 0.012
Nickel (in forgings)	1.00	Sulfur	0.015
Nickel (in welds)	0.10	Vanadium	0.030

E

ECGB

Open Item 5.3.1-2

The applicant should revise the estimate of the shift in the reference temperature for its reactor vessel surveillance program.

Response to Open Item 5.3.1-2

Predicted shifts in the reference temperature for the reactor vessel materials are calculated using the methodology of Regulatory Guide 1.99, Revision 2. This methodology provides reasonably accurate and conservative predictions of adjusted reference temperatures for RPV beltline materials, including low copper base and weld metals with phosphorus impurities controlled to low levels.

It has been suggested that for purposes of establishing the surveillance program the EOL shift should be estimated from the largest of the R.G. 1.99, Rev. 2 prediction, the R.G. 1.99, Rev. 1 prediction or a shift between 100 F and 200 F. The estimate in shift does not need to be revised to develope a surveillance program for design certification.

The surveillance program is based on a reasonably conservative estimate of the temperature shift. However, the surveillance program does not consist of the minimum requirements based on estimated shift. Additional capsules are included for contingency in the event that the actual shift is higher than originally estimated. Based on the predicted shift for the beltline materials, only 3 capsules are recommended by ASTM E 185. The recommended minimum number of surveillance capsules in ASTM E 185 for a reactor vessel with an EOL shift between 100 F and 200 F is four (4). The System 80+ surveillance program includes 6 capsules with archive materials available for at least two additional complete replacement capsules which can be installed in the reactor at any time when circumstances indicate that an additional capsule is required and when there is an available holder location.

ECGB

Open Item 5.3.2-1:

The applicant should revise the predicted shift in reference temperature.

Response to Open Item 5.3.2-1:

It has been suggested that for the purpose of establishing the pressure-temperature limits the EOL shift should be estimated from the larger of the R.G. 1.99, Rev. 2 prediction based on copper and nickel content or the R.G. 1.99, Rev. 1 prediction based on copper and phosphorus content. The estimate in shift does not need to be revised to develop the pressure-temperature limits for design certification.

Predicted shifts in the reference temperature for the reactor vessel materials are calculated using the methodology of Regulatory Guide 1.99, Revision 2. This methodology provides reasonably accurate and conservative predictions of the adjusted reference temperatures for RPV beltline materials, including low copper base and weld metals with phosphorus impurities controlled to low levels.

As discussed in the response to Open Item 5.3.1-1, the phosphorus content in the reactor vessel forgings and weld metal is controlled to a maximum of 0.012 %. The possible effect of phosphorus on predicted shifts in low copper RPV beltline materials was evaluated by ORNL and the NRC using the EDB database. Only a marginal correlation was observed between phosphorus content and predicted shift for low copper materials. The conclusion that there was any significant correlation was judged to be weak. Even in those cases where a possible trend with phosphorus content was suggested by the data, the differences between the observed and predicted shifts were within the margins applied by Regulatory Guide 1.99, Rev. 2. In addition, in the cases where a trend with phosphorus was suggested, it was generally the high phosphorus (>0.012%) materials which indicated the possible trend. Based on this previous assessment of the effects of impurity elements on the irradiation response of materials, the existing controls on residual elements are concluded to be sufficient. The limits on copper, phosphorus and other residual elements will minimize the extent of radiation damage to the RPV beltline materials. The radiation induced shifts in reference temperatures for these materials can be predicted with reasonable accuracy and conservatism using the methodology of Regulatory Guide 1.99, Rev. 2.

Open Item 5.4.1.1-5: ECGB

The applicant has not submitted the basis for the assumed design overspeed for staff review as recommended in SRP Section 5.4.1.1. The applicant describes the design speed as 125 percent of normal operating speed. The applicant should clarify that the design overspeed of a flywheel is at least 10 percent above the highest anticipated overspeed as stateu in SRP Section 5.4.1.1.

Response to Open Item 5.4.1.1-5:

ABB-CE will clarify the basis for design overspeed by revising CESSAR-DC, Section 5.4.1.1.B.2 as follows:

> "The design overspeed of the flywheel will be 125 percent of normal operating speed.
>
> The design overspeed will be at least 10% above the highest anticipated overspeed of the pump. The highest anticipated overspeed is predicted for the largest break size remaining after application of leak before break as described in Section 3.6."

Open Item 5.4.1.1-8:

ECGB

Although the guideline in RG 1.14, "Reactor Coolant Pump Flywheel Integrity," has been excerpted in CESSAR, the applicant should clarify in CESSAR that it intends to meet this regulatory guide.

Response to Open Item 5.4.1.1-8:

CESSAR-DC Section 5.4.1.1-2 will be revised to state that the flywheel will meet the requirements of RG 1.14 "Reactor Coolant Pump Flywheel Integrity." The manner by which ABB-CE will meet the requirements is shown in the attached marked-up copy of Section 5.4.1.1.

Ref: Open Items 5.4.1.1-5
5.4.1.1-8

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

The reactor coolant pumps provide sufficient forced circulation flow through the Reactor Coolant System to assure adequate heat removal from the reactor core during power operation. A low limit on reactor coolant pump flow rate (i.e., design flow) is established to assure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded. Design flow is derived on the basis of the thermal-hydraulic considerations presented in Section 5.2.

The reactor coolant pump and motor assembly in conjunction with the flywheel, provide sufficient coastdown flow following loss of power to the pumps to assure adequate core cooling.

The reactor coolant pump pressure boundary is designed for the transients given in Section 3.9 so that the ASME Code Section III allowable stress limits are not exceeded for the specified number of cycles. Stress criteria concerning earthquake and pipe rupture conditions are presented in Section 3.9.3.

The design overspeed of the reactor coolant pump is 125 percent of normal speed.

5.4.1.1 Pump Flywheel Integrity

A. The material used to manufacture the flywheel of the reactor coolant pump motor will be produced by a commercially acceptable process that minimizes flaws, such as the vacuum melt and degassing process. This provides adequate fracture toughness properties under reactor operating conditions. The acceptance criteria for flywheel design will be compatible with the safety philosophy of the Pressure Vessel Research Committee (PVRC) of the Welding Research Council (WRC) primary coolant pressure boundary criteria as appropriate considering the inherent design and functional requirement differences between the pressure boundary and the flywheel. D

1. The reference nil-ductility transition temperature (RT_{NDT}) of the material, as determined per ASME Code NB-2331(a), will be no greater than 10°F. D

2. The Charpy V-notch (Cv) upper shelf energy level, in the "weak" (Wr) direction, as obtained per ASTM-A-370 will be no less than 50 ft-lb. A minimum of three Cv specimens will be tested from each plate or forging.

The pump flywheel meets the requirements of RG 1.14 "Reactor Coolant Pump Flywheel Integrity" as stated below:

Ref: Open Items 5.4.1.1-5
5.4.1.1-8

3. The minimum static fracture critical toughness of the material at the normal operating temperature of the flywheel will be equivalent to a ~~dynamic~~ stress Intensity factor (K_{IC}) ~~(dynamic)~~ of at least ~~100~~ 150 ksi$\sqrt{in.}$ Compliance will be demonstrated by either of the following: I

   a. Testing of the actual material of the flywheel to establish the K_{IC} ~~(dynamic)~~ value at the normal operating temperature, or

   b. ~~Use of a lower bound fracture toughness curve obtained from tests on the same type of material. The curve will be translated along the temperature coordinate until the K_{IC} (dynamic) value of 45 ksi$\sqrt{in.}$ is indicated at the NDT of the material, as obtained from drop-weight tests.~~ Determining that the normal operating temperate is at least 100°F above the RT_{NDT} I

4. Each finished flywheel will be subjected to a 100 percent volumetric ultrasonic inspection from the flat surface per ASME BPVC Section III.

   This inspection will be performed on the flywheel after final machining and the overspeed test.

5. If the flywheel is flame cut, at least 1/2 inch of stock will be left on the outer and bore radii, for machining to final dimensions.

6. The flywheel will be subjected to a magnetic particle or liquid-penetrant examination per "Section III" before final assembly. The inspection will be performed on finished machined bores, keyways, splines and drilled holes. ~~and on both flat surfaces to a radial distance of 8 inches minimum beyond the final largest machined bore diameter but not including small drilled holes. There will be no stress concentrations such as stamp marks, center punch marks, or drilled or tapped holes within 8 inches of the edge of the largest flywheel bore.~~

B. The flywheels will be designed to withstand normal operating conditions, anticipated transients, and the largest mechanistic pipe break size remaining after application of leak before break as described in Section 3.6, combined with the Safe Shutdown Earthquake. D

The following criteria will be satisfied:

1. The combined stress, both centrifugal and interference, at normal operating speed will not exceed one-third of the minimum specified yield strength or 1/3 of the measured yield strength in the weak direction of the I

Ref: Open Items 5.4.1.1-5
5.4.1.1-8

material if appropriate tensile tests have been performed on the actual material of the flywheel. I

2. The design ~~speed~~ overspeed of the flywheel will be 125 percent of normal operating speed.

~~The lowest of the critical speeds of the flywheel~~ The design overspeed will be at least 10% above the highest anticipated overspeed of the pump. The highest anticipated overspeed is predicted for the largest break size remaining after application of leak before break as described in Section 3.6. D

3. The combined centrifugal and interference stresses at the design speed will be limited to two-thirds of the minimum specified yield strength or 2/3 of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel. Design speed is defined as 125 percent of normal operating speed. I

4. The motor and pump shaft or bearings and coupling will withstand any combination of normal operating loads or anticipated transients, and the largest remaining pipe break after application of leak before break as described in Section 3.6, combined with the Safe Earthquake Shutdown. D

Each flywheel will be tested at design speed, 125 percent of normal operating speed, as defined in B.2 above.

The flywheel will be accessible for 100 percent in-place volumetric ultrasonic inspection. The flywheel-motor assembly is designed to allow such inspection with a minimum of motor disassembly. The in-service inspection program will include ultrasonic examinations of the areas of high stress concentration at the bore and keyway at about 3 1/3 year intervals, during the refueling or maintenance shutdown coinciding with the in-service inspection schedule as required by the ASME Code, Section XI. Removal of the flywheel is not required. I

A surface examination of all exposed surfaces ~~Liquid penetrant examination or magnetic particle methods~~ and 100% volumetric examination by ultrasonic methods will be conducted at about ten-year intervals during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME Code, Section XI.

Add → Each flywheel will receive a preservice baseline inspection incorporating the methods defined above for an inservice inspection. Examination procedures and acceptance criteria will be in accordance with the ASME BPVC Section III.

Open Item 5.4.2-2:

The applicant should revise the CESSAR to describe the ISI program for steam generator tubes.

ABB-CE Response:

System 80+ Technical Specification SR 3.4.4.2 requires verification of steam generator tube integrity in accordance with the Steam Generator Tube Surveillance Program. The Steam Generator Tube Surveillance Program will be managed and implemented by the combined license applicant using controls similar to those used for the ISI Program. The initial Steam Generator Tube Surveillance Program shall be subject to a review and approval process equivalent to that required for the ISI Program, and changes to the surveillance program will be processed in the same manner as relief requests for the ISI Program. The surveillance program specifies the details of the inspection including tube selection and sampling (as well as sample expansion), inspection interval, inspection technique, the actions to be taken when degradation or defects are identified, and reporting requirements. These details are consisitent with the requirements of Regulatory Guides 1.83 and 1.121 supplemented by the recommendations of the industry-prepared "PWR Steam Generator Examination Guidelines, Revision 2" (EPRI Report NP6201, December 1988, or subsequent revisions) and where appropriate, industry-prepared technical support documents for degradation-specific repair criteria.

An example of an initial steam generator tube surveillance program (which may be revised as indicated above) is attached for information.

Steam Generator Tube Surveillance Program (Example)

The surveillance program for the steam generator tubes will include the following:

1. Steam Generator Sample Selection and Inspection - Each steam generator shall be determined operable during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 1.

2. Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Section 3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Section 4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

   a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

   b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

      1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

      2. Tubes in those areas where experience has indicated potential problems.

      3. A tube inspection in accordance with Section 4a.8. shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

   c. The tubes selected as the second and third samples (if required by Table 2) during each inservice inspection may be subjected to a partial tube inspection provided:

      1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

      2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

3. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

   a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under all volatile treatment (AVT) conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

   b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Section 3a.; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the Technical Specification 3.4.12.

2. A seismic occurrence greater than the Safe Shutdown Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

4. Acceptance Criteria

a. As used in this program:

1. Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal tube wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Safe Shutdown Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 3c., above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding action (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 2.

5. Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Two	Two
First Inservice Inspection	All	One
Second & Subsequent Inservice Inspection	One*	One*

TABLE NOTATION

*The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
SAMPLE SIZE	RESULT	ACTION REQUIRED	RESULT	ACTION REQUIRED	RESULT	ACTION REQUIRED
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1 C-2	None Plug defective tubes
			C-3	Perform action for C-3 result of first sample	C-3	Perform action for C-3 result of first sample
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.	All other S.G.'s are C-1	None	N.A.	N.A.
		Notification to NRC pursuant to 50.72 (b)(2) of 10 CFR Part 50	Some S.G.'s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to 50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A.

S = 3N/n % Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

ECGB

Open item 20.2-2:

"The applicant should describe materials selected for the construction of the reactor vessel supports, limits on residual elements to minimize susceptibility to irradiation, limits on initial reference temperature and upper-shelf impact energy, and inspection requirements of supports during fabrication".

Response. The reactor vessel support columns in ABB-CE plants are made of high quality SA508 steel, with additional restrictions on both its chemical composition and its post-fabrication inspection. The specific chemistry restrictions are: (1) maximum phosphorus, 0.012% per heat and 0.018% per product analysis, (2) maximum copper, 0.15% per heat and per product analysis. Other compositional requirements consistent with SA508 chemistry continue to apply.

The initial RTndt is specified as 40 Degrees F, maximum. In actual practice, initial RTndt values of 10--30 Degrees F are typically achieved. The upper-shelf impact energy is specified to meet the fracture toughness requirements of ASME Section III, Subsection NB-2300 at 40 Degrees F.

Post-fabrication inspection is performed in accordance with ASME Section III, Subsection NF, and ASME Section II, Specification SA508. Magnetic particle inspections in accordance with Method A275 are performed after final machining; and forgings are ultrasonically inspected in accordance with Recommended Practice A388.

ECGB

Open Item 20.2-3:

"The applicant should provide the estimated 60-year neutron fluence level at the reactor vessel support, which should be expressed in "displacements-per-atom" to account for the neutron energy as discussed in Reference 1."

$\boxed{\text{Response.}}$ The 60-year neutron fluence level is estimated to be 3.0 x 10**18 neutrons per square centimeter (E > 1.0 Mev). This is based on an 80 percent capacity factor, i.e. after 48 effective full power years. This fluence pertains to the surface of the support column facing the reactor, at core midplane. The actual fluence depends significantly on fuel management procedures employed over the life of the plant. The estimated fluence of 3.0 x 10**18 is based on conservative physics calculations, and could exceed the fluence realized in actual practice by 30% or more. This fluence corresponds to approximately 0.0045 dpa.

RTndt shifts for the reactor vessel supports can be reliably estimated using the methodology of Regulatory Guide 1.99, while including conservative correction factors to account for the effects of temperature and neutron energy spectra.

ECGB

Open Item 20.2-4:

"The applicant should describe its procedures in estimating the extent of irradiation embrittlement and provide the results. The applicant should provide technical justification for any procedures that are different from those in Reference 1."

Response. The effective fast fluence is used to calculate the irradiation-induced RTndt shift according to NRC Regulatory Guide 1.99, Revision 2. RTndt shifts are calculated based on fluences at the locations of hypothetical crack tips within a structure. Crack tip fluences are somewhat less than corresponding surface fluence values; the function describing the attenuation of fluence with depth is given in equation (3), part 1.1 of this Regulatory Guide. For RV column analyses, predictions of RTndt shift are based on SA508 chemistry for which additional impurity restrictions have also been specified. The operating temperature range of the RV column supports at core midplane, well below 400 Degrees F, is then addressed. Regulatory Guide 1.99, Revision 2 states that temperatures below 525°F should be considered to produce greater embrittlement than that predicted by its methodology. The available data indicate a constant effect from temperatures below 400 Degrees F, in terms of the RTndt shift that such colder temperatures would cause; in other words, this effect saturates below 400°F. The experimentally-observed RTndt shifts below 400°F exceed those at 550°F by somewhat more than a factor of two. Accordingly, a conservative temperature correction factor of 2.25 is applied to the RTndt shift predicted by RG 1.99 Rev. 2.

For the RV column analysis, the surface value for the RTndt shift is predicted to be 171 Degrees F, after 60 years (48 EFPY) and a fluence (E > 1.0 Mev) of 3.0 x 10**18; this prediction includes the conservate factor of 2.25 for colder temperatures.

ECGB

Open Item 20.2-5:

"The applicant should provide additional information on its fracture mechanics analysis, including assumptions and acceptance criteria."

Response. The fracture mechanics evaluation of the RV column considers hypothetical cracks located at core midplane, one on the side facing the reactor, one on the side facing away. The method of ASME Section XI is used to determine an Applied Stress Intensity Factor, Ki, associated with a hypothetical crack tip, using Design Condition static forces and moments in the column at core midplane, plus dynamic loadings from SSE. Since the RV columns are fabricated from SA508, ASME Section III Appendix G is then invoked. Figure G-2210-1 determines the minimum acceptable column temperature relative to as-irradiated RTndt. The use of ASME III Appendix G requires a further conservatism in that an applied Ki associated with any primary membrane or primary bending stress is doubled before entering Figure G-2210-1. Figure G-2210-1 then determines the miniumu acceptable algebraic difference between the actual RV column temperature, and the end-of-life (EOL) as-irradiated RTndt. This algebraic temperature difference is then added to an additional margin requirement from 10 CFR 50, Appendix G. An acceptab[illegible]es that the actual RV column temperature, minus the a[illegible]Tn[illegible] must exceed the sum of (1) minimum acceptable algeb[illegible] ASME Section III, Figure G-2210-1, plus (2) the 10 [illegible] 50, [illegible]x G margin requirement. Algebraically this is summar[illegible] follows:

Initial RTndt + RTndt shift = As-irradiated RTndt

(where the RTndt shift is conservatively predicted with the factor of 2.25 for colder temperatures);

RV Column Temperature, MINUS, As-irradiated RTndt > (IS GREATER THAN)

ASME III, Appendix G, Figure G-2210-1 required algebraic difference,

PLUS, 10 CFR 50, Appendix G Margin requirement

Since the ASME III, Appendix G requirement ultimately depends upon the dimensions of any hypothetical crack, the above inequality is then tested against crack dimensions which are increasingly larger until the inequality can no longer satisfied; this determines a limiting crack dimension. An acceptable result for the RV column supports is indicated when this limiting crack size is shown to be larger than the post-fabrication inspection flaw detection limits, since any detected flaw must be repaired prior to certifying the RV column supports as acceptable.

ECGB

Open Item 20.2-6

If the fracture mechanics analysis for the reactor vessel supports is based on LBB assumptions, the applicant should provide technical justifications.

Response:

Preliminary LBB evaluations are provided in accordance with staff requirements for approval of LBB application to System 80+ designed piping. With LBB satisfied for selected piping systems, the dynamic effects from postulated pipe breaks in these piping systems are eliminated.

Materials and Chemical Engineering Branch

NUMBER	TYPE	TITLE	BRANCH
03.09.6-1	COL ITEM	The COL applicant should develop an in-service testing program for ASME Code Class 1, 2 and 3 pumps and valves.	EMCB
06.1.1-1	COL ITEM	The COL applicant will review vendor fabrication procedures to ensure that unstabilized austenitic stainless steel is not exposed to improper temperature range.	EMCB
06.1.1-2	COL ITEM	The COL applicant w ll perform grinding with resin- or rubber-bonded aluminum oxide or silicon carbide wheels that have not been previously used on other materials.	EMCB
06.1.1-3	COL ITEM	The COL applicant will add hydrazine to inhibit flushing water to prevent halide-induced intergranular corrosion.	EMCB
06.1.1-4	COL ITEM	The COL applicant will follow the recommendations of RG 1.50 and Section III of the ASME Code.	EMCB
06.6-1	COL ITEM	The COL applicant should submit PSI and ISI program plans for staff review and approval.	EMCB
03.09.6.1-1	CONF ITEM	The staff will confirm that commitments relating to inservice testing of pumps will be incorporated into the CESSAR.	EMCB
03.09.1-1	OPEN ITEM	The applicant should modify its commitment to perform fatigue analyses in accordance with NRC-approved methods.	EMCB
03.09.1-2	OPEN ITEM	The applicant should revise the number of cycles specified for the transients in CESSAR Table 3.9-1 for 60-year design life.	EMCB
03.09.1-3	OPEN ITEM	The applicant should justify only considering those design transients that do not require forced shutdown in ASME Code CS component designs.	EMCB
03.09.1-4	OPEN ITEM	The applicant should revise CESSAR Section 3.9.1.2.3 to identify computer programs used in stress analyses for non-NSSS components.	EMCB
03.09.1-5	OPEN ITEM	The applicant should clarify CESSAR Section 3.9.1.3.	EMCB
03.09.1-6	OPEN ITEM	The applicant should include the forces associated with postulated pipe breaks in the Level D analyses.	EMCB
03.09.1-7	OPEN ITEM	The applicant should revise CESSAR Section 3.9.1.4.1 to include LOCA loads in the evaluation of the RCS faulted condition.	EMCB
03.09.2.1-1	OPEN ITEM	The applicant should justify the applicability of the piping displacement stresses on which limits for the 60-year life are based.	EMCB
03.09.2.1-2	OPEN ITEM	The applicant should revise CESSAR Section 3.9.2.1 to include non-ASME Code piping systems identified in SRP 3.9.2, in the preoperational test program.	EMCB
03.09.2.3-1	OPEN ITEM	The applicant should provide information to validate the designation of Palo Verde as the prototype for System 80+.	EMCB
03.09.2.3-2	OPEN ITEM	The applicant should justify the use of the factor of 3 X RMS to account for considering peak responses.	EMCB
03.09.2.4-1	OPEN ITEM	The applicant should evaluate faulted condition for RX internals and unbroken loops of reactor coolant piping in accordance with SRP Section 3.9.2.	EMCB
03.09.4-1	OPEN ITEM	The applicant should revise CESSAR 3.9.4.1 to include justification of the adequacy of control element assembly travel requirement for a 60-year design life.	EMCB
03.09.4-2	OPEN ITEM	The applicant should revise CESSAR Section 3.9.4.2 to specify that pressure boundary portions of CEDM are constructed in accordance with the ASME Code.	EMCB
03.09.4-3	OPEN ITEM	The applicant should address Palo Verde CEDM problems to ensure system 80+ is not susceptible to the same problems.	EMCB
03.09.4-4	OPEN ITEM	The applicant should specify the 1974 edition of IEEE 323 as endorsed by RG 1.89.	EMCB
03.09.4-5	OPEN ITEM	The applicant should revise CESSAR Section 3.9.4.3 and Table 3.9-15 to include LOCA loadings.	EMCB
03.09.4-6	OPEN ITEM	The applicant should specify in CESSAR Table 3.9-15 that NUREG-0484 load combination methodology applies to all dynamic loads.	EMCB
03.09.5-1	OPEN ITEM	The applicant should include CESSAR Section 3.9.5.3.2 Item B loads in the loadings identified in CESSAR Section 3.9.5.2 as requested in RAI Q210.75.	EMCB
03.09.5-2	OPEN ITEM	The applicant should justify the lack of identification of any Level C conditions for the applicable reactor internal components.	EMCB
03.09.5-3	OPEN ITEM	As requested in RAI Q210.76, the proposed revision to CESSAR Table 3.9-16 should state that "construction" is as defined by ASME Code Section III, NG-1100(a).	EMCB
03.09.5-4	OPEN ITEM	The applicant should revise CESSAR Section 3.9.5.4 fatigue analysis basis to be consistent with the response to RAI Q210.52.	EMCB
03.09.6-1	OPEN ITEM	The staff requires that a schedule for the IST program for equipment and components be submitted for review.	EMCB
03.09.6-2	OPEN ITEM	Staff believes there is sufficient lead time for the applicant to include provisions in piping systems to accomodate IST requirements.	EMCB
03.09.6.1-1	OPEN ITEM	The applicant must develop a program to establish frequency and extent of disassembly and inspection of pumps and valves.	EMCB
03.09.6.2-1	OPEN ITEM	The applicant's response regarding full-flow testing of check valves needs to be revised.	EMCB
03.09.6.2-2	OPEN ITEM	The applicant's response to forward and reverse flow testing of check valves should be revised in accordance with RAI Q210.81(e).	EMCB
03.09.6.2-3	OPEN ITEM	The applicant should revise response to RAI Q210.81(e) in accordance with the staff position on the use of non-intrusive diagnostic techniques.	EMCB
03.09.6.2-4	OPEN ITEM	The staff requests that the response to RAI Q210.81(g) be revised in accordance with the staff position stated in the RAI.	EMCB
03.09.6.2-5	OPEN ITEM	The applicant's response to RAI Q210.81(f) is not acceptable because it does not address the staff's concerns.	EMCB
03.09.6.2-6	OPEN ITEM	The applicant should commit to an analysis of the leakage rates and corrective action requirements contained in the ASME Code.	EMCB
03.09.6.2-7	OPEN ITEM	Until the TS are approved, the staff considers the valve list and surveillance requirements of the pressure isololation valves to be an open item.	EMCB

NUMBER	TYPE	TITLE	BRANCH
05.3.1-08	OPEN ITEM	The applicant should limit the sulfur content in welds and forgings to less than 0.01 percent and specify the heat treatments.	EMCB
06.5-3	OPEN ITEM	The applicant should verify that the containment spray system is designed to operate for at least 2 hours in all cases.	EMCB
06.6-1	OPEN ITEM	All Class 2 and 3 components requiring ASME Section XI inspections must be accessible.	EMCB
06.6-2	OPEN ITEM	The applicant must state that PSI will meet Section XI of same edition of ASME code used for construction and ISI will be in accordance with 10 CFR 50.55a(g).	EMCB
06.6-3	OPEN ITEM	The applicant should clarify that all PSI examination requirements will be practical (no relief will be permitted).	EMCB
06.6-4	OPEN ITEM	The applicant should revise the CESSAR to define the division of responsibility for PSI and ISI.	EMCB
06.6-5	OPEN ITEM	The PSI and ISI should be conducted using equivalent equipment and techniques.	EMCB
06.6-6	OPEN ITEM	The staff is recommending the use of ASME Section XI Appendices VII and VIII for the CESSAR.	EMCB
10.2-1	OPEN ITEM	The applicant should identify deviations from ASTM A-370 and provide technical justification.	EMCB
10.2-2	OPEN ITEM	The applicant has not specified the acceptance criteria for the low pressure turbine disk material fracture toughness.	EMCB
10.2-3	OPEN ITEM	The applicant has not cited the specific method used to determine the fracture toughness.	EMCB
10.2-4	OPEN ITEM	The applicant has not described the method of determining the yield strength of the material of the turbine generator wheels and rotors.	EMCB
10.2-5	OPEN ITEM	The applicant should commit to a turbine disk design that facilitates ISI of all high stress areas without removing the disks from the shaft.	EMCB
10.2-6	OPEN ITEM	The applicant should justify the use of acceptance criteria not specified in ASME Sections III and V.	EMCB
10.2-7	OPEN ITEM	The applicant should state whether a shrunk-on disk design or a one-piece forged rotor design will be used.	EMCB
10.3-3	OPEN ITEM	The applicant has not met RG 1.71 and the proposed alternative does not adequately address the staff's concerns regarding limited accessibility.	EMCB
10.3-4	OPEN ITEM	The applicant should clarify its intent regarding the use of stainless steel materials in the steam and feedwater systems.	EMCB
10.3-5	OPEN ITEM	The applicant has not identified specific materials for use in the steam and feedwater systems.	EMCB
10.3-6	OPEN ITEM	The applicant should provide a corrosion allowance for a 60-year design life for the steam and feedwater systems.	EMCB
10.3-7	OPEN ITEM	The applicant should describe the methodologies for identifying the corrosion/erosion susceptible locations and for selecting resistant materials.	EMCB
10.3-8	OPEN ITEM	The applicant should explicitly account for the effects of the environment in the fatigue analysis of the steam and feedwater system components.	EMCB
10.3-9	OPEN ITEM	The applicant should consider the effects of dynamic strain aging on the steam and feedwater system components.	EMCB
10.4.8-1	OPEN ITEM	The applicant should specify if the blowdown piping material is compatible with a 2 phase blowdown fluid that may have a high erosion/corrosion potential.	EMCB
20.2-01	OPEN ITEM	The applicant should reference SRP Section 6.6, and CESSAR Section 6.6 in its response to Generic Issue 14.	EMCB

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Open Item 3.9.1-4

The applicant should revise CESSAR Section 3.9.1.2.3 to identify computer programs used in stress analyses for non-NSSS components.

Response:

CESSAR Section 3.9.1.2.1 discusses the computer programs used in the stress and structural analyses for the reactor coolant as well as non-reactor coolant systems, components and supports. Section 3.9.1.2.3. should be deleted. The computer programs used in the analysis of structures are discussed in Section 3.7 of CESSAR and are not to be included in Section 3.9.

CESSAR Section 3.9.1.2.3 will be deleted and section 3.9.1.2.1 will be revised, as attached, to cover Code Class systems, components and supports rather than just the reactor coolant system.

In addition to the design transients listed above and included in the fatigue analysis, the loadings produced by the OBE and SSE were also applied in the design of components and support structures of the RCS. The OBE and SSE are classified as upset and faulted condition events respectively. For the number of cycles pertaining to the OBE, refer to Section 3.7.3.2.

3.9.1.2 Computer Programs Used in Stress Analyses

3.9.1.2.1 CODE CLASS SYSTEMS, COMPONENTS, AND SUPPORTS ~~Reactor Coolant System~~

The following paragraphs provide a summary of the applicable computer programs used in ~~the~~ STRESS AND structural analyses for ASME Code Class ~~1~~ systems, components, ~~and~~ supports in the CESSAR-DC scope. The summaries include individual descriptions and applicability data. The computer codes employed in these analyses have been verified in conformance with design control methods, consistent with the quality assurance program described in Chapter 17.

3.9.1.2.1.1 MDC STRUDL

The MDC STRUDL computer program provides the ability to specify characteristics of framed structure and three-dimensional solid structure problems, perform static and dynamic analyses, and reduce and combine results.

Analytic procedures in the pertinent portions of MDC STRUDL apply to framed structures. Framed structures are two- or three-dimensional structures composed of slender, linear members that can be represented by properties along a centroidal axis. Such a structure is modeled with joints, including support joints, and members connecting the joints. A variety of force conditions on members or joints can be specified. The member stiffness matrix is computed from beam theory. The total stiffness matrix of the modeled structures is obtained by appropriately combining the individual member stiffness.

The stiffness analysis method of solution treats the joint displacements as unknowns. The solution procedure provides results for joints and members. Joint results include displacements and reactions and joint loads as calculated from member end forces. Member results are member end forces and distortions. The assumptions governing the beam element representation of the structure are as follows: linear, elastic, homogeneous, and isotopic behavior, small deformation, plane sections remain plane, and no coupling of axial, torque, and bending.

The program is used to define the dynamic characteristics of the structural models used in the dynamic seismic analyses of the reactor coolant system components. The natural frequencies and

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DSER OPEN ITEM
3.9.1-4

CEFLASH-4B provides transient pressures, flow rates and densities throughout the primary system following a postulated pipe break in the reactor coolant system.

The CEFLASH-4B computer code is a modified version of the CEFLASH-4A code (References 15 through 17). The CEFLASH-4A computer code has been approved by the NRC (References 18 and 19). The capability of CEFLASH-4B to predict experimental blowdown data is presented in Reference 14.

3.9.1.2.2.9 LOAD

LOAD calculates the applied forces of the axial internals model which is contained within water control volumes using results from the CEFLASH-4B blowdown loads analysis as input. The fluid momentum equation is applied to each volume and a resultant force is calculated. Each force is then apportioned to the various structural nodes contained within the volume. Use of the fluid momentum equation takes into account pressure forces, fluid friction, water weight, and momentum changes within each volume. The resultant forces are combined with the reactor vessel motions obtained from the reactor coolant system analysis before the structural responses are determined. The LOAD code has been verified by demonstrating that its solutions are substantially identical to those obtained from hand calculations.

DELETE

~~3.9.1.2.3 Non-NSSS Structures and Components~~

~~Computer programs used in the analysis of non-NSSS structures are discussed in Sections 3.7.1.4, 3.8.2.4 and Appendix 3.7B.~~ I

3.9.1.3 Experimental Stress Analyses

Requirements for experimental stress analysis have not been imposed on any equipment in the CESSAR-DC scope.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition E

3.9.1.4.1 Seismic Category I RCS Items

The major components of the reactor coolant system (RCS) are designed to withstand the forces associated with the design basis pipe breaks discussed in Section 3.6, in combination with the forces associated with the Safe Shutdown Earthquake and normal operating conditions. For structural evaluation, the design basis pipe breaks are those breaks for which leak-before-break cannot be demonstrated. Since the dynamic effects of breaks in

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Open Item 3.9.2.4-1

The applicant should evaluate faulted condition for reactor internals and unbroken loops of reactor coolant piping in accordance with SRP Section 3.9.2.

Response:

Structural evaluations of the reactor internals and fuel were performed for primary side branch line pipe breaks (BLPB) and safe shutdown earthquakes (SSE) to demonstrate that ten percent of the SSE provides a conservative upper bound for these BLPB effects on the reactor internals when these loads are combined with the SSE loads by the SRSS method. The BLPB evaluations were performed for a three inch (3") spray line break and a six inch (6") safety valve nozzle break in both the axial and horizontal direction using non-linear System 80 structural models of the internals and fuel. These models were subjected to time dependent blowdown loads and reactor vessel motions obtained from reactor coolant system evaluations. For the horizontal direction responses, the analyses were performed both parallel and perpendicular to the hot legs and the maximum shear loads and bending moments were combined by the SRSS method. The response loads were determined using the CESHOCK computer code (CESSAR-DC, Section 3.9.1.2.2.4).

The SSE evaluations were also performed in the horizontal directions for hard (rock) and soft soil conditions using the same structural models. These extreme soil conditions were chosen since they represent a range of sites and provide a measure of structural responses for the various conditions. The structural responses were again determined using the CESHOCK code for SSE time durations of up to 30 seconds. As expected, the structural responses were worse for the rock soil. The peak responses were then combined by the SRSS method, for each soil condition, and were compared to the corresponding BLPB loads to determine the validity of the proposed BLPB and SSE load combination criterion.

The stated criterion considers these BLPB loadings to be negligible, such that, when combined with SSE loadings, the result will be less than a 10% increase in Faulted Condition design loads. When the combined responses were compared to this criterion for both soil conditions, the results showed that all component loadings were easily satisfied. Also, for the vertical direction responses, a comparison of the reactor vessel SSE motion spectra for the soft soil site is worse than that for the Palo Verde nuclear power plant site. Since the use of the lower Palo Verde SSE loads in combination with the System 80+ BLPB loads also meets the proposed criterion, the use of the System 80+ SSE loads also meets the criterion.

Section 3.9.2.5 of CESSAR-DC is not intended to address evaluation of RCS unbroken loops under faulted conditions, only evaluation of reactor internals. The statement applying 10% of SSE effects for BLPB is only intended to apply to reactor internals loads. Following application of LBB to main coolant loop, main steam and branch line piping, the

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largest primary side pipe break that loads the RCS unbroken loops is the three-inch spray line break. The loading effects of this break on main coolant loop piping, which is 36 inches (cold leg) and 49 inches (hot leg) in outside diameter, is negligible.

EMC6

Open Item 3.9.4-1

The applicant should revise CESSAR 3.9.4.1 to include justification on the adequacy of control element assembly travel requirement for a 60-year design life.

Response:

CEDM materials were used in an inextensively tested CEDm assembly that exceeded lifetime requirements, as described in Section 3.9.4.4.1. The design duty or lifetime requirement as defined in CESSAR Section 3.9.4.1 is a total cumulative CEA travel of 100,000 feet of operation without loss of function and not the 60-year plant life. As indicated in CESSAR Section 3.9.4.1, the CEDM is designed to operate without maintenance for a minimum of 1-1/2 years and without replacing components for a minimum of 3 years. Therefore the test results of the extensively tested CEDM do not need to be verified to the 60-year plant life.

The operational requirement for the System 80+ CEA's, with the possible exception of the lead regulating CEA group, is expected to be less than the 100,000 feet of travel (the tested life) over the 60 year plant life. If plants institute daily load cycle operation on a regular basis, the lead regulating CEA group may exceed 100,000 feet of travel.

The regulating CEA's are much lighter than the CEA weight used during accelerated CEDM motor life tests, and it is expected that, when operating a regulating CEA, the System 80+ CEDM motors are capable of operation in excess of 100,000 feet of cumulative travel. Depending on the extent that the lead regulating CEA group is utilized, a one time CEDM motor replacement for this bank of CEA's may be required during the 60-year plant life.

As indicated in CESSAR Section 3.9.4.1, all CEDM pressure boundary components have a design life of 60 years.

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Open Item 3.9.4-3

The applicant should address Palo Verde CEDM problems to ensure System 80+ is not susceptible to the same problems.

Response:

Problems were encountered with the lower latch coils in the Palo Verde CEDM coil stack assemblies. Inductive coupling between the lower lift coil and the lower latch coil caused movement in the lower latch coil. This motion resulted in abrasion of the insulating varnish applied to the coil leads, causing intermittent grounding. The problem was corrected by the addition of insulating jackets to the coil leads and by reversing the polarity of the lower latch coil, which significantly reduced the intensity of adverse motion. These changes were instituted at Palo Verde and incorporated into the present CEDM design.

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Open Item 3.9.4-5

The applicant should revise Cessar Section 3.9.4.3 and Table 3.9-15 to include LOCA loadings.

Response:

The required changes are indicated on the attached marked pages of the CESSAR (2 pages). The changes will be incorporated into the next amendment of the CESSAR.

Response To open
iTem 3.9.4-5

709-3.doc(8814)bh-24.1

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3.9.4.3 continuation

C. Dynamic stresses produced by seismic loading and design bases pipe breaks and/or LOCA loading

D. Dynamic stresses produced by mechanical excitations

Full length RSPT assemblies are subjected to biaxial random multi-frequency input motions corresponding to design bases excitations. Testing is performed using four RSPT orientations to account for asymmetries in the design.

E. Loads produced by the operation and tripping of the mechanism

F. Dynamic stresses produced by excitations from pipe breaks other than those eliminated by LBB.

The methods used to demonstrate that the CEDMs operate properly under seismic conditions are presented in Section 3.7.3.14.

The design and fabrication of the CEDM pressure boundary components fulfills the requirements of the ASME Code, Section III, for Class I vessels. The pressure housings are capable of withstanding throughout the design life all the steady state and transient operating conditions specified in Table 3.9-16.

The adequacy of the design of the CEDM pressure boundary and non-pressure boundary components has been verified by prototype accelerated life testing as discussed in Section 3.9.4.4.

Clearances for thermal growth and for dimensional tolerances were investigated, and tests have proven that adequate clearances are provided for proper operation of the CEDM.

The latch locations are set by a master gauge, and settings are verified by testing at reactor conditions.

A weldable seal closure, per Section III of the ASME Code, is provided for the vent valve in case of leakage.

The motor housing fasteners are mechanically positively captured, and all threaded connections are preloaded before capturing.

The coil stack assembly can be installed or removed simply by lowering or lifting the stack, relative to the CEDM pressure housing, for ease of coil replacement or maintenance.

EMCB

Response To open item 3.9.4-5

TABLE 3.9-15

STRESS LIMITS FOR CEDM PRESSURE HOUSINGS

Operating Condition	Stress Categories and Limits of Stress Intensities (a)(b)
1. Level A and Level B: Normal Operating Loading plus Normal Operating & Upset Plant Transients plus Operating Basis Earthquake Forces.	Figures NB-3221-1 and 3222-1, including notes.
2. Level D: Normal Operating Loadings plus Faulted Plant Transients plus Safe Shutdown Earthquake Forces Plus Design Bases Pipe Breaks and/or LOCA loads	Article F-1000, Appendix F, Rules for Evaluation of Service Conditions Loading with Level D Service Limits~~(a)~~.
3. Testing: Testing Plant Transients	Paragraph NB-3226

For the above listed operating conditions, the following limits regarding function apply:

1. Level A and Level B: The CEDMs are designed to function normally during and after exposure to these conditions.

2. Level D: For SSE, the deflections of the CEDM pressure housing are limited to the elastic design limits of Article F-1330, Appendix F (defined above) so that the CEAs can be inserted after exposure to these conditions.

NOTES: a. References listed are taken from Section III of the ASME Boiler and Pressure Vessel Code.

b. ~~Level D~~ dynamic loads ~~due to~~ including SSE, LOCA and design bases pipe breaks are combined by the SRSS method in accordance with the guidelines of NUREG-0484.

EMCB

Open Item 3.9.4-6

The applicant should specify in CESSAR Table 3.9-15 that NUREG-0484 load combination methodology applies to all dynamic loads.

Response:

The required changes are indicated on the attached marked pages of the CESSAR (1-page). The changes will be incorporated into the next revision of the CESSAR.

EMCB Response To open ITem 3.9.4-6

TABLE 3.9-15

STRESS LIMITS FOR CEDM PRESSURE HOUSINGS

	Operating Condition	Stress Categories and Limits of Stress Intensities (a) (b)
1.	Level A and Level B: Normal Operating Loading plus Normal Operating & Upset Plant Transients plus Operating Basis Earthquake Forces.	Figures NB-3221-1 and 3222-1, including notes.
2.	Level D: Normal Operating Loadings plus Faulted Plant Transients plus Safe Shutdown Earthquake Forces Plus Design Bases Pipe Breaks and/or LOCA loads.	Article F-1000, Appendix F, Rules for Evaluation of Service Conditions Loading with Level D Service Limits ~~(b)~~.
3.	Testing: Testing Plant Transients	Paragraph NB-3226

For the above listed operating conditions, the following limits regarding function apply:

1. Level A and Level B: The CEDMs are designed to function normally during and after exposure to these conditions.

2. Level D: For SSE, the deflections of the CEDM pressure housing are limited to the elastic design limits of Article F-1330, Appendix F (defined above) so that the CEAs can be inserted after exposure to these conditions.

NOTE:

a. References listed are taken from Section III of the ASME Boiler and Pressure Vessel Code.

b. ~~Level D~~ dynamic loads ~~due to~~ including SSE, LOCA and design bases pipe breaks are combined by the SRSS method in accordance with the guidelines of NUREG-0484.

EMCB

Open Item 3.9.5-4

...In the design of critical reactor vessel internals, the applicant states in CESSAR Section 3.9.5.4 that for components subject to fatigue, the stress analysis will be performed utilizing the design fatigue curve of I-9-2 of Section III of the ASME Code. The staff has raised concerns relating to possible detrimental environmental effects not currently reflected in current ASME Code design fatigue curves (see RAI Q210.52). Accordingly, the applicant should revise the basis for the fatigue evaluation of components of reactor vessel internals subject to fatigue described in Section 3.9.5.4 to be consistent with the applicants's response to RAI Q210.52 and as discussed in Section 3.9.2 of this DSER.

Response:

The design of reactor core supports will address the potential influence of environmental effects on the fatigue life of materials over the 60 year design life.

The issue of environmental effects on fatigue is currently under consideration by a special Steering Committee for Cyclic Life and Environmental Effects in Nuclear Applications of the Pressure Vessel Research Council (PVRC). These activities were initiated based on requests from the ASME Boiler & Pressure Vessel (BP&V) Code Committee and the Board on Nuclear Codes & Standards (BNCS). The charter of the PVRC Steering Committee is to provide guidance and direction related to determining the effects of light water reactor (LWR) service environments on the cyclic life properties of applicable materials. The Steering Committee is also evaluating application methodologies that include these effects in the fatigue analysis process.

Preliminary recommendations were provided to the BNCS in September 1992. The initial findings reported to BNCS were that the current S/N curves should be appropriate for PWR environments. There was not complete agreement of the Steering Committee on this position and the issue is not yet finally resolved. ABB-CE will continue to monitor the industry activities on the fatigue curves and fatigue analysis methodoloy.

System 80+ components will be designed to ASME B&PV Code rules. If the influence of environmental effects has not been incorporated into the Code rules at the time of design, the potential effects will be addressed based on the technical understanding of the materials data and anticipated operating conditions.

EMCB

Open Item 3.9.6.2-7

Until the TS are approved, the staff considers the valve list and surveillance requirements of the pressure isolation valves to be an open item.

Response:

Revised System 80+ Technical Specifications are provided in Amendment K to CESSAR-DC, of which an advance copy has been provided to the staff. Therefore, ABB-CE considers this item closed. This Open Item duplicates Open Item 20.2-15.

EMCB

5.3.1-8 The applicant should limit the sulfur content in welds and forgings to less the 0.01 percent and specify the heat treatments.

RESPONSE

The sulfur content of reactor vessel beltline forgings and welds is controlled to a maximum of 0.015%. This limit is the same as is specified for the restricted chemistry limits for SA 508 Class 3 forgings which are used for the reactor pressure vessel beltline region.

Data compiled in EPRI Report NP-933 indicates that this control on the level of sulfur will provide the fracture toughness required to ensure the structural integrity of the reactor vessel as specified by 10 CFR 50, Appendix G.

The heat treatment applied to the forgings is as follows:

Austenitizing:	1550-1650°F for 4 hours followed by water quenching.
Tempering:	1200°F +/-25°F for 4 hours.
Post-Weld Stress Relief:	1125°F +/-25°F for up to 40 hours.

The heat treatment for welds involves only the Post-Weld Stress Relief.

Confirmatory Item 20.2-2

The staff will confirm that limiting conditions for operation and surveillance requirements are included in the T.S. in accordance with RG 1.133.

ABB-CE Response

In conformance with NUREG-1432 the Loose Parts Monitoring System has been eliminated as a TS item in the System 80+ Technical Specifications. This confirmatory item should therefore be closed.

Technical Specifications Branch

NUMBER	TYPE	TITLE	BRANCH
✓✓ 20.2-02	CONF ITEM	The staff will confirm that limiting conditions for operation and surveillance requirements are included in the TS in accordance with RG 1.133.	OTSB
# 16-1	OPEN ITEM	The applicant must resubmit the proposed TS after the resolution of NUREG-1432.	OTSB
# 16-2	OPEN ITEM	The applicant must consider items for TS that go beyond the scope of present plants, such as the safety depressurization system and the alternate ac power source.	OTSB

Performance and Quality Evaluation Branch

NUMBER	TYPE	TITLE	BRANCH
14.2.01-1	COL ITEM	The COL applicant will need to provide a startup administrative manual (procedures).	RPEB
14.2.02-1	COL ITEM	The COL applicant will provide organization and staffing information.	RPEB
14.2.03-1	COL ITEM	The initial test procedures will be the responsibility of the COL applicant.	RPEB
14.2.04-1	COL ITEM	The initial test program will be conducted by the COL applicant.	RPEB
14.2.06-1	COL ITEM	The COL applicant must retain test records for the life of the plant.	RPEB
14.2.09-1	COL ITEM	The trial use of plant operating and emergency procedures is a COL applicant action item.	RPEB
14.2.10-1	COL ITEM	The overall direction, coordination and control of the initial fuel loading will be the responsibility of the COL applicant.	RPEB
14.2.11-1	COL ITEM	The COL applicant must provide the test program schedule.	RPEB
14.2.12.3-1	COL ITEM	The COL applicant will need to provide scoping documents containing testing objectives and acceptance criteria.	RPEB
14.2.12.3-2	COL ITEM	The COL applicant must provide the startup administrative manual and any other documents which list plant conditions required during testing.	RPEB
14.2.12.3-3	COL ITEM	The COL applicant must provide reconcilation methods for test conditions where testing is not performed with representative design operating conditions.	RPEB
14.2.12.3-4	COL ITEM	The COL applicant will need to provide the approved preoperational and startup test procedures 60 days prior to their use.	RPEB
17.3.1.2-1	COL ITEM	The COL applicant must develop and implement an O-RAP.	RPEB
14.2.07-1	CONF ITEM	The staff will confirm that RG 1.95 is included in a revision to CESSAR Section 14.2.	RPEB
14.2.07-2	CONF ITEM	The staff will confirm that RG 1.139 will be included in a revision to CESSAR Section 14.2.7.	RPEB
14.2.12.1-1	CONF ITEM	The applicant's test abstracts provided in a February 18, 1992 letter were reviewed and found acceptable by the staff. These abstracts should be included in the CESSAR.	RPEB
14.2.12.1-2	CONF ITEM	The staff will confirm that the identification of startup test 14.2.12.2.1 is incorporated into a revision of the CESSAR since it is used during the post-core load testing.	RPEB
14.2.12.2-1	CONF ITEM	The applicant's exception to RG 1.68, Appendix A, Item 1.h.(5) should be identified in CESSAR Section 14.2.7.	RPEB
14.2.12.2-2	CONF ITEM	The applicant's exception to RG 1.68, Appendix A, Item 1.I.(21) should be identified in CESSAR Section 14.2.7.	RPEB
14.2.12.2-3	CONF ITEM	The applicant's exception to RG 1.68, Appendix A, Item 1.n.(15) should be identified in CESSAR Section 14.2.7.	RPEB
14.2.12.2-4	CONF ITEM	The applicant's exception to RG 1.68, Appendix A, Item 4.i should be identified in CESSAR Section 14.2.7.	RPEB
14.2.12.3-1	CONF ITEM	The staff will confirm that CESSAR Section 14.2.12.4.7 will be revised as previously proposed.	RPEB
14.2.12.3-2	CONF ITEM	The staff will confirm that CESSAR Section 14.2.12.1.88 is revised as previously proposed.	RPEB
14.1-1	OPEN ITEM	Review of ITAAC will be provided in the FSER.	RPEB
14.2.05-1	OPEN ITEM	CESSAR Section 14.2.5 or 14.2.11 should clearly state that Phase I testing is required prior to fuel loading.	RPEB
14.2.07-1	OPEN ITEM	The applicant needs to describe the stringent requirements imposed to allow reduced startup testing.	RPEB
14.2.12.1-1	OPEN ITEM	The applicant should identify the support systems required to be available for designated test phases.	RPEB
14.2.12.1-2	OPEN ITEM	The staff will review the revised test acceptance criteria.	RPEB
14.2.12.2-1	OPEN ITEM	The staff believes that personnel monitors and radiation survey instruments should be identified as COL action items.	RPEB
14.2.12.2-2	OPEN ITEM	The staff will determine the acceptability of new CESSAR Section 14.2.12.2.11 when the revision is made.	RPEB
14.2.12.2-3	OPEN ITEM	The test abstract is required for the control rod withdrawal and insertion sequencer testing, and the control rod inhibit or block function testing.	RPEB
14.2.12.2-4	OPEN ITEM	The applicant must address the ability of all auxiliary systems required to support operation of ESF to perform under limiting accident conditions.	RPEB
14.2.12.3-1	OPEN ITEM	The applicant's exception to RG 1.68.3, Position C.9 should be identified in CESSAR Section 14.2.7.	RPEB
17.1.3-1	OPEN ITEM	The applicant should revise CESSAR Table 1.8-1 to delete reference to RGs 1.30, 1.58, 1.64, 1.74, 1.88, 1.123, 1.144 and 1.146. These RGs have been incorporated into RG 1.28.	RPEB
17.1.3-2	OPEN ITEM	The applicant should revise CESSAR Table 1.8-1 to reference NQA-2 vice ANSI standards referenced in RGs 1.30, 1.37, 1.38, 1.39, 1.94 and 1.116.	RPEB
17.1.4-1	OPEN ITEM	The applicant should identify the core support structures that are not important to safety.	RPEB
17.1.4-2	OPEN ITEM	The applicant should justify the exclusion of new fuel racks, and hydrogen igniters from 10CFR50 Appendix B QA requirements.	RPEB
17.1.4-3	OPEN ITEM	The applicant must state the QA requirements for non-Appendix B items.	RPEB
17.1.4-4	OPEN ITEM	The applicant must state the QA requirements for ATWS equipment.	RPEB
17.1.4-5	OPEN ITEM	The reference to RAI Q270.1 in response to RAI Q260.7 needs to be clarified by the applicant.	RPEB
17.1.4-6	OPEN ITEM	The applicant should revise its response to RAI Q260.27.	RPEB
17.1.4-7	OPEN ITEM	A portion of the response to RAI Q260.28 should be included in the CESSAR.	RPEB
17.3.1.1-1	OPEN ITEM	The applicant should clarify Section 1.1 of the RAP.	RPEB
17.3.1.2-1	OPEN ITEM	Section 1.2 of the RAP should define the scope and objective of a RAP, state basic the definitions and discuss the selection criteria.	RPEB
17.3.2-1	OPEN ITEM	The control of PRA design assumptions for the RAP should be clarified by the applicant. Also the applicant develop a method to identify and prioritize risk-significant SSCs.	RPEB
17.3.3-1	OPEN ITEM	Maximizing plant availability may conflict with maintaining acceptable PRA risk levels. This potential conflict must be addressed by the applicant.	RPEB

NUMBER	TYPE	TITLE	BRANCH
17.3.3.1-1	OPEN ITEM	The staff needs clarification on the intent of RAMI and needs the priority of safety requirements to be explicitly stated.	RPEB
17.3.3.2-1	OPEN ITEM	Will Nuclear Plant Reliability Data System (NPRDS) be used for the plant reliability data base? If not, how will applicant assist in establishing a data base?	RPEB
17.3.3.3-1	OPEN ITEM	The applicant must provide additional information on RAP as identified in DSER Section 17.4.3.3.	RPEB
17.3.4-1	OPEN ITEM	Clarification on the intent of making changes to the system or its reliability model in the evaluation phase of the RCM is needed.	RPEB
17.3.4-2	OPEN ITEM	It is not clear to the staff what will be included in the RCM program guide.	RPEB
17.3.5-1	OPEN ITEM	The applicant should clarify its intent regarding consistency between the PRA and plant procedures and Technical Specifications.	RPEB
17.3.6-1	OPEN ITEM	Clarification of the organizational accountability for implementing the design portion of the RAP is required.	RPEB
17.3.7-1	OPEN ITEM	The staff requests an example of how the RAP will function throughout plant life.	RPEB
17.3.7-2	OPEN ITEM	The applicant should provide a detailed discussion on how RAP differs from EPRI URD.	RPEB

CL ACTION ITEM 17.3.1.2-1

The development and implementation of the O-RAP is the responsibility of the referencing applicant, and the staff's position on the review of an O-RAP is that it will be evaluated as part of a referencing applicant's submittal for a CL.

RESPONSE CL 17.3.1.2-1

ABB-CE concurs with the above item. The development and implementation of the O-RAP is the responsibility of the referencing applicant.

Revised

OPEN ITEM 17.3.1.1-1

The applicant should clarify Section 1.1 of the RAP.

RESPONSE 17.3.1.1-1

The D-RAP as attached will be included in the CESSAR-DC as Section 17.3. Since the November 15 submittal, Section 17.3.1 has been rewritten to reflect comments from NRC, INPO, EPRI, and others. A copy of the completely revised D-RAP is attached.

Revised

OPEN ITEM 17.3.1.2-1

Section 1.2 of the RAP should define the scope and objective of a RAP, state the basic definitions and discuss the selection criteria.

RESPONSE 17.3.1.2-1

See Response 17.3.1.1-1

OPEN ITEM 17.3.2-1

The control of PRA design assumptions for the RAP should be clarified by the applicant. Also, the applicant should develop a method to identify and prioritized risk-significant SSCs.

RESPONSE 17.3.2-1

Section 17.3.7 of the re ised report discusses control of design assumptions and Section 17.3.6 discusses prioritization of risk-significant SSCs.

OPEN ITEM 17.3.3-1

Maximizing plant availability may conflict with maintaining acceptable PRA risk levels. This potential conflict must be addressed by the applicant.

RESPONSE 17.3.3-1

The RAMI analysis has been removed from the revised D-RAP. The safety requirements produced by the PRA models will have a higher priority than those of the RAMI analysis. It is possible that during the RAMI and PRA models development, the PRA and RAMI objectives may be conflicting (i.e., some of the means of maximizing plant availability may be in conflict with the objective of maintaining the risk levels assumed in the PRA). During this process, the reliability and design engineers will be looking for a balance between both the PRA and RAMI results to produce a safe design from a PRA point of view as well as the most economical (determined by RAMI analysis) <u>without sacrificing safety</u>. This will be ensured by comparing the PRA and RAMI results of several proposed designs for a particular system, and selecting the most cost effective design which meets the safety goal. In all cases, the safety goal will have a higher priority than the RAMI results. This item will be a usual topic for consideration during the Design Review Meetings between reliability and design engineers.

OPEN ITEM 17.3.3.1-1

The staff needs clarification on the intent of RAMI and needs the priority of safety requirements to be explicitly stated.

RESPONSE 17.3.3.1-1

See the response to 17.3.3-1

OPEN ITEM 17.3.3.2-1

Will nuclear plant reliability data system (NPRDS) be used for the plant reliability data base? If not, how will applicant assist in establishing a data base?

RESPONSE 17.3.3.2-1

The D-RAP no longer specifies which data base the designer or CL applicant should use.

OPEN ITEM 17.3.3.3-1

The applicant must provide additional information on RAP, as identified in DSER Section 17.4.3.3.

RESPONSE 17.3.3.3-1

The revised D-RAP provides additional information on RAP, as identified in DSER Section 17.4.3.3.

OPEN ITEM 17.3.4-1

Clarification on the intent of making changes to the system or its reliability model in the evaluation phase of the RCM is needed.

RESPONSE 17.3.4-1

The revised D-RAP no longer contains an RCM section but instead, lists various options that the CL applicant might use including RCM.

OPEN ITEM 17.3.4-2

It is not clear to the staff what will be included in the RCM program guide.

RESPONSE 17.3.4-2

See response 17.3.4-1.

OPEN ITEM 17.3.5-1

The applicant should clarify its intent regarding consistency between the PRA and plant procedures and Technical Specifications.

RESPONSE 17.3.5-1

The revised D-RAP no longer contains any wording regarding consistency between the PRA and plant procedures and Technical Specifications. Such discussions are the responsibility of the CL applicant.

OPEN ITEM 17.3.6-1

Clarification of the organizational accountability for implementing the design portion of the RAP is required.

RESPONSE 17.3.6-1

Section 17.3.5 of the revised D-RAP describes the organization.

OPEN ITEM 17.3.7-1

The staff requests an example of how the RAP will function throughout plant life.

RESPONSE 17.3.7-1

Section 17.3.11 of the revised D-RAP contains an example.

OPEN ITEM 17.3.7-2

The applicant should provide a detailed discussion on how RAP differs from EPRI URD.

RESPONSE 17.3.7-2

EPRI has reviewed our D-RAP and their comments have been incorporated into the attached D-RAP. For the most part, the D-RAP presented here follows the guidelines established by the ALWR URD. Some minor differences with the current URD are discussed below.

RAMI Analysis
The URD suggested including the RAMI program as part of the D-RAP. We did this in the first draft but now believe that the two programs should be separate.

NPRDS
The URD guide recommended the use of NPRDS in the RAP. The revised D-RAP no longer recommends a specific data base but deals more with the procedures and organization. The primary source of data used for the PRA in the Preliminary Design Phase is the "PRA Key Assumptions and Groundrules" (KAG) document (Appendix A to Chapter 1 of the ALWR Requirements Document). This source of data is considered to be very consistent and robust. Other industry-accepted generic data sources will be used as needed to supplement the data in the KAG as the plant design moves to the construction phase. Initially the Nuclear Plant Reliability Data System (NPRDS) maintained by INPO will only be used for data used in the RAMI analyses (i.e., mean time to repair data). The CL applicant may then update this database with plant specific data as well as other industry-accepted source of information.

The URD guide has recommended the use of NPRDS for the design phase of the RAP. We have elected not to specify what data base is to be used in the D-RAP.

Performance Standards
The URD guideline provides not only a safety goal (core damage frequency) of less than or equal to 10^{-5}/R-Y, but also several availability goals such as inadvertent RCS depressurization, station blackout, trip frequency, production availability, and plant outages (forced, planned, major, and refueling). The D-RAP program should only address the safety goal throughout the design phase. The availability standards only provide an economic benefit to the design and operation of the plant, and therefore, should not be included as part of the D-RAP plan. RAMI modeling is not included in this D-RAP since it addresses optimization of the plant design from an economic point of view. See response 17.3.3-1 for a discussion of the potential conflict between D-RAP and RAMI.