Westinghouse Electric Corporation
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NSD-NRC-97-5064
DCP/NRC0810
Docket No.: STN-52-003

April 11, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: T. R. QUAY

SUBJECT: AP600 DESIGN CERTIFICATION; FORMAL NOTIFICATION OF RESOLUTION OF ITEMS ASSOCIATED WITH SECTION 3.8.2, STEEL CONTAINMENT

References: SECY-97-051, "Schedule for the Staff's Review of the AP600 Design Certification Application," dated February 26, 1997, forwarded by NRC letter "Westinghouse's Support of the Nuclear Regulatory Commission Review of the AP600 Design Certification Review," dated March 6, 1997.

Dear Mr. Quay:

This letter is to formally consolidate responses and resolutions of items associated with SSAR Section 3.8.2 and to confirm completion of submittal of final documentation related to SSAR Section 3.8.2 for our application for AP600 Design Certification. The Reference includes a milestone "Applicant Submits Final SSAR Revision & Documentation" by May 1997. Westinghouse interprets this to require NRC acknowledgement of receipt of final documentation supporting our application for AP600 Design Certification. To support this milestone, NRC and Westinghouse maintain a detailed activity plan which provides schedule goals for most SSAR/FSER sections and related activities, such as, the PRA, code validation, and ITAACs. In this detail activity plan, Westinghouse application input and NRC internal FSER input for Section 3.8.2 of the SSAR has a schedule goal of March 31, 1997. NRC and Westinghouse also maintains a joint open item tracking system to informally monitor the status and history of open items (DSER, RAI, meeting, and other) associated with our application.

NRC has requested that, although most items have been discussed and resolved using SSAR and RAI markups followed by formal revisions, Westinghouse consolidate their remaining resolutions into a single, formal response. Attachment 1 to this letter provides a chronology for each item discussed. Westinghouse believes it has submitted resolution for all items for SSAR Section 3.8.2. Attachment 2 provides formalized copies resubmitting the resolving documentation for items not acknowledged by NRC. Note that some responses were provided 2 months ago. NRC is requested to acknowledge receipt of this information by directing Westinghouse to change the "NRC Status" to "Action N" or "Resolved".

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NSD-NRC-96-4904
DCP/NRC0674
Docket No.: STN-52-003

December 9, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: AP600 RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION, AND AP600 PRA PAGE MARKUPS

Dear Mr. Quay:

Enclosure 1 provides Westinghouse responses to NRC requests for additional information pertaining to the conditional containment failure probability distribution used in Chapter 42 of the AP600 Probabilistic Risk Assessment (PRA). Specifically, the responses to the following RAIs are included: 480.190 through 480.192, 220.95 through 220.99. These responses close, from a Westinghouse perspective, the addressed questions. The NRC technical staff should review these responses. The status of these RAIs will be changed to "Action N" in the OITS on January 2, 1997.

Enclosure 2 provides the draft responses to NRC requests for additional information pertaining to the AP600 at-power fire PRA. These responses are being provided to the NRC for discussion purposes at the December 18, 1996 NRC and Westinghouse AP600 fire PRA meeting. The staff is expected to review these draft responses and be prepared to discuss them at the meeting.

Enclosure 3 contains markup page changes to the AP600 PRA. These markups primarily fix typos found in the report since the issuance of Revision 8. These changes will be included in the next revision to the PRA. The staff reviewers should include these markup pages with their PRA report.

A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

Please contact Cynthia L. Haag on (412) 374-4277 if you have any questions concerning this transmittal.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

3013A

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Enclosures

cc: J. Sebrosky, NRC (enclosures)
J. Kudrick, NRC (w/o enclosures)
J. Flack, NRC (w/o enclosures)
N. J. Liparulo, Westinghouse (w/o enclosures)

3013A

Question: 480.191 Containment Pressure Capacity

Provide an assessment of the pressure capability of the main steam line and main feedline bellows, and a corresponding failure probability distribution curve.

Response:

The pressure capability of the main steam line and main feedline bellows are discussed in SSAR subsection 3.8.2.4.5. The pressure capability exceeds the pressure at which the containment vessel cylinder yields. It is assumed that once general yielding of the cylinder occurs, the bellows may fail due to the large deflection of the cylinder. However, it is not necessary to provide the probability of failure for bellows separate from the probability of cylinder yield. The probability of failure of the bellows due to large deformation of the cylinder is a part of the probability given in PRA Chapter 42 for vessel failure due to the general yielding of the cylinder.

PRA Revision: None.

AP600 Open Item Tracking System Database: Executive Summary

Date: 4/3/97

Selection: [nrc st code]='Action W' And [DSER Section] like '3.8.2*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
706	NRR/ECGB	3.8.2.4-28	DSER-OI		Orr	Closed	Action W	NTD-NRC-95-4464	
708	NRR/ECGB	3.8.2.4-30	DSER-OI		Orr/CBI	Closed	Action W		
1888	NRR/ECGB	3.8.2.4-1	DSER-COL		Orr	Closed	Action W	NSD-NRC-97-4981	

706

Westinghouse should provide in the SSAR an assessment of the pressure capability of the main steamline and main feedwater line bellows, a corresponding failure probability distribution curve, and the impact on the overall cumulative failure probability curve.

The effect of containment pressure on the bellows was addressed in the response to RAI 720.206. This response has been incorporated in SSAR Subsection 3.8.2.4.2.6, Rev 3. The bellows remain intact when the containment shell remains elastic and imposed deflections remain close to the design conditions. Failure of the bellows is assumed to occur once the containment cylinder yields. This mechanism is already included in the failure probability curve for the cylinder.

Discussed in meeting at CBI 8/30 - 31/95. Additional discussion was provided in draft SSAR or PRA report that bellows failure mode and fragility is included in the containment vessel cylinder yield failure mode.
Closed: further review is under new RAI 220.99 transmitted by NRC letter of April 4, 1996

NRC Status Update provided in September 5, 1996 letter:
This staff does not agree that this item is closed or resolved due to RAI# 220.99. Both this open item and OI# 3268 (RAI #220.99) should be tracked individually to resolution. Action Westinghouse
Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.

708

Westinghouse should increase the thickness or use stiffeners (as in the ABB-CE System 80+ design) to meet the ASME Service Level C limits at the ambient temperature of 908 kPa (117 psig) for a 6.7 m (22-ft) diameter hatch, and 763.2 kPa (96 psig) for a 4.9-m (16- ft) diameter hatch.

ASME have confirmed that the method used for the AP600 complies with ASME Code Case N 284.
Westinghouse position is that use of code case N284 satisfies the deterministic Service Level C criteria approved by the commissioners.
NRC staff will review N284, Revision 1 and the ASME confirmation of the AP600 interpretation.
Closed: further review is under new RAI 220.100 transmitted by NRC letter of April 4, 1996
NRC Status Update provided in September 5, 1996 letter:
This staff does not agree that this item is closed or resolved due to AI# 220.100. Both this open item and OI 3269 (RAI #220.100) should be tracked individually to resolution.
Response to OI# 3269 transmitted to NRC on 2/11/97, NSD-NRC-97-4981 also applies to this open item.

1888

3.8.2.4-1 The COL applicant should demonstrate that EPAs to be used shall be at least as strong as the AP600 SCV.

Discussed in meeting at CBI 8/30 - 31/95. Expand COL information to include demonstration that EPA satisfies Service Level C pressure and temperature requirement. Revise SSAR 3.8.6.1 to change "ultimate capacities" to "ultimate pressure capacities"
Closed: additional clarification is requested under RAI 220.102 transmitted by NRC letter dated April 4, 1996.
Closed - Response provided by NSD-NRC-97-4981 of 2/11/97.

Attachment 1 to NSD-NRC-97-5064

Chronology for Open Items Associated with Section 3.8.2

Open Item Number	NRC Status	Response Vehicle	Response Date	Appendix 2 Page
706	Action W	NSD-NRC-97-4981 NSD-NRC-96-4904	2/11/97 12/9/96	1, 3 & 10
708	Action W	NSD-NRC-97-4981	2/11/97	1 & 3-7
1888	Action W	NSD-NRC-97-4981	2/11/97	1 & 3

Westinghouse Electric Corporation
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NSD-NRC-97-4981
DCP/NRC0737
Docket No.: STN-52-003

February 11, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: T. R. QUAY

SUBJECT: AP600 CONTAINMENT OPEN ITEMS

Dear Mr. Quay:

Attached are responses to resolve DSER open items, requests for additional information, and a telecon item related to AP600 containment structural issues. The responses include draft SSAR changes that will be included in Revision 11 of the AP600 SSAR. The list below includes the DSER item number or RAI number and the open item tracking system (OITS) number.

DSER OI 3.8.2.4-3 (681)
DSER OI 3.8.2.4-20 (698)
DSER OI 3.8.2.4-28 (706)
DSER COL 3.8.2.4-1 (1888)
Telecon Item June 23, 1995 (2515)
RAI 220.100 (3269)
RAI 220.101 (3270)
RAI 220.102 (3271)

These responses provide a way to resolve these items and will permit the NRC staff to provide input for the final safety evaluation report.

If you have any questions please contact Donald A. Lindgren at (412) 374-4856.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/jml

Attachment

cc: D. Jackson, NRC
T. Cheng, NRC

Open Item #706 - DSER Open Item # 3.8.2.4-28

Westinghouse should provide in the SSAR an assessment of the pressure capability of the main steamline and main feedwater line bellows, a corresponding failure probability distribution curve, and the impact on the overall cumulative failure probability curve.

Westinghouse response

This is the same question as RAI 480.191. See response to RAI 480.191 provided in letter NSD NRC-96-4904, dated 12/9/96.

Open Item #1888 - DSER Open Item # 3.8.2.4-1

The COL applicant should demonstrate that EPAs to be used shall be at least as strong as the AP600 SCV.

Westinghouse response

Additional information has been included in SSAR subsection 3.8.2.4.2.5, Revision 7. See also response to RAI 220.102 (OI# 3271).

Open Item #2515 - Telecon June 23, 1995

Westinghouse should address the issue of fatigue and corrosion of containment bellows. The number of thermal cycles and loading information included in the design specification should be addressed. The material requirements and effect of corrosion should also be included.

Westinghouse response

SSAR section 3.8.2.1.5, Rev 7 has been revised to include material and additional information on the displacement cycles. Fatigue is evaluated in accordance with ASME subsection NE as stated in SSAR subsection 3.8.2.1.5. Bellows materials are stainless steel or nickel alloy. Corrosion is not expected; if there is any degradation it would be observed by inservice inspection or testing. The bellows are included in the ISI of the containment vessel as well as the containment leak rate testing.

DSER Open Item # 3269 (NRC letter dated 4/4/96) RAI # 220.100

In SSAR Section 3.8.2.4.2.3, the factor of safety (FS) of 1.67 is used for equipment hatch covers ASME Service Level C limits.

Westinghouse estimated the critical buckling pressures for equipment hatches as 1.45 MPa (196 psig) for a 6.7 m (22 ft) diameter hatch and 1.21 MPa (161 psig) for a 4.9 m (16 ft) diameter hatch based on the classical buckling capacity of spherical shells subjected to external pressure and the capacity reduction factors specified in Baker et al., "Structural Analysis of Shells,"pp. 253-254, McGraw-Hill, 1972, and in ASME Code Case N-284. The corresponding ASME Service Level C limits are 908 kPa (117 psig) and 763.2 kPa (96 psig) using the factor of safety (FS) of 1.67 as specified in Code Case N-284, respectively.

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For the FS to be applied to the Service Level C pressure capacity, Westinghouse considered the equipment hatch cover buckling due to external pressure as the local buckling (FS = 1.67 from Code Case N-284). The hatch cover is a complete shell by itself with its own independent boundary and is subjected to pressure on its convex side due to the containment internal pressure. Therefore, the staff position is that the global buckling (FS = 2.5 from NE-3222) is the appropriate value. The ASME Service Level C pressure capacity is 763.2 kPa (96 psig) with FS of 1.67 and 545.4 kPa (64.4 psig) with an FS of 2.5.

Based on Code Case N-284, the local buckling is defined as the buckling of the shell plate between stiffeners. The flange of the cover can act as a stiffening element around the periphery of the spherical cap. However, the stiffening effect is limited to $(Rt)^{1/2}$ or 35.3 cm (13.9 in) from the edge. The entire arc length from the center of the hatch cover to the flange is 255.3 cm (100.5 in). The remaining 218.4 cm (86 in) arc should be considered as unstiffened, therefore, the global buckling criteria should be applied to this unstiffened region. In the draft safety evaluation report (DSER), the staff noted that Westinghouse's assumption of local buckling for the equipment hatch cover under external pressure was not acceptable. The staff requested that Westinghouse increase the thickness or use stiffeners (e.g., ABB-CE System 80+ design) to meet the ASME Service Level C limits at the ambient temperature of 908 kPa (117 psig) for a 6.7 m (22 ft) diameter hatch and 763.2 kPa (96 psig) for a 4.9 m (16 ft) diameter hatch. This was Open Item 3.8.2.4-30.

The staff performed independent analysis for the equipment hatch covers using the ALGOR computer code with fixed boundary conditions and no imperfection. Using ALGOR, the staff predicted the buckling pressure, Pbuckling, as 1.38 MPa (185.12 psig) and 1.57 MPa (212.96 psig) for 4.9 m (16 ft) and 6.7 m (22 ft) equipment hatch covers, respectively. In both cases, the buckling was predicted to occur near the top portion.

For the reasons discussed above, the staff considers the equipment hatch covers buckling as a global failure mode. There is a potential for radioactive gas leakage through the equipment hatch sleeve/gasket once buckling occurs. Thus, the leaktight integrity of the containment is jeopardized. On this basis, the staff finds that a higher FS of 2.5 based on NE-3222 should be applied.

Westinghouse Response

The SSAR will be revised to show capacities using the factor of safety of 2.5. Information will be retained with a factor of safety of 1.67 using Code Case N 284, the use of which has been endorsed by ASME (see attached letter dated 2/6/96). At the design temperature of 280° F, the capacity is 62 psig with a safety factor of 2.5, and 90 psig with a safety factor of 1.67. The PRA report shows that none of the more likely severe accident sequences exceed either of these pressures.

SSAR changes

3.8.2.4.2 Evaluation of Ultimate Capacity

The capacity of the containment vessel has been calculated for internal pressure loads for use in the probabilistic risk assessment analyses and severe accident evaluations. Each element of the containment vessel boundary was evaluated to estimate the maximum pressure at an ambient temperature of 100°F corresponding to the following stress and buckling criteria:

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- Deterministic severe accident pressure capacity corresponding to ASME Service Level C limits on stress intensity, ASME paragraph NE-3222 and ASME Code Case N-284 for buckling of the equipment hatch covers, and 60 percent of critical buckling for the top head. The deterministic severe accident pressure capacity corresponds to the approach in SECY 93-087, to maintain a reliable leak-tight barrier approximately 24 hours following the onset of core damage under the more likely severe accident challenges. This approach was approved by the Nuclear Regulatory Commission as outline in the Staff Requirements Memorandum on SECY-93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs, Dated July 21, 1993.

- Best estimate capacity corresponding to gross membrane yield at the ASME-specified minimum yield stress (SA537, Class 2, yield stress = 60 ksi, ultimate stress = 80 ksi), and critical buckling for the equipment hatch covers and top head.

The results are shown in Table 3.8.2-2. The analyses at a temperature of 100°F are described in the following paragraphs for each element. The critical regions identified in this table are then examined further for their response at higher temperatures. This results in the best-estimate capacity based on the ASME-specified minimum yield properties. The evaluation considered the containment boundary elements including:

- Cylindrical shell
- Top and bottom heads
- Equipment hatches and covers
- Personnel airlocks
- Mechanical and electrical penetrations

The evaluation identified the most likely failure mode to be that associated with gross yield of the cylindrical shell. Loss of containment function would be expected to occur because the large post-yield deflections would lead to local failures at penetrations, bellows, or other local discontinuities.

3.8.2.4.2.3 Equipment Hatches

SECY 93-087 permits evaluation of certain severe accident scenarios against ASME Service Level C limits. The equipment hatch covers were evaluated for buckling against ASME paragraph NE-3222 and according to ASME Code, Case N-284. Use of ASME Code, Case N-284 for this application was confirmed to be appropriate by ASME. The containment internal pressure acts on the convex face of the dished head and the hatch covers are in compression under containment internal pressure loads. The critical buckling capacity is based on classical buckling capacities reduced by capacity reduction factors to account for the effects of imperfections and plasticity. These capacity reduction factors are based on test data and are generally lower-bound values for the tolerances specified in the ASME Code.

The critical buckling pressures are 195 psig for the 22-foot-diameter hatch and 160 psig for the 16-foot-diameter hatch at an ambient temperature of 100°F. For the Service Level C limits in accordance with paragraph NE 3222, a safety factor of 2.50 is specified, resulting in capabilities of 78 psig (22-foot-diameter) and 64 psig (16-foot-diameter). For the Service Level

C limits in accordance with Code Case N284, a safety factor of 1.67 is specified, resulting in capabilities of 117 psig (22-foot-diameter) and 96 psig (16-foot-diameter).

Typical gaskets have been tested for severe accident conditions as described in NUREG/CR-5096 (Reference 25). The gaskets for the AP600 will be similar to those tested with material such as Presray EPDM E 603. For such gaskets the onset of leakage occurred at a temperature of about 600°F.

3.8.2.4.2.8 Summary of Containment Pressure Capacity

The ultimate pressure capacity for containment function is expected to be associated with leakage caused by excessive radial deflection of the containment cylindrical shell. This radial deflection causes distress to the mechanical penetrations, and leakage would be expected at the expansion bellows for the main steam and feedwater piping. There is high confidence that this failure would not occur before stresses in the shell reach the minimum specified material yield. This is calculated to occur at a pressure of 144 psig at ambient temperature and 120 psig at 400°F. Failure would be more likely to occur at a pressure about 15 percent higher based on expected actual material properties.

The deterministic severe accident pressure that can be accommodated according to the ASME Service Level C stress intensity limits and using a factor of safety of 1.67 for buckling of the top head is determined by the capacity of the 16-foot-diameter equipment hatch cover and the ellipsoidal head. The maximum capacity of the hatch cover, calculated according to ASME paragraph NE-3222, Service Level C, is 64 psig at an ambient temperature of 100°F and 62 psig at 280°F. When calculated in accordance with ASME Code, Case N-284, Service Level C, the maximum capacity is 96 psig at an ambient temperature of 100°F and 93 psig at 280°F. The maximum capacity of the ellipsoidal head is 104 psig at 100°F and 92 psig at 280°F.

The maximum pressure that can be accommodated according to the ASME Service Level C stress intensity limits, excluding evaluation of instability, is determined by yield of the cylinder and is 125 psig at an ambient temperature of 100°F and 110 psig at 280°F. This limit is used in the evaluations required by 10 CFR 50.34(f).

Table 3.8.2-2

CONTAINMENT VESSEL PRESSURE CAPABILITIES

Containment Element		**Pressure Capability**				
		Deterministic Severe Accident Capacity[1]			**Maximum Pressure Capability**[2]	
Temperature		100°F	280°F	400°F	100°F	400°F
Cylinder		125 psig	110 psig	104 psig	144 psig	120 psig
Ellipsoidal Head		104 psig	92 psig	87 psig	174 psig	145 psig
22-foot equipment hatch	F.S = 1.67	117 psig	114 psig	110 psig	195 psig	184 psig
	F.S = 2.50	78 psig	76 psig	73 psig		
16-foot equipment hatch	F.S = 1.67	96 psig	93 psig	90 psig	160 psig	151 psig
	F.S = 2.50	64 psig	62 psig	60 psig		
Personnel airlocks[3]		> 163 psig	> 163 psig	> 163 psig	> 300 psig	> 300 psig

<u>Notes</u>:

1. The buckling capacity of the ellipsoidal head is taken as 60 percent of the critical buckling pressure calculated by the BOSOR-5 nonlinear analyses; the buckling capacity at higher temperatures is calculated by reducing the capacity at 100°F by the ratio of yield at 100°F to yield at the higher temperature. Evaluations of the equipment hatch covers are shown both for ASME paragraph NE-3222 (F.S. = 2.50) and Code Case N-284 (F.S. = 1.67). Evaluations of the other elements are according to ASME Service Level C~~ and include use of Code Case N284~~.

2. The estimated maximum pressure capability is based on minimum specified material properties.

3. The capacities of the personnel airlocks are estimated from test results.

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