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Your ref: Docket No. 52-006
Our ref: DCP/NRC1603

July 8, 2003

SUBJECT: Transmittal of Westinghouse Responses to Open Items Identified in the AP1000 Draft Safety Evaluation Report

This letter transmits Westinghouse responses to open items identified in the AP1000 Draft Safety Evaluation Report (DSER) that was issued on June 16, 2003. A list of the DSER Open Item responses that are transmitted with this letter is provided in Attachment 1. Attachment 2 provides the DSER Open Item responses. Attachment 3 transmits a report referenced in the response to DSER OI 9.2.3.3-1, "Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility."

Please contact me if you have questions regarding this transmittal.

Very truly yours,

M. M. Corletti

M. M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects

/Attachments

1. Table 1, "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1603"

2. Westinghouse Non-Proprietary Responses to US Nuclear Regulatory Commission DSER Open Items dated July 8, 2003

3. "Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility," University of California Santa Barbara, CRSS-03/06, dated June 30, 2003

D063

DCP/NRC1603
Docket 52-006

July 8, 2003

Attachment 1

"List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1603"

Attachment 1

Table 1 "List of Westinghouse's Responses to DSER Open Items Transmitted in DCP/NRC1603"	
3.8.2.1-1, Rev. 1	
14.3.2-15	
16.2-3	
19.2.6-1 19.2.6-2 19.2.3.3-1 19.3.10-1	

Attachment 2

Westinghouse Non-Proprietary Response to
AP1000 Draft Safety Evaluation Report (DSER) Open Items

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **3.8.2.1-1 R1**

Original RAI Number(s): None (April 3, 2003, meeting summary)

Summary of Issue:

The containment vessel is an ASME metal containment. The information contained in this subsection is based on the design specification and preliminary design and analyses of the vessel. During the April 2-5, 2003 audit at Westinghouse, the applicant informed the staff that the final detailed analyses, to be documented in the ASME Design Report, are not available and will be the responsibility of the COL applicant. The staff expected that the final detailed analyses for the AP1000 steel containment would be submitted for staff review as part of the design certification process for AP1000. To complete the staff evaluation of the AP1000 steel containment design, the staff will need to audit the final detailed analyses. This is Open Item 3.8.2.1-1.

Westinghouse Response:

Westinghouse identified additional detailed analyses to be performed for the containment vessel in letter DCP/NRC1583, dated May 1, 2003. These analyses are available for NRC staff review and demonstrate that the AP1000 containment vessel satisfies the acceptance criteria documented in the DCD.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 14.3.2-15

Original RAI Number(s): None

Summary of Issue:

Section 3.7, "Design Reliability Assurance Program" (D-RAP). The staff found that the list of risk significant components in Table 3.7-1 was not updated to include all risk-significant structures, systems, and components (SSCs) from the list of risk significant SSCs identified in Tier 2 Section 17.4, Table 17.4-1, "Risk Significant SSCs within the Scope of D-RAP." Specifically, the list of risk significant components should include:

- Compressed and Instrument Air System Air Compressor Transmitter
- Passive Containment Cooling System Diverse (3rd) Motor Operated Drain Isolation Valve function
- In-containment Refueling Water Storage Tank Vents
- Normal Residual Heat Removal Valve V055 function
- Feedwater Isolation Valves

As discussed in Section 17.4 of this report, the staff determined that Table 17.4-1 contained an acceptable list of risk significant SSCs under the scope of D-RAP. In Table 17.4-1, the applicant also removed the safety related passive core cooling condensate sump re-circulation valves' automatic open function from the D-RAP for the AP1000 design and this should be reflected in ITAAC Table 3.7-1. This is Open Item 14.3.2-15.

Westinghouse Response:

We have performed a review of the DCD D-RAP Table 17.4-1 and ITAAC Table 3.7-1. Based on that review we have the following comments:

1. The PRA importance of the Compressed and Instrument Air System, Air Compressor Pressure Transmitter has been re-evaluated. Based on the current AP1000 PRA this instrument just meets the DRAP selection criteria (RAW, RRW) for LRF although it does not meet the DRAP selection criteria for CDF. Furthermore, it has been determined that there are conservatisms in the PRA that have resulted in the RAW / RRW values for this instrument being over estimated. These conservatisms are due to not modeling some plant features that would have reduced the PRA importance of this instrument. Based on this re-evaluation, this instrument should no longer be listed in the DRAP tables in the DCD or the ITAAC. Therefore it has been removed from DCD Table 17.4-1 and it has not been added to ITAAC Table 3.7-1.

2. We agree that that the following items should be added to ITAAC Table 3.7-1:
   - IRWST vents
   - Main Feedwater Isolation Valves

3. The 3rd PCS water drain valve does not have to be added to ITAAC Table 3.7-1 because it is already listed the table. Under item PCCWST Drain Isolation Valves are listed 3 valves, PCS-PL-V001A/B/C. The C valve is the diverse (3rd) drain valve.

4. We agree that RNS valve 055 should be added. However, as indicated in DCD Table 17.4-1, other RNS MOVs are also required to allow the RNS to provide RCS makeup following ADS actuation, including:
   - V011 RNS discharge containment isolation
   - V022 RNS suction containment isolation
   - V055 RNS suction from the SFS Cask Loading Pit
   - V062 RNS suction from the IRWST

5. We agree that the PXS containment recirculation MOVs (PXS-PL-V117A/B) should be removed from ITAAC Table 3.7-1, since they have been removed from DCD Table 17.4-1.

6. Our review also indicates that the following additional changes should be made to ITAAC Table 3.7-1.
   - Add CVS Makeup Pump suction and discharge check valves
   - Add inverters and battery chargers for the 24 hour batteries
   - Add reactor vessel insulation water inlet and steam vent devices
   - Add reactor cavity doorway damper
   - Add service water cooling tower fans
   - Add low capacity chilled water subsystem
   - Add standby diesel generator room cooling fans
   - Add fuel assemblies
   - Remove PXS valves PXS-PL-V125A/B from the IRWST injection squib valve group since these valves are not squibs and –V123A/B and –V125A/B lists the four squibs in these lines

Design Control Document (DCD) Revision:

DCD Table 17.4-1 will be changed as shown below:

DCD Table 17.4-1 (Sheet 1 of 10)

RISK-SIGNIFICANT SSCS WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
~~System: Compressed and Instrument Air System (CAS)~~		
~~Air Compressor Transmitter~~	~~RRW/LRF~~	~~Failure of air compressor transmitter.~~
System: Component Cooling Water (CCS)		
CCS Pumps	EP	These pumps provide cooling of the normal residual heat removal system (RNS) and the spent fuel pool heat exchanger. Cooling the RNS heat exchanger is important to investment protection during shutdown reduced-inventory conditions. CCS valve realignment is not required for reduced-inventory conditions.
System: Containment System (CNS)		
Containment Vessel	EP, L2	The containment vessel provides a barrier to steam and radioactivity released to the atmosphere following accidents.
Hydrogen Igniters	EP, L2, Regulations	The hydrogen igniters provide a means to control H_2 concentration in the containment atmosphere, consistent with the hydrogen control requirements of 10 CFR 50.34f.

ITAAC Table 3.7-1 will be changed as shown below:

Table 3.7-1
Risk-Significant Components

Equipment Name	Tag No.
PMS Actuation Software (used to provide automatic control functions listed in Tables 2.5.2-2 and 2.5.2-3)	–
PMS Actuation Hardware (used to provide automatic control functions listed in Tables 2.5.2-2 and 2.5.2-3)	–
MCR 1E Displays	OCS-JC-010, OCS-JC-011
MCR 1E System Level Controls	OCS-JC-010, OCS-JC-011
Reactor Trip Switch Gear	PMS-JP-RTS A01/2 PMS-JP-RTS B01/2 PMS-JP-RTS C01/2 PMS-JP-RTS D01/2
Reactor Coolant Pump Circuit Breakers	ECS-ES-31, -32, -41, -42 ECS-ES-51, -52, -61, -62
Annex Building UPS Distribution Panels (provide power to DAS)	EDS1-EA-14 EDS2-EA-14
PLS Actuation Software and Hardware (used to provide automatic control functions listed in Table 3.7-2)	–
DAS Actuation Hardware (used to provide automatic and manual actuation)	DAS-JD-001 DAS-JD-002 OCS-JC-020
Containment Isolation Valves Controlled by DAS	Refer to Table 2.2.1-1

Note: Dash (-) indicates not applicable.

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Control Rod MG Set Field Breakers	PLS-MG-01A, PLS-MG-01B
Makeup Pumps	CVS-MP-01A, -01B
Makeup Pump Suction and Discharge Check Valves	**CVS-PL-V113 CVS-PL-V160A, -V160B**
RNS Pumps	RNS-MP-01A, -01B
RNS MOVs	**RNS-PL-V011, -V022, -V055, -V062**
Startup Feedwater Pumps	FWS-MP-03A, -03B
SFS Pumps	SFS-MP-01A, -01B
CCS Pumps	CCS-MP-01A, -01B
Service Water Pumps	SWS-MP-01A, -01B
Service Water Cooling Tower Fans	**MA-01A, -01B**
PCCWST Recirculation Pumps	PCS-MP-01A, -01B
PCCWST Drain Isolation Valves	PCS-PL-V001A/B/C
Standby Diesel Generators	ZOS-MG-02A, -02B
Ancillary Diesel Generators	ECS-MG-01, -02
MCR Ancillary Fans	VBS-MA-10A, -10B
I&C Room B/C Ancillary Fans	VBS-MA-11, -12
Air Cooled Chiller Package	**VWS-MS-02, -03**
Air Cooled Chiller Pumps	**VWS-MP-02, -03**
Standby Diesel Generator Room Cooling Fan	**VZS-MA-03A, -03B**
Hydrogen Ignitors	VLS-EH-1 through -60
Containment Vessel	CNS-MV-50
Reactor Vessel Insulation Water Inlet and Steam Vent Devices	**RCS-MN-01**
Reactor Cavity Doorway Damper	-
Pressurizer Safety Valves	RCS-PL-V005A RCS-PL-V005B

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
First-Stage ADS MOV	RCS-PL-V001A RCS-PL-V001B RCS-PL-V011A RCS-PL-V011B
Second-Stage ADS MOV	RCS-PL-V002A RCS-PL-V002B RCS-PL-V012A RCS-PL-V012B
Third-Stage ADS MOV	RCS-PL-V003A RCS-PL-V003B RCS-PL-V013A RCS-PL-V013B
Fourth-Stage ADS Squib Valves	RCS-PL-V004A RCS-PL-V004B RCS-PL-V004C RCS-PL-V004D
RCS Hot Leg Level Sensors	RCS-160A RCS-160B
Pressurizer Pressure Sensors	RCS-191A RCS-191B RCS-191C RCS-191D
Pressurizer Level Sensors	RCS-195A RCS-195B RCS-195C RCS-195D
Main Steam Line Isolation Valves	SGS-PL-V040A SGS-PL-V040B
Main Feedwater Isolation Valves	**SGS-PL-V057A** **SGS-PL-V057B**
Steam Generator Narrow-Range Level Sensors	SGS-001 SGS-002 SGS-003 SGS-004 SGS-005 SGS-006 SGS-007 SGS-008

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
Steam Generator Wide-Range Level Sensors	SGS-011 SGS-012 SGS-013 SGS-014 SGS-015 SGS-016 SGS-017 SGS-018
Steam Line Pressure Sensors	SGS-030 SGS-031 SGS-032 SGS-033 SGS-034 SGS-035 SGS-036 SGS-037
Main Steam Safety Valves	SGS-PL-V030A SGS-PL-V030B SGS-PL-V031A SGS-PL-V031B SGS-PL-V032A SGS-PL-V032B SGS-PL-V033A SGS-PL-V033B SGS-PL-V034A SGS-PL-V034B SGS-PL-V035A SGS-PL-V035B
IRWST Screens	PXS-MY-Y01A PXS-MY-Y01B
Containment Recirculation Screens	PXS-MY-Y02A PXS-MY-Y02B
IRWST Vents	**PXS-MT-03**
CMT Discharge Isolation Valves	PXS-PL-V014A PXS-PL-V014B PXS-PL-V015A PXS-PL-V015B

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
CMT Discharge Check Valves	PXS-PL-V016A PXS-PL-V016B PXS-PL-V017A PXS-PL-V017B
IRWST Gutter Bypass Isolation Valves	PXS-PL-V130A PXS-PL-V130B
Accumulator Discharge Check Valves	PXS-PL-V028A PXS-PL-V028B PXS-PL-V029A PXS-PL-V029B
PRHR HX Control Valves	PXS-PL-V108A PXS-PL-V108B
~~Containment Recirculation Isolation Motor-operated Valves~~	~~PXS-PL-V117A~~ ~~PXS-PL-V117B~~
Containment Recirculation Squib Valves	PXS-PL-V118A PXS-PL-V118B PXS-PL-V120A PXS-PL-V120B
IRWST Injection Check Valves	PXS-PL-V122A PXS-PL-V122B

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
IRWST Injection Squib Valves	PXS-PL-V123A PXS-PL-V123B ~~PXS-PL-V124A~~ ~~PXS-PL-V124B~~ PXS-PL-V125A PXS-PL-V125B
CMT Level Sensors	PXS-011A PXS-011B PXS-011C PXS-011D PXS-012A PXS-012B PXS-012C PXS-012D PXS-013A PXS-013B PXS-013C PXS-013D PXS-014A PXS-014B PXS-014C PXS-014D
IRWST Level Sensors	PXS-045 PXS-046 PXS-047 PXS-048
125 Vdc 24-Hour Battery	IDSA-DB-1A IDSA-DB-1B IDSB-DB-1A IDSB-DB-1B IDSC-DB-1A IDSC-DB-1B IDSD-DB-1A IDSD-DB-1B

Table 3.7-1 (cont.)
Risk-Significant Components

Equipment Name	Tag No.
125Vdc Distribution Panels	IDSA-DD-1 IDSB-DD-1 IDSC-DD-1 IDSD-DD-1 IDSA-EA-1 IDSA-EA-2 IDSB-EA-1 IDSB-EA-2 IDSB-EA-3 IDSC-EA-1 IDSC-EA-2 IDSC-EA-3 IDSD-EA-1 IDSD-EA-2
125 Vdc 24-Hour Battery Charger	**IDSA-DC-1 IDSB-DC-1 IDSC-DC-1 IDSD-DC-1**
Inverter, 125 Vdc 24-Hour Battery	**IDSA-DU-1 IDSB-DU-1 IDSC-DU-1 IDSD-DU-1**
Fused Transfer Switch Box	IDSA-DF-1 IDSB-DF-1 IDSB-DF-2 IDSC-DF-1 IDSC-DF-2 IDSD-DF-1
125 Vdc MCC	IDSA-DK-1 IDSB-DK-1 IDSC-DK-1 IDSD-DK-1
Fuel Assembly	**RXS-FA-A04 through -N10**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **16.2-3**

Original RAI Number(s): **630.052**

Summary of Issue:

The time interval between the time the reactor was last critical and the initial movement of an irradiated fuel assembly from the reactor core is a key assumption in AP1000 design basis fuel handling accident analysis dose consequence estimates, and spent fuel pool cooling requirements. As such, this decay time satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii), and is required to be included in an LCO in AP1000 TS, preferably in TS Section 3.9. Westinghouse did not propose a decay time specification in the AP1000 TS. This is Open Item 16.2-3.

Westinghouse Response:

A new AP1000 TS 3.9.7, Decay Time, and the associated Bases were developed and will be included in the next revision to the DCD. The decay time TS LCO limit is 100 hours, which is the assumed radioactive decay time for the fuel handling accident in DCD 15.7.4.

Although NUREG-1431, Rev. 2 STS does not include a TS for decay time, the AP1000 TS 3.9.7 was prepared in the STS format based on available information from TS 3/4.9.3, Decay Time, in NUREG-0452, Revision 5, "Standard Technical Specifications (STS) for Westinghouse Pressurized Water Reactors."

In addition, the information in the AP1000 TS 3.9.7 and its Bases is consistent with information in an improved TS 3.9.7 for decay time developed by Westinghouse as part of a TS improvement program using the NUREG-1431 format for a European nuclear power plant.

AP1000 TS 3.9.7 and its Bases discussion is very similar to the information for AP1000 TS 3.9.4 and its Bases, since both provide protection for fuel handling accidents.

Design Control Document (DCD) Revision:

See the attached TS 3.9.7 to be added to Section 3.9 of the AP1000 TS.

See the attached TS B 3.9.7 to be added to Section 3.9 of the AP1000 TS Bases.

PRA Revision:

None

3.9 REFUELING OPERATIONS

3.9.7 Decay Time

LCO 3.9.4 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTIONS

- NOTE -

LCO 3.0.8 is not applicable.

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	Reactor subcritical for less than 100 hours.	A.1	Suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify that the reactor has been subcritical for at least 100 hours by verification of the date and time of subcriticality.	Prior to movement of irradiated fuel in the reactor vessel

B 3.9 REFUELING OPERATIONS

B 3.9.7 Decay Time

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment or in the fuel handling area inside the auxiliary building requires allowing at least 100 hours for radioactive decay time before fuel assembly handling can be initiated. During fuel handling, this ensures that sufficient radioactive decay has occurred in the event of a fuel handling accident (Refs. 1 and 2). Sufficient radioactive decay of short-lived fission products would have occurred to limit offsite doses from the accident to within the values reported in Chapter 15.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the radioactivity decay time is an initial condition design parameter in the analysis of a fuel-handling accident inside containment or in the fuel handling area inside the auxiliary building, as postulated by Regulatory Guide 1.183 (Ref. 1).
	The fuel handling accident analysis inside containment or in the fuel handling area inside the auxiliary building is described in Reference 2. This analysis assumes a minimum radioactive decay time of 100 hours.
	Radioactive decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum radioactive decay time of 100 hours is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area inside the auxiliary building are within the values calculated in Reference 2.
APPLICABILITY	Radioactive decay time is applicable when moving irradiated fuel assemblies in containment or in the fuel handling area inside the auxiliary building. The LCO minimizes the possibility of radioactive release due to a fuel handling accident that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are also covered by LCO 3.7.11, "Fuel Storage Pool Water Level."

BASES

ACTIONS	LCO 3.0.8 is applicable while in MODE 5 or 6. Since movement of irradiated fuel assemblies with less than 100 hours of decay time can occur in MODE 6 after removing the reactor vessel head following the reactor shutdown, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 6, the fuel movement is independent of shutdown reactor operations since the reactor is already shutdown. Entering LCO 3.0.8 while in MODE 6 would not specify any action.
	A.1 With a decay time of less than 100 hours, all operations involving movement of irradiated fuel assemblies within containment or in the fuel handling area inside the auxiliary building shall be suspended immediately to ensure that a fuel handling accident cannot occur.
	The suspension of fuel movement shall not preclude completion of movement to safe position.
SURVEILLANCE REQUIREMENTS	SR 3.9.7 1 Verification that the reactor has been subcritical for at least 100 hours prior to movement of irradiated fuel in the reactor pressure vessel to the refueling cavity in containment or to the fuel handing area inside the auxiliary building ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Specifying radioactive decay time limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident (Ref. 2).
REFERENCES	1. Regulatory Guide 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."
	2. Section 15.7.4, "Fuel Handling Accident."

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.2.3.3-1

Original RAI Number(s): None

Summary of Issue:

The AP1000 insulation design was refined based on insights from the Configuration IV tests, and a prototypical insulation design for AP1000 was evaluated as part of the ULPU Configuration V test program. The applicant has indicated that the Configuration V test results show a further improvement in coolability performance relative to Configuration IV, and also include information on transient pressure loads needed by the COL-applicant to establish the pressure loads for the structural analysis of the final insulation design. The applicant has not provided documentation of: the RPV insulation design evaluated in Configuration V, the results of the Configuration V testing, or the functional requirements for the AP1000 RPV insulation system. Such information is needed in order for the staff to conclude on the margins to lower head failure for AP1000, and the viability of Westinghouse's proposal that the COL applicant complete the RPV insulation design. This is Open Item 19.2.3.3-1.

Westinghouse Response:

Attachment 3 to Westinghouse letter DCP/NRC1603 dated July 8, 2003 provides the ULPU V test report that can be used by the COL applicant to complete the RPV insulation design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: 19.2.6-1

Original RAI Number(s): None

Summary of Issue:

Deterministic Containment Capacity

The evaluation of ultimate capacity of the AP1000 containment is presented in DCD Tier 2 Section 3.8.2.4.2. In this section, the applicant evaluates the containment capacity at Service Level C limit by examining various parts of the containment structure, cylindrical shell, top and bottom heads, equipment hatches and covers, personnel airlocks, and mechanical and electrical penetrations. At Service Level C, the applicant determined that the capacity of the ellipsoidal head is 627 KPa (91 psig) at 149 ° C (300 ° F) and the capacity of the equipment hatch covers is 558 KPa (81 psig) at 149 ° C (300 °F) using NE 3222. Using Code Case N284, the capacity of the equipment hatch covers was determined to be 834 KPa (121 psig) at 149 ° C (300 ° F). The staff has always maintained that the provisions of Code Case N284 apply to local buckling cases only. The equipment hatch cover buckling is a global buckling phenomenon and therefore, the use of Code Case N284 is not appropriate. The Service Level C capacity of the AP1000 containment structure should be the lowest value, 558 KPa (81 psig) at 149_C (300 ° F). In Section 42.3.1 of the PRA, the applicant states, "The 90 psig [620 KPa] is the Service Level C containment failure pressure at 300_F." The staff does not agree with this assessment. The applicant should address why 558 KPa (81 psig) at 149 ° C (300 ° F) is not the limiting severe-accident pressure for the AP1000 containment. This is Open Item 19.2.6-1

Westinghouse Response:

Westinghouse has performed a stress analysis of the AP1000 containment structure following the requirements set forth in 10CFR50.34 noting that evaluation of instability is not required:

> (A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE - 3220, Service Level C Limits, *except that evaluation of instability is not required, considering pressure and dead load alone.* For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subsubarticle CC - 3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

The maximum internal pressure for the containment vessel satisfying the ASME Service Level C stress intensity limits (without consideration of buckling instability) is 117.2 psig at the 300°F design temperature. The results of the stress analysis show that the maximum pressure is

governed by the circumferential membrane stress in the cylindrical shell. This pressure is above the 90 psig pressure that is calculated in accordance with 10CFR 50.34 for Service Level C severe accident phenomena evaluation that includes hydrogen burn. It is noted that the capacity of the ellipsoidal head (91 psig at 300°F) and the equipment hatch covers (81 psig at 300°F) include buckling instability evaluation that is not part of the Service Level C evaluation requirements per 10CFR 50.34.

For AP600 and AP1000, Westinghouse has supplemented the requirements of 10CFR 50.34 by inclusion of an evaluation for buckling. Westinghouse has used the Service Level C pressure capacity of the equipment hatch cover calculated in accordance with ASME Code Case N284. This interpretation was reviewed by the ASME Code Committee and found to be acceptable.

Recognizing that regulations and structural criteria are now being evaluated on a risk informed basis, it is considered conservative to evaluate maximum pressure that is calculated for the very low probability event specified in accordance with 10CFR 50.34 against the limiting pressure corresponding to the 5th percentile failure probability. From this evaluation it is determined that the controlling failure mode is the cylindrical shell where failure is due to membrane yield since it has the lowest median pressure. The failure pressure will not be governed by which code criteria is used to define the buckling failure mode for the equipment hatch, and therefore, hatch buckling does not control the limiting severe accident pressure. For containment temperature of 400°F and 5th percentile failure probability the containment pressure is 106 psig.

Design Control Document (DCD) Revision:

None

PRA Revision:

Revise Section 42.3.1

It is noted that a containment conditional failure probability distribution for a containment temperature at 331°F which corresponds to saturation at 90 psig is also developed. This distribution is referenced in the discussion on passive containment cooling system (PCS) failure and fission-product release category CFL (see Chapters 34 and 45). ~~The 90 psig [620 KPa] is the Service Level C containment failure pressure at 300_F.~~ **The 90 psig [620 Kpa] is the maximum pressure that is calculated in accordance with 10CFR 50.34 for the severe accident phenomena for Service Level C evaluation that includes hydrogen burn.**

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19.2.6-2**

Original RAI Number(s): **None**

Summary of Issue:

Buckling of the Two 4.87 m (16 ft) Diameter Equipment Hatch Covers

The calculated critical buckling pressure for the equipment hatch covers is 1.45 MPa (211 psig) at ambient condition, as discussed in DCD Tier 2 Section 3.8.2.4.2.3. In Section 42.4.3 of the AP1000 PRA, it is shown that a factor of 1.5 was used as a multiplier to the calculated buckling pressure at ambient condition of 38 °C (100 ° F), based on the test head data. Using the multiplier of 1.5 and adjusting for the reduction in material strength due to temperature, the applicant has calculated the median capacity value for the buckling of the two 4.87 m (16 ft) diameter equipment hatch covers as 2.14 MPa (311 psig) at 166 ° C (331 ° F) and 2.05 Mpa (297 psig) at 204 ° C (400 ° F). However, as noted in DCD Tier 2 Section 3.8.2.4.2.2, one of the test results shows a reduction of 0.79 and the other test result shows a factor of 1.0 on the predicted BOSOR-5 value. Therefore, the staff considers that the use of the multiplier of 1.5 is not justified. Consequently, the staff does not agree with the values shown in Tables 42-1 and 42-2 of the PRA. This is Open Item 19.2.6-2.

Westinghouse Response:

The multiplier of 1.5 was addressed for the AP600 plant in DSER Open Item 19.2.6.3-6 and found acceptable. It is stated in NUREG-1512 (AP600 FSER, Section 19.2.6.3, page 19-193):

> For the containment equipment hatches, Westinghouse used 150 percent of the critical buckling pressure as the best estimate failure pressure on the basis of the test data. In the DSER, the staff requested that Westinghouse clarify in the SSAR whether this 50 percent increment is founded on either the lower bound or the median value of test data and justify the application of these test data to the AP600 equipment hatches. This was DSER Open Item 19.2.6.3-6.
>
> Westinghouse responded that the 50 percent increment of critical pressure for the best estimate failure pressure was based on the curve in ASME Code Case N-284, Revision 0 that was derived from the lower bound of tests. There was only one test specimen that was similar to the AP600 containment configuration ($M_i = 14.5$). However, using test data points provided by Westinghouse, the staff performed a regression analysis on the bases of the methodology provided in NUREG/CR-4604, and found that the median point at M_i of 14.5 is higher than 50 percent increment. Therefore, the 50 percent increment of critical pressure for the best estimate failure pressure is acceptable and, thus, DSER Open Item 19.2.6.3-6 is closed.

Welding Research Council Bulletin 267 shows a comparison of BOSOR-5 predictions of buckling against the results of 20 tests of small head models. Ratios (capacity reduction factors) of actual buckling to the BOSOR-5 prediction are summarized in Bulletin 267. Only one of the tests had a reduction factor less than 1.0. This low-capacity reduction factor of 0.79 is attributed to excessive imperfections associated with the fabrication of relatively thin plate (0.196 inch). These imperfections were visible and outside the tolerances permitted by the ASME Code. The results of test head 1 are therefore not considered applicable to the AP1000. This comparison data was used to support a capacity reduction factor of 1.0 for BOSOR-5 predictions.

Another approach was also used to investigate the AP1000 head capacity that is permitted in ASME Code, Case N284. This approach was developed after a detailed review and evaluation of test data. This was the approach that was used to close out AP600 DSER Open Item 19.2.6.3-6 as discussed in AP600 RAI Response 220.32. A copy of AP600 RAI response 220.32 is attached to this AP1000 DSER OI response.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Question 220.32

Westinghouse estimates critical buckling pressures for equipment hatches as 196 psig for a 22-foot-diameter hatch and 161 psig for a 16-foot-diameter hatch. The corresponding ASME Level C Service Limits are 117 psig and 96 psig using the Code Case N-284, respectively. From Figure 3.8.2-2, the equipment hatch covers appear convex to the center line of the containment. Therefore, the use of the Code Case N-284 (i.e., the factor of safety of 1.67 for the Level C Service Limit) is not acceptable because the internal pressure of the containment acts as the external pressure to the spherical cap covers and subjects the cap covers to compression. In the case of external pressure, ASME NE-3222 (i.e., the factor of safety of 2.5 for the Level C Service Limit) should be used for the compressive stresses. Note (1) in Table 3.8.2-2 should be revised to reflect the factor of safety of 2.5, or acceptable justification should be provided for not doing so (Section 3.8.2 of the SSAR).

Response:

Code Case N-284 provides criteria for evaluation of unstiffened spherical caps subjected to compressive stress due to pressure loading. In the Code Case, the theoretical buckling value is given in paragraph 1712.1.3, the capacity reduction factor is given in paragraph 1512 (b), and the plasticity reduction factor is given in paragraph 1620 and 1610 (a). The capacities of the hatch covers as described in the SSAR are in accordance with this code case.

ASME Code Case N-284 was developed by the code committee based on detailed review and evaluation of test data. Figure 220.32-1 shows test results from references 220.32-1 and 220.32-2 for fabricated steel hemispherical shells and spherical segments. The ratio of test buckling stress to theoretical buckling stress (α) is shown as a function of the non-dimensional unsupported length along the shell ($M = L_t/\sqrt{Rt}$, where L_t is the unsupported length along the spherical shell, R is the radius of the shell and t is the thickness of the shell). The lower bound curve to these data points, as shown in the figure, is used in Code Case N-284. For the AP600 16 foot diameter equipment hatch, M = 14.5, and the capacity reduction factor, corresponding to α in the figure, is 0.167. The stresses in the hatch cover are well below yield and the plasticity reduction factor is unity. The test data for shell lengths of 10 to 20 show capacities significantly above those of the Code Case. The capacity of the hatch covers, as calculated by the ASME Code Case, corresponds to the lower bound of the test data. As a result the 1.67 factor of safety specified in paragraph 1400 of the ASME Code Case is considered appropriate for calculating the Service Level C pressure capacities of the hatch covers.

SSAR Revision: NONE

References:

220.32-1 Kiernan, T.J. and Nishida, K., "The Buckling Strength of fabricated HY-80 Steel Spherical Shells," DTMB 1721, July 1966.

220.32-2 Arne, C., "Stiffened Spherical Shell Tests," CBI Contract C-1752, 1959

ATTACHMENT TO AP1000 DSER OPEN ITEM RESPONSE
19.2.6-2

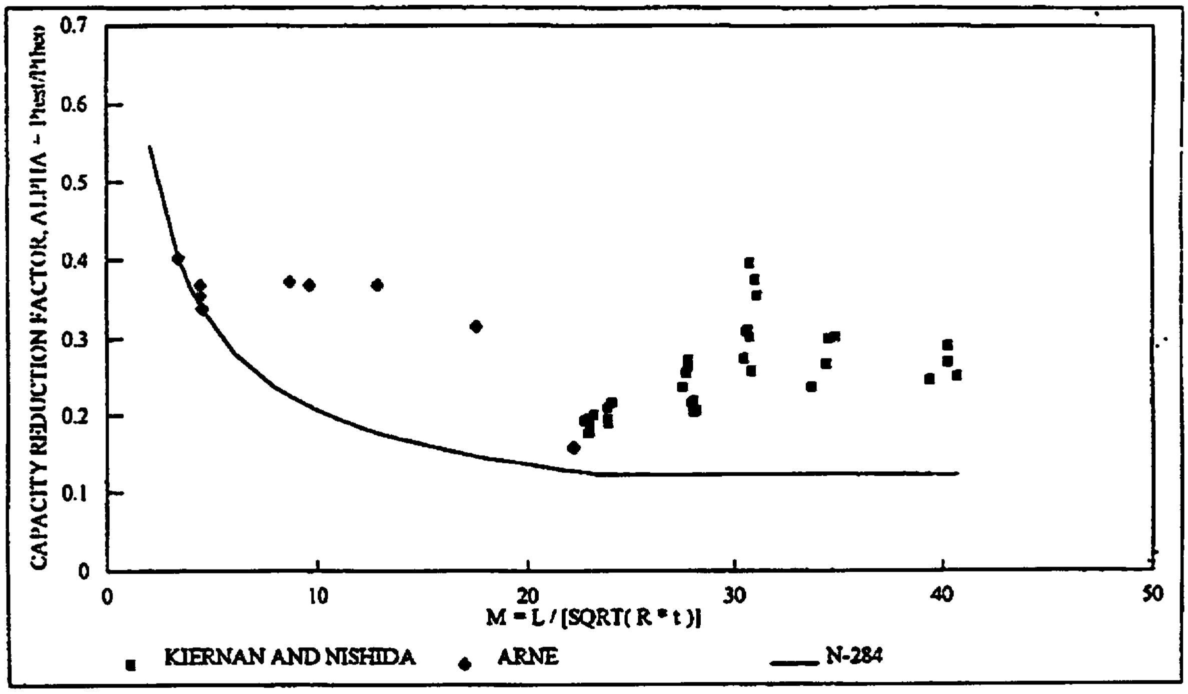

Figure 220.32-1 Comparison of Capacity Reduction Factors for Tests with Code Case N-284

AP1000 DESIGN CERTIFICATION REVIEW

Draft Safety Evaluation Report Open Item Response

DSER Open Item Number: **19.3.10-1**

Original RAI Number(s): 720.038

Summary of Issue:

In a letter dated March 28,2003, the applicant responded to RAI 720.038 by providing an evaluation of plant risk associated with internal floods at shutdown. The objective of this study was to confirm that the design incorporates adequate capability to achieve safe shutdown following these events, by showing that the associated plant risk is sufficiently small. Deterministic criteria were used to screen out any areas in which the risk from flooding is clearly insignificant, on the basis of the lack of flood initiation sources or absence of equipment important to safe shutdown, as modeled in the internal events PRA. Because the plant is already in shutdown, an initiating event for the shutdown analysis was considered an event leading to a threat to equipment needed for the normal decay heat removal function.

Based on the staff's preliminary review of this letter, it appears to have errors in the calculated CDF for two of the eight sequences. The applicant needs to address these errors and the staff needs to complete its review. This is Open Item 19.3.10-1.

Westinghouse Response:

Based on discussions with the NRC project, the errors referred to in this DSER open item are related to the description of Scenarios 5 and 6 of the shutdown flooding evaluation that was included in PRA Chapter 56.6.4. For these scenarios, the calculation of CDF is shown as the product of the initiating event frequency (IEV), and the conditional core damage frequency (CCDF), and a penalty factor of 100 to account for a reduced reliability of the class 1E signals to the safety-related systems resulting from consequences of the flood.

For scenario 5, the following calculation is shown:

$$\underset{\text{(CCDF)}}{4.43\text{E-}05} \times 100 \times \underset{\text{(IEV)}}{3.14\text{E-}09} = \underset{\text{(CDF)}}{1.39\text{E-}13} \text{ per year}$$

This equation contains two errors, however the calculated CDF for this scenario is correct as shown. The CCDF as shown in the equation above already includes the penalty factor of 100. This can be seen by inspection of PRA tables 56-5 and 56-6, which shows the shutdown IEV CCDP for this scenario to be 4.432E-07.

The same errors are also made in the Scenario 6 calculation of the CDF, however the reported CDF for this scenario is correct. The PRA will be revised as shown.

Design Control Document (DCD) Revision:

None

PRA Revision:

PRA Chapter 56 will be revised as follows:

Scenario 5: Annex Building 135'-3" North Air Handling Equipment Area – Hot/Cold Shutdown – Rupture of 8-inch Fire Main Extension

The applicable initiating event from the shutdown PRA for this scenario is loss of decay heat removal due to RNS failure (IEV-RNSND). Due to the small initiating event frequency, 3.14E-09, and the plant systems that would remain available to respond to this event, this initiator was not explicitly quantified. Although the class 1E dc power batteries are failed, the ac feeds to the inverters and to the protection and safety monitoring system (PMS) cabinets remain intact and the safety-related systems should operate as expected. The non-class 1E dc failures would fail the diverse actuation system backup signals in the shutdown model. Conceding a reduced reliability of the signals to the safety-related systems, the flooding CDF for this initiator is estimated by putting a factor of 100 penalty on the conditional core damage probability for the loss of decay heat removal due to RNS failure (hot/cold shutdown):

4.43E-07~~5~~ x 100 x 3.14E-09 = 1.39E-13 per year |

Scenario 6: Annex Building 135'-3" North Air Handling Equipment Area – Reactor Coolant System Drained – Rupture of 8-inch Fire Main Extension

The applicable initiating event from the shutdown PRA for this scenario is loss of decay heat removal due to RNS failure (IEV-RNSD). Due to the small initiating event frequency, 6.39E-10, and the plant systems that would remain available to respond to this event, this initiator is not explicitly quantified. Although the class 1E dc power batteries are failed, the ac feeds to the inverters and to the protection and safety monitoring system cabinets remain intact and the safety-related systems should operate as expected. The non-class 1E dc failures would fail the diverse actuation system backup signals in the shutdown model. Conceding a reduced reliability of the signals to the safety-related systems, the flooding CDF for this initiator is estimated by putting a factor of 100 penalty on the conditional core damage probability for the loss of decay heat removal due to RNS failure (reactor coolant system drained):

1.18E-04~~2~~ x 100 x 6.39E-10 = 7.53E-12 per year |

DCP/NRC1603
Docket 52-006

July 8, 2003

Attachment 3

CRSS-03/06

"Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility"

University of California Santa Barbara

dated June 30, 2003