Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5148
DCP/NRC0882
Docket No.: STN-52-003

May 21, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: INFORMAL CORRESPONDENCE

Dear Mr. Quay:

Please find enclosed a formal transmittal of correspondence we have previously sent to you informally. This informal correspondence was sent over the period May 8, 1997 through May 14, 1997.

Attachment 1 provides the index of the attached material as you have requested.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment
Enclosure

cc: N. J. Liparulo, Westinghouse (w/o Attachment, Enclosure)
M. M. Slosson, NRC (w/o Enclosure)

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Attachment 1 to Westinghouse Letter DCP/NRC0882

DATE	ADDRESSEE	DESCRIPTION
5/12/97	Huffman	Markup of TS 3.7.6 based on 5/9 telecon.
5/14/97	Huffman	Markup of SSAR chapter 18 addressing J. Wilson's concerns from telecon.
5/13/97	Sebrosky	Fax of additional page to letter of 5/9/97 on initial test program.
5/9/97	Scaletti	Information related to open item 750. Originally submitted 4/15/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/14/97	Huffman	Information related to open item 2277 concerning LTOP valve sticking open.
5/13/97	Scaletti	Information related to open item 4117. Originally submitted 2/21/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/13/97	Scaletti	Information related to open item 4118. Originally submitted 2/21/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/13/97	Scaletti	Information related to open item 4119. Originally submitted 2/21/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/13/97	Scaletti	Information related to open item 4121. Originally submitted in letter 2/21/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/8/97	Scaletti	Information related to open item 628. Originally submitted in letter 3/26/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/8/97	Scaletti	Information related to open item 649. Originally submitted in letter 3/26/97 and included in SSAR revision 12. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".

5/8/97	Scaletti	Information related to open item 662. Originally submitted in letter 3/26/97 and included in SSAR revision 12. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/8/97	Scaletti	Information related to open item 664. Originally submitted in letters 3/18/97 and 3/26/97 and included in PRA revision 9. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/9/97	Scaletti	Information related to open item 5032. Originally submitted in letter 4/15/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/9/97	Scaletti	Information related to open item 5030. Originally submitted in letter 4/15/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/9/97	Scaletti	Information related to open item 5028. Originally submitted in letter 4/15/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/9/97	Scaletti	Information related to open item 766. Originally submitted in letter 4/15/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/8/97	Scaletti	Information related to open item 5031. Originally submitted in letter 4/15/97 and revision 12 of the SSAR. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/9/97	Scaletti	Information related to open item 668. Originally submitted in letter 5/2/97. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/8/97	Scaletti	Information related to open item 1885. Originally submitted in letter 3/26/97 and revision 12 of the SSAR. Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/13/97	Jackson	OITS status.
5/9/97	Scaletti	Information related to open item 820. Originally in SSAR revision 12.
5/12/97	Jackson	Information related to open item 363. References open item 358 which was answered by letter of 4/25/97. Awaiting your decision to change NRC status.

5/12/97	Jackson	SSAR markup to resolve open item 3967. Same as item 11-5.2 from 11/5/96 phone call. Will go into SSAR revision 13 unless we hear otherwise.
5/12/97	Scaletti	Information related to open item 2271. Originally submitted in letter of 8/13/96 and revision 10 of the SSAR (12/20/96). Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".
5/12/97	Scaletti	Information related to open item 1792. Originally submitted in letters of 8/13/96 10/23/96 and revision 11 of the SSAR (2/28/96). Request NRC review material and provide definitive action or provide direction to change NRC status to "Action N" or "closed".

To: Bill Huffman / Janek Raval
From: Robin Nydes

cc: Chip Suggs
Dan McDermott
Mark Wills
Ed Cummins
Jim Winters
Ron Vijuk
Brian McIntyre

Attached is a markup of TS 3.7.6 based on the telecon we had Friday 5/9. Also attached is the 1st draft of the transmittal letter and a ~~cop~~ report of 2 new OITS items resulting from our telecon. Red Team review is requested (since SSAR sections 16.1 and 3.9 are affected) by COB Thursday so I can get this out 5/16.

Please let me know if you have any comments. One item worth noting is in the rewritten BASES for SR 3.7.6.9: although this isn't the SECY wording verbatim, I believe it resolves the NRC comment without ambiguity. (SECY words attached.)

Nydes, Robin K.

From: Nydes, Robin K.
Sent: Monday, May 12, 1997 2:20 PM
To: Lovato, Janet M.
Cc: Suggs, Charles W.; McDermott, Daniel J.; Wills, Mark E.; McIntyre, Brian A; Nydes, Robin K.
Subject: TS 3.7.6 Letter for Typing Please

Janet - Please put degree signs in.

To: DCD

cc: Bill Huffman (NRC)
Janek Raval (NRC)
Harold Walker (NRC)
Angela Chu (NRC)
Nick Liparulo

bcc: Robin Nydes
Chip Suggs
Dan McDermott
Mark Wills
+ "std" dist

Subject: Advance Copy of Revised AP600 Technical Specification 3.7.6

Reference: 1. NSD-NRC-97-5029 (DCP/NRC0776), "Response to Some of the Technical Specification Comments from NRC Letter of 12/24/96", dated 3/27/97.

Attached is a markup of AP600 Technical Specification 3.7.6, Main Control Room Habitability System. These changes resolve comments discussed during a 5/9/97 telecon involving Messrs. Huffman, Raval, and Walker of the NRC and Messrs. Suggs, Wills, and McDermott and Ms. Nydes of Westinghouse. During this telecon, we discussed the Reference 1 NRC comment and Westinghouse response for item 31, as well as a markup provided by the NRC via fax on 5/8/97.

The changes are described as follows.

1) To address the NRC concern regarding LCO 3.7.6 applicability in MODE 4 and the REQUIRED ACTIONS (D.2 and F.2) leading to MODE 4, a markup of the LCO and BASES is attached. The action for D.2 is now to go to MODE 5 while the action for F.2 is to restore one VES train to OPERABLE status. Westinghouse believes this resolves the MODE 4 applicability/endstate issue for LCO 3.7.6 ~~is comment~~.

2) The main control room (MCR) air temperature for SR 3.7.6.1, and in the BACKGROUND section, is corrected to 78 **(degree sign)** F (from 80 **x**F).

3) The APPLICABLE SAFETY ANALYSIS discussion of the safety signals which actuate the MCR habitability system (VES) has been clarified to include ~~high-high~~ High-2 particulate or iodine radioactivity.

4) The BASES for SR 3.7.6.2 has been revised to include the compressed air storage tank pressure range from the LCO.

5) The BASES for SR 3.7.6.9, regarding the VES air delivery train test has been revised extensively to include a pressure and flow rate, as well as a more detailed discussion regarding the duration of the test. SECY-95-132 was added as a reference for this revision.

A markup of SSAR page 3.9-168 Note 8 is also included to incorporate the pressure and flow rate of SR 3.7.6.9.

Item 5250 was added to the Open Item Tracking System (OITS) to track completion of Technical Specification 3.7.6. While this submittal completes the Action -W portions of OITS item 5250, two actions remain:

g. NRC to confirm acceptance of MODE APPLICABILITY for LCO 3.7.6.

h. NRC to determine acceptability of the damper surveillance frequencies SRs 3.7.6.5 and 3.7.6.7.

In addition, although not a Technical Specification issue, during the telecon, Westinghouse (Mark Wills) took an action to determine the status of developing the sampling program which will include air storage tank air quality testing. This action has been logged as OITS item 5251.

Given NRC concurrence with the attached markups by 5/15/97, these changes can be incorporated into SSAR Revision 13 at the end of May. To meet the schedular objectives of SECY-97-051 for Westinghouse to provide the NRC with the final SSAR, which includes the AP600 Technical Specifications in Section 16.1, by 5/30/97, the NRC is requested to provide:

1) concurrence with (or comments on) the attached markups,
2) completion of NRC action for OITS item 5250 parts g and h, and
3) an NRC status revision for an OITS update of 5250.

Please contact Robin K. Nydes at (412)374-4125 if you have any questions regarding this transmittal or the AP600 Technical Specifications.

BAM

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/12/97

Selection: [item no] between 5250 And 5251 Sorted by Item #

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
5250	NRR/TSB	16.1	TEL-OI		TECHSPEC	Action N	Action W		

A telecon was held 5/9/97 for TS 3.7.6. The following actions resulted:

a) Action -W: Address NRC concern regarding LCO 3.7.6 applicability in MODE 4 for REQUIRED ACTIONS D.2 and F.2.

b) Action-W: Correct the MCR air temperature frm 80 to 78 deg F for SR 3.7.6.1, and in the BACKGROUND section.

c) Action-W: Clarify the APPLICABLE SAFETY ANALYSIS discussion of the safety signals which actuate the MCR habitability system (VES) to include high high particulate or iodine radioactivity.

d.) Action-W: Revise the BASES for SR 3.7.6.2 to include the compressed air storage tank pressure range from the LCO.

e) Action-W: Revise the BASES for SR 3.7.6.9, regarding the VES air delivery train test to include a pressure and flow rate, as well as a more detailed discussion regarding the duration of the test. Ref: SECY-95-132

f) Markup SSAR page 3.9-168 Note 8 to incorporate the pressure and flow rate of SR 3.7.6.9.

g) Action-N: NRC to confirm acceptance of MODE APPLICABILITY for LCO 3.7.6.

h) Action-N: NRC to determine acceptability of the damper surveillance frequencies SRs 3.7.6.5 and 3.7.6.7.

Letter transmitting TS and SSAR markup went to typing 5/12.
Copy was faxed to NRC for concurrence prior to formal transmittal 5/12.

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
5251			TEL-OI		Wills, Mark	Action W	Action W		

During a 5/9/97 Tech Spec 3.7.6 (VES) telecon, Westinghouse (Mark Wills) took an action to determine the status of developing the sampling program which will include air storage tank air quality testing.

 Total Records: 2

3.7 PLANT SYSTEMS

3.7.6 Main Control Room Habitability System (VES)

LCO 3.7.6 Two Main Control Room (MCR) Habitability System trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One VES train inoperable.	A.1 Restore VES train to OPERABLE status.	7 days
B. MCR air temperature not within limit.	B.1 Restore MCR air temperature to within limit.	24 hours
C. Loss of integrity of MCR pressure boundary.	C.1 Restore MCR pressure boundary to OPERABLE status	24 hours
D. Required Action and associated Completion Time of Conditions A, B, or C not met in MODE 1, 2, 3, or 4.	D.1 Be in MODE 3. __AND__ D.2 Be in MODE ~~4~~ 5.	8 hours ~~24~~ 36 hours
E. Required Action and associated Completion Time of Conditions A, B, or C not met during movement of irradiated fuel.	E.1 Suspend CORE ALTERATIONS. __AND__ E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two VES trains inoperable in MODE 1, 2, 3, or 4.	F.1 Be in MODE 3.	8 hours
	AND	
	F.2 Be in MODE 4.	24 hours
	AND	
	F.3 Restore one VES train to OPERABLE status.	36 hours
G. Two VES trains inoperable during movement of irradiated fuel.	G.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	G.2 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify Main Control Room air temperature is ≤ ~~80~~°F. 78	24 hours
SR 3.7.6.2	Verify that the compressed air storage tanks are pressurized to [≥ 3400 psig but ≤ 3600 psig].	24 hours
SR 3.7.6.3	Verify that each VES air delivery isolation valve is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.4	Verify that each VES air header manual isolation valve is in an open position.	31 days
SR 3.7.6.5	Verify that all VBS Main Control Room isolation dampers are OPERABLE and will close upon receipt of an actual or simulated actuation signal.	24 months
SR 3.7.6.6	Verify that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.7	Verify that each VES pressure relief damper is OPERABLE.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.6.8	Verify that the self contained pressure regulating valve in each VES train is OPERABLE.	In accordance with the Inservice Testing Program
SR 3.7.6.9	Verify that one VES air delivery train maintains a positive pressure in the MCR, relative to the adjacent areas, at the required air addition flowrate.	In accordance with the Inservice Testing Program

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System

BASES

BACKGROUND

The Main Control Room Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 main control room (MCR) radiation signal is received, the VES is actuated. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air for the MCR occupants; 2) to provide forced ventilation to maintain the MCR at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; and 3) to limit the temperature increase of the MCR equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

The VES consists of two redundant trains each with compressed air storage tanks and associated valves, piping, and instrumentation. Each set of tanks contains enougn breathable air to supply the required air flow to the MCR for at least 72 hours. The VES system is designed to maintain CO_2 concentration less than 0.5% for up to 11 MCR occupants with both trains operating. With one train operating, VES maintains CO_2 concentration less than 0.5% for up to 5 MCR occupants, and maintains CO_2 concentration less than 1.0% for up to 11 MCR occupants.

Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCR, which is initially at or below ~~80~~ 78°F. Heat sources inside the MCR include operator workstations, emergency lighting and occupants. Sufficient insulation is provided surrounding the MCR pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling.

(continued)

BASES

BACKGROUND (continued)

If the VBS remains unavailable following the 72 hour period, cooling of the MCR air is achieved by portable air coolers.

The compressed air storage tanks are initially pressurized to 3400 psig. During operation of the VES, a self contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. An orifice downstream of the regulating valve is used to control the air flow rate into the MCR. The MCR is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas.

APPLICABLE SAFETY ANALYSES

Two redundant sets of compressed air storage tanks are sized such that each set of tanks has a combined capacity that provides at least 72 hours of VES operation.

Operation of the VES is automatically initiated by either of two safety related signals: 1) loss of all AC power, or 2) ~~high-high MCR radiation~~ High-2 particulate or iodine radioactivity.

In the event of a loss of all AC power, the VES functions to provide ventilation, pressurization, and cooling of the MCR pressure boundary.

In the event of a high ~~radiation environment~~ level of gaseous radioactivity outside of the MCR, the VBS continues to operate to provide pressurization and filtration functions. The MCR air supply downstream of the filtration units is monitored by a safety related radiation detector. for High-2 particulate or iodine radioactivity Upon exceeding a predetermined setpoint, or a loss of all ac power, a safety related signal is generated to isolate the MCR from the VBS and to initiate air flow from the VES storage tanks. Isolation of the VBS consists of closing safety related ~~devices~~ dampers in the supply and exhaust ducts that penetrate the MCR pressure boundary. VES air flow is initiated by a safety related signal which opens the isolation valves in the VES supply lines.

The VES functions to mitigate a DBA or transient that either assumes the failure of or challenges the integrity of the fission product barrier.

The VES satisfies the requirements of Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The VES limits the MCR temperature rise and maintains the MCR at a positive pressure relative to the surrounding environment.

Two independent and redundant VES trains are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train.

The VES is considered OPERABLE when the individual components necessary to deliver a supply of breathable air to the MCR are OPERABLE in both trains. This includes components listed in SR 3.7.6.2 through 3.7.6.8. In addition, the MCR pressure boundary must be maintained, including the integrity of the walls, floors, ceilings, duct work, electrical and mechanical penetrations, and access doors.

APPLICABILITY

The VES is required to be OPERABLE in MODES 1, 2, 3, and 4 and during movement of irradiated fuel because of the potential for a fission product release following a DBA. The VES is not required to be OPERABLE in MODES 5 and 6 when irradiated fuel is not being moved because accidents resulting in fission product release are not postulated.

ACTIONS

A.1

When one VES train is inoperable, action is required to restore the system to OPERABLE status. A Completion Time of 7 days is permitted to restore the train to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the fuel, the low probability of not containing the radiation, and that the remaining train can provide the required capability.

B.1

When the main control room air temperature is outside the acceptable range during VBS operation, action is required to restore it to an acceptable range. A Completion Time of 24 hours is permitted based upon the availability of temperature indication in the MCR. It is judged to be a sufficient amount of time allotted to correct the deficiency in the nonsafety ventilation system before shutting down.

(continued)

BASES

ACTIONS (continued)

C.1

If the MCR pressure boundary is damaged or otherwise degraded, action is required to restore the integrity of the pressure boundary and restore it to OPERABLE status within 24 hours. A Completion Time of 24 hours is permitted based upon operating experience. It is judged to be a sufficient amount of time allotted to correct the deficiency in the pressure boundary.

D.1 and D.2

In MODE 1, 2, 3, or 4 if Conditions A, B, or C cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. This is done by entering MODE 3 within 8 hours and MODE ~~4~~ 5 within ~~24~~ 36 hours.

E.1 and E.2

During movement of irradiated fuel assemblies, if the inoperable VES train cannot be restored to OPERABLE status, Required Actions A.1, B.1, or C.1 cannot be completed within the required Completion Time, the movement of fuel and core alterations must be suspended. Performance of Required Action E.1 and E.2 shall not preclude completion of actions to establish a safe condition.

F.1 and F.2

If both VES trains are inoperable in MODE 1, 2, 3, or 4, the VES may not be capable of performing the intended function, and must be brought to MODE 4, where the probability and consequences of an event are minimized. ~~Therefore,~~ To accomplish the MODE reduction, MODE 3 must be achieved within 8 hours and MODE 4 within 24 hours. and one VES train must be restored to OPERABLE status within 36 hours.

G.1 and G.2

During movement of irradiated fuel assemblies with two VES trains inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity that might enter the MCR. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

The MCR air temperature is checked at a frequency of 24 hours to verify that the VBS is performing as required to maintain the initial condition temperature assumed in the safety analysis, and to ensure that the MCR temperature will not exceed the required conditions after loss of VBS cooling. The 24 hour Frequency is acceptable based on the availability of temperature indication in the MCR.

SR 3.7.6.2

Verification every 24 hours that compressed air storage tanks are pressurized to [≥ 3400 psig but ≤ 3600 psig] ~~[3400 psig]~~ is sufficient to ensure that there will be an adequate supply of breathable air to maintain MCR habitability for a period of 72 hours. The Frequency of 24 hours is based on the availability of pressure indication in the MCR.

SR 3.7.6.3

VES air delivery isolation valves are required to be verified as OPERABLE. The Frequency required is in accordance with the Inservice Testing Program.

SR 3.7.6.4

VES air header isolation valves are required to be verified open at 31 day intervals. This SR is designed to ensure that the pathways for supplying breathable air to the MCR are available should loss of VBS occur. These valves should be closed only during required testing or maintenance of downstream components, or to preclude complete depressurization of the system should the VES isolation valves in the air delivery line open inadvertently or begin to leak.

SR 3.7.6.5

Verification that all VBS isolation devices are operable and will actuate upon demand is required every 24 months to ensure that the MCR can be isolated upon loss of VBS operation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.6

Verification that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE is required in accordance with the Inservice Testing Program. The SR is used in combination with SR 3.7.6.7 to ensure that adequate vent area is available to mitigate MCR overpressurization.

SR 3.7.6.7

Verification that the VES pressure relief damper is OPERABLE is required at 24 month intervals. The SR is used in combination with SR 3.7.6.6 to ensure that adequate vent area is available to mitigate MCR overpressurization.

SR 3.7.6.8

Verification of the operability of the self-contained pressure regulating valve in each VES train is required in accordance with the Inservice Testing Program. This is done to ensure that a sufficient supply of air is provided as required, and that uncontrolled air flow into the MCR will not occur.

SR 3.7.6.9

Per Reference 3, a functional test is required to establish ~~verify demonstrate~~ that one VES air delivery train, using ~~including~~ the ***safety-related*** compressed air source, is able to pressurize the MCR envelope to a positive ***1/8 inch water gauge*** pressure relative to the surrounding spaces ~~adjacent areas~~, at the required air addition flowrate ***of 25 ± 2 scfm***. The test ***need not last 72 hours***, ~~is~~ only ~~required to last~~ long enough to demonstrate ~~attainment of~~ ***ability to achieve*** the desired differential pressure~~; it is not meant or intended to last 72 hours in duration~~. ***The control room leakage rate must be within the design capacity of the safety-related systems to pressurize the control room for 72 hours.*** One air delivery train is tested on an alternating basis, in accordance with the System Level Inservice Testing Program.

REFERENCES

1. AP600 SSAR, Section 6.4, "Main Control Room Habitability Systems."
2. AP600 SSAR, Section 9.4.1, "Nuclear Island Non-Radioactive Ventilation System."
3. SECY-95-132, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)", May 22, 1995.

available for maintaining control room habitability during a postulated DBA. Therefore, the amount of credit that can be taken for the non-safety system in the safety analysis for design-basis accidents will be determined as part of the regulatory treatment of non-safety systems process as discussed in Section A of this paper.

The SRP guidance for ventilation systems that will pressurize the control room during a radiation emergency distinguishes between three categories of systems: systems having pressurization rates of (1) greater than or equal to 0.5 volume changes per hour, (2) less than 0.5 and greater than or equal to 0.25 volume changes per hour, and (3) less than 0.25 volume changes per hour. The guidance states that, for systems in the second and third categories, the planned leaktight design features should be analyzed to ensure that the design makeup air flow can maintain 1/8-inch water gauge differential. For systems in the third category, tests should be performed every 18 months to verify that the makeup rate is within ± 10 percent of the design rate and that the control room can be pressurized to at least 1/8-inch water gauge relative to all surrounding air spaces while makeup air is applied at the design rate.

The staff's position is that the pressurization tests for passive ALWR control rooms should be performed every refueling outage using the safety-related pressurized air bottles, although the staff would consider other testing methods proposed by a COL applicant. The test would not need to last 72 hours, only long enough to demonstrate that the main control room envelope can be maintained at a positive 1/8-inch water gauge relative to all surrounding air spaces while makeup air is applied at ± 10 percent of the design rate. The control room leakage rate must be within the design capacity of the safety-related systems to pressurize the control room for 72 hours. As with current plants, the test only pressurizes the main control room envelope to 1/8-inch water gauge, so the main control room would be manned as usual and access to the control room need not be restricted during the test.

Robin, wants that [illegible] would like to use

In addition to the pressurization tests performed at every refueling outage, an initial test using the safety-related air bottles will be conducted as part of the ITAAC as proposed by industry. This test would establish that the air bottles are capable of maintaining the required positive pressure in the control room envelope for 72 hours. As with the refueling outage surveillance, the test only pressurizes the main control room to 1/8-inch water gauge, so the main control room would be manned as usual and access to the control room need not be restricted during the test.

The staff reviewed the EPRI proposal for a safety-grade pressurization system and determined the following:

- The present licensing of nuclear power plants does not require the licensee to have ESF ventilation systems unless the licensee cannot meet the dose criteria associated with the DBAs or other safety criteria. If the licensee cannot meet these criteria, it must ensure that an ESF system or some other safety-grade system is available to mitigate the consequences of a DBA.

- The use of a passive safety-grade pressurization system, such as a bottled air system, may not preclude the need for other safety-grade equipment within the control room. For example, such safety-grade equipment could be required to maintain cooling to the electrical instruments in the control room.

- An initial test using the safety-related air bottles will be conducted as part of the ITAAC, as proposed by industry. This would establish that the air bottles are capable of maintaining the required positive pressure in the control room for 72 hours.

- At least once each refueling cycle, the licensee must perform pressurization tests to demonstrate that the control room can be pressurized for a 72-hour period. The pressurization tests for passive ALWR control rooms should be performed using the safety-related pressurized air bottles or other approved testing methods. The tests would not need to last 72 hours, only long enough to establish that the safety-related air system can pressurize the main control room envelope to 1/8-inch water gauge with respect to the surrounding spaces. The control room leakage rate must be within the design capacity of the safety-related air bottles to pressurize the control room for 72 hours.

- The regulatory treatment of the portable air supply and the non-safety-grade ventilation system will be in accordance with the staff's position on the regulatory treatment of non-safety systems process as described in Section A of this paper.

- The staff is continuing to discuss with industry the number of people in control room that can be supported for 72 hours by the safety-related air bottles.

In its letter of August 17, 1992, the Advisory Committee on Reactor Safeguards (ACRS) stated that the members had discussed control room habitability with EPRI and the staff during a June 4 and 5, 1992, meeting. At that meeting, the staff said that it was evaluating the EPRI proposal for the safety-grade pressurization system. ACRS stated that it had several comments about the design features of the passive control room pressurization system proposed by EPRI. The ACRS stated that the staff should consider these comments in evaluating this issue and that the ACRS may make additional recommendations after the staff has completed its evaluation. As committed to in the October 29, 1992, response to the ACRS, the staff has considered the ACRS comments in finalizing its position on this issue. Further discussions with the ACRS regarding the passive control room habitability systems will be conducted during the vendor-specific reviews.

The staff will evaluate the feasibility and the capability of the proposed pressurization systems on a vendor-specific basis. The staff will review the designs for control room habitability, including the refueling outage pressurization surveillance tests as discussed above, to ensure that the requirements in GDC 19 and guidelines in SRP Section 6.4 are met and that personnel and

****
*** TX REPORT ***
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TRANSMISSION OK

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RESULT	OK

5. The flow capability of each IRWST injection line is demonstrated every 10 years. This demonstration is accomplished by conducting flow tests and inspections. A flow test is conducted to demonstrate the flow capability of the injection line from the IRWST through the IRWST injection check valves. Water flow from the IRWST through the IRWST injection check valve demonstrates the flow capability of this portion of the line. The test is terminated when the flow measurement is obtained. The portion of the line from the IRWST squib valve to the DVI line is demonstrated by an inspection of the inside of the line. The inspection will show that the lines are not obstructed. It is not necessary to operate the IRWST injection squib valves for this inspection.

6. The flow capability of each containment recirculation line is demonstrated every 10 years. This demonstration is accomplished by conducting an inspection. The line from the containment to the containment recirculation squib valve is inspected from the containment side. The line from the squib valve to the IRWST injection line is inspected from the IRWST side. The inspection will show that the lines are not obstructed. It is not necessary to operate the containment recirculation squib valves for this inspection.

7. The heat transfer capability of the passive residual heat exchanger is demonstrated every 10 years. This demonstration is accomplished by conducting a test during cold shutdown conditions. The test is conducted with the RCPs in operation and the RCS at a reduced temperature. Flow through the heat exchanger is initiated by opening one outlet isolation valve. The test is terminated when the flow and temperature measurements are obtained.

8. The MCR pressurization capability is demonstrated during each refueling cycle. The test is conducted with the normal HVAC lines connected to the MCR isolated. Pressurization of the MCR is initiated by opening one of the emergency MCR habitability air supply lines. The test is a limited duration test and is terminated when the MCR pressurization is measured, with an air flow rate of 25 ± 2 scfm.
to 1/8 inch water gauge with respect to surrounding areas

9. The hydrogen recombination capability is demonstrated by performing a surveillance bench test of samples removed from each passive autocatalytic recombiner during each refueling outage. In addition, each passive autocatalytic recombiner device is visually inspected to verify that there is no obstruction of blockage of the inlets or outlets.

To: Bill Huffman
From: Robin Nydes

To: Bill Huffman —

Resend of
Last 4 pgs.
Did you get all
of this?

Please copy Joe Sebrosky
and Jim Bongarra. Thanks

Here is a markup of SSAR Ch/8 for NRC consideration. I believe we addressed Jerry Wilson's concerns during the telecon; this markup is expected to address/resolve Jim's concerns. To make the SECY date of end-May for our submittals, I should turn in any SSAR changes by May 16 so we'd appreciate a quick turnaround on this.

Thanks,

Robin Nydes
5/14 p1 of 13

Please give me a call as soon as NRC can provide feedback.

18.8 Human System Interface Design

This section provides an implementation plan for the design of the human system interface (HSI) and information on the human factors design for the non-HSI portion of the plant. The human system interface includes the design of the operation and control centers (OCS) and each of the human system interface resources. Execution and documentation of this implementation plan is the responsibility of the Combined License applicant.

The operation and control centers includes the main control room, the technical support center, the remote shutdown ~~facility~~ room, ~~operational support center~~ emergency operations facility, local control stations and associated workstations for each of these centers. The AP600 human system interface resources include:

- Wall panel information system
- Alarm system
- Plant information system
- ~~Computerized procedure system~~ Procedure System
- Soft controls/dedicated controls
- Qualified data processing system

The wall panel information station presents information about the plant for use by the operators. No control capabilities are included. The wall panel information station provides dynamic display of plant parameters and alarm information so that a high level understanding of current plant status can be readily ascertained. It is located at one end of the main control area at a height such that both operators and the shift supervisor can view it while sitting at their respective workstations. This panel provides information important to maintaining the situation awareness of the crew and for supporting crew coordination. The wall panel information station provides a dynamic plant display of the plant. It also serves as the alarm system overview panel display. The display of plant disturbances (alarms) and plant process data are integrated on this wall panel information station display. The wall panel information station is a nonsafety-related system. It is designed to have a high level of reliability.

The mission of the AP600 alarm system, together with the other human system interface resources, is to provide the operations and control centers operating staff with the means for acquiring and understanding the plant's behavior. The alarm system improves the performance of the operating crew members, when acting both as individuals and as a team, by improving the presentation of the plant's process alarms. The alarm system supports the control room crew members in the following steps or activities of Rasmussen's operator decision-making model (Reference 25):

- The "alert" activity, which alerts the operator to off-normal conditions

- The "observe what is abnormal" activity, which aids the user in focusing on the important issue(s)

- The process "state identification" activity, which aids the user in understanding the abnormal conditions and provides corrective action guidance. It guides the operating crew into the information display system.

The plant information system presents plant process information for use by the operators. The plant information system provides dynamic display of plant parameters and alarm information so that an understanding of current plant conditions and status is readily ascertained. The plant information system uses color-graphic video display units located on the operations and control centers workstations to display plant process data. These displays provide information important to monitoring, planning, and controlling the operation of plant systems and obtaining feedback on control actions.

The ~~computerized~~ procedure ~~system~~ system has a mission to assist plant operators in monitoring and controlling the execution of plant procedures. ~~The computerized procedures system is a software system. It runs on the hardware selected for the operations control centers.~~ The ~~computerized~~ procedure ~~system~~ system is accessible from the operator workstations in the main control room. Procedure development, as stated in Section 13.5 and 18.9, is the responsibility of the Combined License applicant. A procedure writer's guide is developed as part of the human system interface design implementation plan for the ~~computerized~~ procedure ~~system~~ system. The writer's guide is the design guidelines document for the ~~computerized~~ procedure ~~system~~ system. Information on the writer's guide and on the ~~computerized~~ procedure ~~system~~ system is found in Reference 31. Man-in-the-loop concept tests (Reference 9) are planned as part of the human system interface design implementation plan. These tests determine how effectively the ~~computerized~~ procedure~~s~~ system handles plant situations and ~~whether computer-based procedures adequately~~ support operator performance. ~~The design of a backup to the computerized procedure system, to handle the unlikely event of a loss of the computerized procedure system, is developed as part of the human system interface design process. Design options include the use of a paper backup. The acceptability of the backup is evaluated through concept testing or by executing a walk-through using the full-scale mockup of the AP600 main control room.~~ The ~~computerized~~ procedure ~~system and its backup are~~ system is evaluated as part of the integrated system validation phase of the human factors verification and validation (Reference 24).

The mission of the controls in the main control room is to allow the operator to operate the plant safely under normal conditions, and to maintain it in a safe condition under accident conditions. The types of controls in the main control room include both discrete (dedicated) control switches and soft controls. The discrete control switches are controls dedicated to a single function, with each switch having a single action. As shown in Figure 18.8-1, the soft control units are control devices whose resulting actions are selectable by the operator. The instrumentation and control architecture uses both discrete control switches and soft control units. The soft control units are used to provide a compact alternative to the traditional control board switches by substituting virtual switches in the place of the discrete switches. The final configuration of these elements is dependent upon the results of the human system interface design process described in subsection 18.8.1 below.

tasks to be supported by the control center or human system interface resource. Performance requirements represent high level design goals and help to clarify the functional designer's intent. They are high level requirements that may not be readily verifiable by testing or other quantitative means, but are important considerations for meeting the goals defined in the mission statements. The design bases establish the foundation for the design and the rationale behind engineering decisions made and criteria established for the design. Functional requirements include requirements needed to meet the criteria defined in the applicable codes, standards, and customer requirements. The functional requirement documents include requirements to meet failure, diversity, electrical separation, and other applicable criteria; they establish requirements related to access control, redundancy, independence, identification and test capability; they define requirements on system inputs and outputs; they specify the system safety classification and define applicable quality assurance, reliability goals, and environmental qualification requirements. The functional requirements document for each human system interface resource includes a specification of the cognitive activities in the operator decision-making model that the human system interface resource is intended to support.

18.8.1.2 Design Guidelines

Guidelines for the human system interface design are developed for each of the human system interface resources to facilitate the standard and consistent application of human factors engineering (HFE) principles to the design. This guidance is contained in a set of standards and conventions guidelines documents that tailor generic human factors engineering guidance to the AP600 human system interface design and define how those human factors engineering principles are applied.

These guidelines become a tool that enables groups of people to simultaneously develop the human system interface in a consistent manner in accordance with the human factors engineering principles established for the design. The guidelines are used to perform the human factors engineering design verification activity of the human factors verification and validation plan (Reference 24).

Human system interface design guideline documents include:

- Anthropometric guidelines
- Alarm guidelines
- Display guidelines
- Controls guidelines
- ~~Computerized~~ procedure~~s~~ system guidelines

AP600 human system interface guideline documents provide:

- Statements of their intended scope, references to source materials, and instructions for their proper use

System

~~computerized~~ procedure~~s~~) addressing those human performance issues. The man-in-the-loop tests examine human performance issues associated with individual human system interface resources, as well as concept tests that examine the coordination of multiple human system interface resources. Tests are narrowly focused in order to enable testing to begin early in the human system interface design process.

The prototype tests employ individuals with operational experience as test participants. Data collected includes objective operator performance data (performance time; errors) as well as subjective judgments of operators (responses on rating forms; elicited comments and suggestions).

Descriptions of the set of tests are provided in Reference 9. The performance and results of each concept test is documented and, as shown in Figure 18.2-3, used as feedback to the elements and activities of the human factors engineering and human system interface design process.

18.8.1.5 Main Control Room Mockup

A full-scale mockup of the main control room working area, including main control consoles (workstations) and the wall panel display, is constructed as part of the human system interface design process. The mockup consists mainly of non-operational representations of the desks, displays, and panels. The mockups are constructed to the anthropometric profiles and arranged in the floor layout intended for the main control room.

The main control room mockup is used to examine and verify physical layout aspects such as availability of workspace, physical access, visibility, and related anthropometric and human factors engineering issues. It will also be used for walk-through exercises to examine issues such as staffing levels, task allocation, and procedure usage.

18.8.1.6 Human System Interface Design Documentation

The human system interface design is documented through a system specification document for the operation and control centers, functional requirement documents, design criteria documents, design review documents, and documentation of design configuration change control.

18.8.1.7 Task-Related Human System Interface Requirements

As shown in Figure 18.2-3, the results of other human factors engineering program elements are used as input and bases for developing the operation and control center system and human system interface resources functional design (mission statements, performance requirements, design basis, functional requirements), guideline documents and the design specification documents. Staffing assumptions, operating experience reviews, functional requirements analysis and allocations, task analysis, and integration of human reliability analysis provide the bases for identifying the human system interface requirements needed to support human functions and tasks. The resulting human system interface requirements are documented in the human system interface resource functional design documents (operation and control centers system specification document and the individual human system interface resource functional requirements document), guidelines document and design specification documents. Subsections 18.8.1.1 through 18.8.1.3 provide descriptions of these documents.

The AP600 task analysis, described in Section 18.5, includes two complimentary activities: function-based task analysis (FBTA) and traditional task analysis, or operational sequence analysis (OSA). The function-based task analysis identifies the indications, parameters, and controls that the operator needs to make decisions about the respective function. There is also a verification that the indications and controls identified in the process analysis are included in the design. The operational sequence analysis, completed as part of the task analysis process, focuses on specifying the operational requirements for the complete set of tasks selected. One of the guidelines used in selecting tasks for analysis are those tasks that represent the full range of activities in the AP600 emergency response guidelines. One type of information provided by the operational sequence analysis is an inventory of alarms, controls, and parameters needed to perform the task sequences. The operational task analysis results include the identification of controls, alarms, and parameters needed by the operator to execute task sequences found within the emergency response guidelines. These results serve as a cross-check with the function-based task analysis results. Design reviews held during the human system interface design serve as another means of verifying completeness and identifying and correcting omissions. The task support verification activity of the human factors verification and validation (Reference 24) verifies that the human system interface design provides the necessary alarms, displays, and controls to support personnel tasks.

The collective results of the task analysis activities identify the tasks and operational information needed by the operator to execute these tasks. For each display, a display task description is written. The display task description includes the identification of the informational needs to be supported by the display. The features, dynamic characteristics, calculated values, and supporting algorithms for the display are part of the display task description. The design specification of a display includes the range, precision, and measurement units of the parameters provided in the display. These parametric characteristics are chosen to support the task and the operator informational needs. The parametric characteristics, identified in the design specification, are provided using the guidelines presented in the design guidelines document for displays. The basis for the parametric characteristics chosen for the displays is found in the design guidelines document.

18.8.1.8 General Human System Interface Design Feature Selection

system

The AP600 human system interface resources include the wall panel information system, alarm system, plant information system, ~~computerized~~ procedure ~~system~~, controls, (soft and dedicated) and the qualified data processing system. These human system interface resources are used as a starting point to define how the human system interface supports operator performance. Reference 25 describes the operator decision-making model that is used by the task analysis activities to identify the operator's information and control requirements. The human system interface resources are mapped to the major classes of operator activities identified from this model. Figure 18.8-3 and a description in Section 5.0 of Reference 45

As noted in Section 4.1.a of Reference 27 "...the principle purpose and function of the Safety Parameter Display System is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident." Since the main intended use is during relatively rare occurrences, human-factors engineering suggests that the operators will find that the use of data acquisition habits acquired and repeated during the normal operation of the plant will be the most successful. A system in the control room that only varies its output during abnormalities may require a shift in mental focus and in data acquisition habits and subsequent analysis. An effective means for conveying the safety state of the plant is to provide data and displays for normal operation that employs the Safety Parameter Display System required principles for data synthesis, concentration and display. This operator interface is operational over the range of plant conditions specified by the Safety Parameter Display System requirements, as well as during normal operations.

The operator-interface to the plant is improved by integrating Safety Parameter Display System requirements into the overall human system interface design to avoid the need for another system that is infrequently used.

The following subsections describe the approach to meeting the regulatory requirements for an Safety Parameter Display System by addressing the Safety Parameter Display System requirements of References 26 and 27.

18.8.2.1 General Safety Parameter Display System Requirements

system

The AP600 human system interface resources used to address the Safety Parameter Display System requirements are the alarm system, plant information system (workstation visual display unit displays), and the ~~computerized~~ procedure ~~system~~. The AP600 human system interface data display (alarms and visual display unit displays) is organized around the Safety Parameter Display System requirement of plant process functions. Expressing plant state in terms of process functions is incorporated in the AP600 control room design. This is expected to improve the human interface by making the data presentation interface seamless as the plant moves from one operational state to another.

An alarm system which organizes the presentation of alarms by process function and adapts a "dark board" approach (for all plant modes) continually indicates the state of each of the functions. By remaining dark when the process is performing as expected, the process functions are interpreted as being satisfied. An alarm indication displayed in any function indicates that the function is in jeopardy. In this way, the set of alarms that is active is the minimum set. The alarm system is capable of displaying a full range of alarms based on important plant parameters and data trends. The alarms indicate when process limits are being approached and exceeded.

Section 18.7 and Reference 23 present an implementation plan for integrating the human reliability analysis with human factors engineering. The critical human actions and the risk important tasks identified through the execution of this plan are used as an input to the task analysis activities and subsequently to the design of the human system interface. They are also used to evaluate the Safety Parameter Display System functions and parameters selected to monitor these functions. The human system interface, which includes the integration of Safety Parameter Display System requirements, is designed to reduce the likelihood of operator error and provide for error detection and recovery capability for the identified critical human actions and risk important tasks.

18.8.2.2 Display of Safety Parameters

The functionally organized plant information system displays, including the Safety Parameter Display System-related displays, are accessed on the workstation visual display units (VDU) using a cursor. The AP600 operator workstations employ a windowing system which allows a single cursor to cover the visual display unit screens. The design allows the operator to recover a specific parameter within one or two actuations of the pointing device.

The design goal for the AP600 human system interface is to update the displays every 1 to 2 seconds. The process data sampling rate is 1 second or less. Sequence of events (SOE) points can be sampled at a rate of once every milli-second and are available within the AP600 human system interface. The Safety Parameter Display System responds to user commands in less than 10 seconds. The design goal for graphical display response time, from user command to developed graphical display, in the AP600 human system interface is 2 seconds.

The AP600 alarm system includes plant overview alarms that are organized around the concept of plant process functions. These process functions address the five SPDS functions. The alarm system overviews, including the functional organization, are integrated into the wall panel information system displays.

During the execution of emergency operating procedures, the ~~computerized~~ procedure ~~system~~ SYSTEM provides a continuous display of the status of each critical safety function.

The Safety Parameter Display System data and data display organization are available to the control room staff.

The AP600 human system interface process display set (from the plant information system) is organized into two hierarchies that are linked together. One is focused upon providing the process data from a functional perspective and the other from a physical perspective. Both follow the concept of abstraction/aggregation suggested by Rasmussen as described in Reference 25. Top levels in the hierarchy are plant wide summaries, lower levels are component details. The hierarchy is structured so as to reflect the plant process functional decomposition performed during the function based task analysis described in Reference 25.

Process display presentation for the control room users is organized by functions. The function based task analysis integrates the functional organization design principles dictated by the Safety Parameter Display System requirements into the AP600 human system interface.

the operator's decision-making process, and promotes the interaction with other plant personnel, while preventing distractions by non-operating personnel. The main control room provides the interfacing resources between the operation of the plant and the maintenance of the plant. Its areas include the main control area, the switching and tagging area, the shift supervisor's office, the shift supervisor's clerk's office, and the operations staff's area (see Figure 1.2-8). Habitability systems are described in Sections 6.4 and 9.4.

18.8.3.2 Main Control Area Mission and Major Tasks

two

The mission of the main control area is to provide the support facilities necessary for the operators to monitor and control the AP600 efficiently and reliably. Figure 6.4-1 provides a view of the main control area. The main control area includes ~~the~~ reactor operator workstations ~~(Workstations A and B)~~, the supervisor's workstation, the dedicated safety panel and ~~the panel associated with~~ the wall panel information system. The layout, size and ergonomics of the operator workstations and the wall panel information system depicted in this figure does not reflect the results of the human system interface design implementation plan described in subsection 18.8.1. The actual size, shape, ergonomics and layout of the operator workstations and the wall panel information system is an output of the implementation plan.

The major task of the main control area is to provide the human system interface resources that determine the plant state and implement the desired changes to the plant state during both normal and emergency plant operations. The main control area provides alarms to alert the operator to the need for further investigation. Plant process data displays permit the operator to observe abnormal conditions and identify the plant state. The controls enable the operator to execute actions. The process data displays and the alarms provide feedback to enable the operator to observe the effects of the control actions.

Each of the ~~R~~eactor operator ~~W~~orkstations ~~A and B~~ contain the displays and controls to start up the plant, maneuver the plant, and shut down the plant. Reference 44 presents input from the designer to the Combined License applicant for the determination of the staffing level of the operating crew in the main control room. Each workstation is designed to be manned by one operator. There is sufficient space and operator interface devices for two operators. The physical makeup of ~~Workstations A and B~~ the two reactor operator workstations is identical. The human system interface resources available at each workstation are:

- Plant information system displays
- Control displays (soft controls)
- Alarm system support displays
- ~~Computerized~~ system procedure ~~displays~~
- Screen and component selector controls

The supervisor workstation is identical to ~~Workstations A and B~~ the reactor operator workstations, except that its controls are locked-out. The supervisor workstation contains both internal plant and external plant communications systems.

Detection and Monitoring / Situation Awareness

Alert
- Alarm System
- WPIS

Observe
- WPIS
- Plant Information System
- QDPS

Identify State
- WPIS
- Alarm System
- Plant Information System
- QDPS
- ~~Computerized Procedures~~

System

Interpretation / Planning

Implications of State
- Plant Information System
- ~~Computerized Procedures~~

Goal Selection
- ~~Computerized Procedures~~
- Plant Information System
- Alarm System

Plan Success Path
- ~~Computerized Procedures~~
- Plant Information System
- WPIS

Select/ Formulate Actions
- ~~Computerized Procedures~~
- Plant Information System

System

Control

Execute Actions
- Soft Controls
- Dedicated Controls

Feedback

Monitor Goal Achievement
- Alarm System
- WPIS
- Plant Information System
- QDPS

Monitor State
- WPIS
- Alarm System
- Plant Information System
- QDPS

Verify Action
- Soft Controls
- Dedicated Controls

Figure 18.8-3

Mapping Human System Interface Resources to Operator Model

of

Decision-Making

18.9 Procedure Development

WCAP-14690, "Designer's Input to Procedure Development for the AP600" (Reference 1), provides input to the Combined License applicant for the development of plant operating procedures, including information on the development and design of the AP600 emergency response guidelines and emergency operating procedures. This WCAP is provided as input for the Combined Licensed applicant. The WCAP also includes information on the ~~computerized~~ procedure ~~system,~~ which is the human system interface through which operators execute the plant procedures.

system

18.9.1 Combined License Information

See Section 13.5 for a discussion of the responsibility for procedure development.

18.9.2 References

1. WCAP-14690, "Designer's Input to Procedure Development for the AP600," Revision 0, June 1996.

18.11 Human System Interface Design Test Program

This section describes ~~the plan for~~ the AP600 human system~~s~~ interface design test program. This test program consists of two distinct parts:

- Concept tests to be performed as part of the human system interface design process (as described in subsection 18.8.1).

- Verification and validation (V&V) tests to be performed at the completion of the AP600 design process.

The goal of the human system interface design test program is to systematically evaluate human factors concerns that affect plant performance and incorporate the results into the design of the human system interface.

Plant facilities and plant staff activities are addressed in the AP600 human system interface design ~~factors test program for the human system interface.~~ test program. Facilities included in the scope of ~~the AP600~~ this test program are the main control room, technical support center, the remote shutdown facility, and local control stations. Staff activities included are those activities required to operate under normal, abnormal, and emergency conditions.

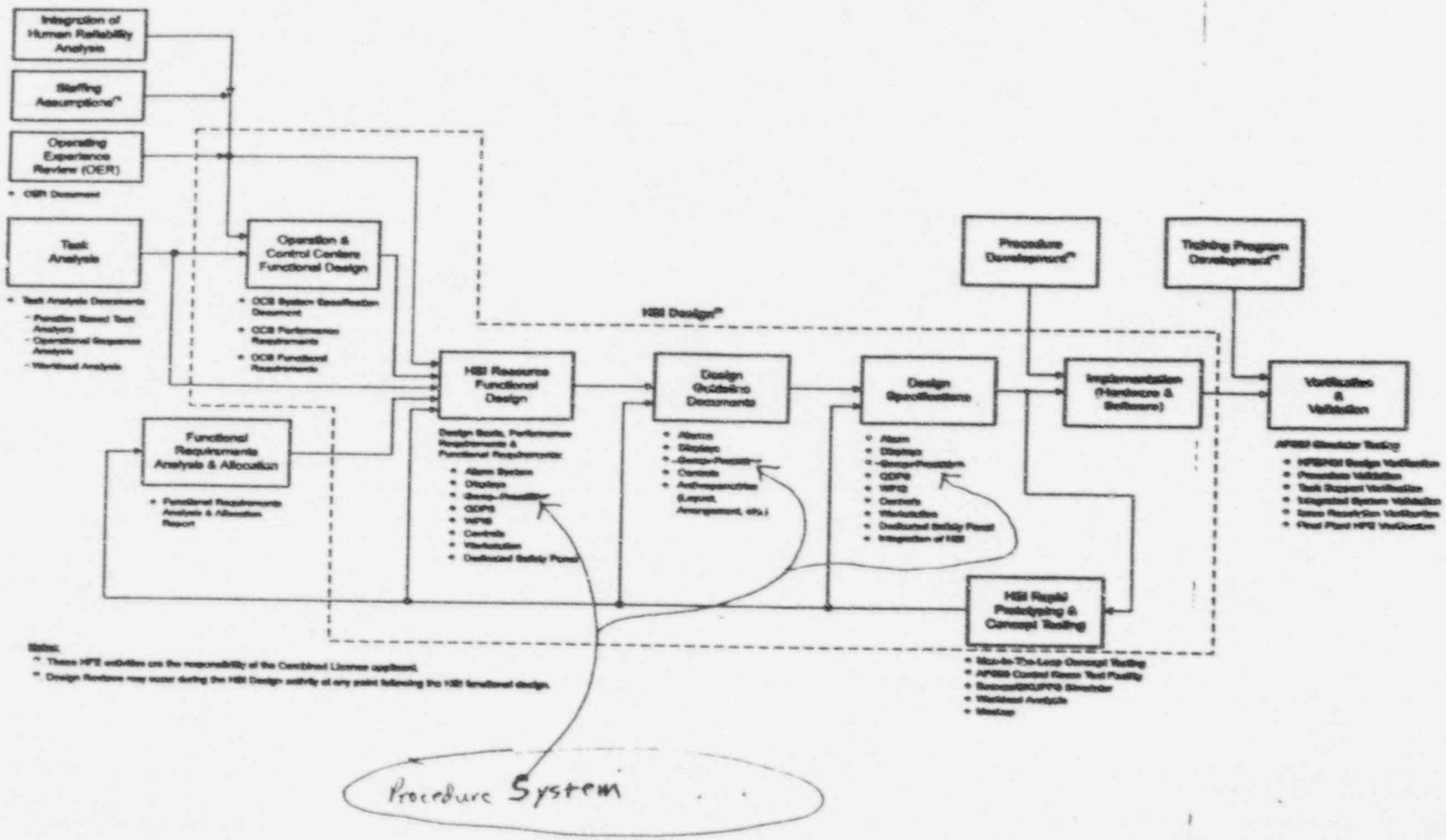
The ~~test program for the~~ AP600 human system interface (design test program) focuses on the following human system interface resources:

- Plant information system (including functional and physical displays of plant processes)
- Alarm system
- ~~Computerized procedure system~~ System
- Dedicated and soft (computer based) controls
- Wall panel information system
- Qualified data processing system

As illustrated in Figure 18.11-1, a two phase process is used to define the (human system interface design) test program. Phase 1 is called issue definition. Its purpose is to integrate major operator activities with the human system interface resources in order to establish a set of human performance evaluation issues. Phase 2 addresses test development. The purpose of this phase is to develop testing plans for each of the evaluation issues identified in Phase 1. Reference 1 presents a description of the methodology, analysis, and the results of executing these two phases. The results include the identification of 17 human performance evaluation issues and a description of the testing approach and requirements for addressing each of the evaluation issues. The 17 human performance evaluation issues are listed in Table 18.11-1.

The 17 human performance issues are organized under five headings:

- Evaluations for detection and monitoring
- Evaluations for interpretation and planning



Notes:
* These HFE activities are the responsibility of the Combined License applicant.
** Design Reviews may occur during the HSI Design activity at any point following the HSI functional design.

Figure 18.2-3

Overview of the AP600 Human Factors Engineering Process

Energy Center Site
4350 Northern Pike
Monroeville, PA 15146

Westinghouse
AP600 Design Certification Project

Fax

To:	Joe Sebrosky	**From:**	Gene Piplica
Fax:	8-1-301-415-2002	**Pages:**	2
Phone:	301-415-1132	**Date:**	May 13, 1997
Re:	[Click **here** and type subject of fax]		

X **Urgent** ☐ **For Review** ☐ **Please Comment** ☐ **Please Reply** ☐ **Please Recycle**

• **Comments:**

Reference: DCP/NRC0857, dated 5/9/97, B. A. Mcintyre to T. R. Quay, "Westinghouse Responses to NRC Followon Questions Regarding the AP600 Initial Test Program (ITP)"

Joe –

Transmitted via this fax is additional page to Enclosure 3 (proposed SSAR changes to Appendix 1A) of referenced transmittal letter which was inadvertently not included in original transmittal.

Criteria Section	Referenced Criteria	AP600 Position	Clarification/Summary Description of Exceptions
	App. A.1.h	Conforms	The characteristics of the AP600 passive safety systems allow the support systems such as the cooling water systems, the heating, ventilating, and air conditioning and the ac power sources to be nonsafety-related and simplified. The capability of these systems is established by testing. Cold water interlocks are not a design feature of the AP600.
	App. A.1.i	Conforms	The AP600 has no secondary containment. Therefore, this guideline applies only to primary containment. The following systems or functions are not design features of the AP600 and are therefore not tested: • Containment and containment annulus vacuum breaker • Containment supplementary leak collection • Standby gas treatment • Secondary containment system • Containment annulus and cleanup • Bypass leakage tests on pressure suppression • Ice condenser systems • Containment penetration cooling
	App. A.1.j	Conforms	Recirculation flow control, traversing incore probes, automatic dispatching control systems and hotwell level control are not design features of the AP600.
	App. A.1.k	Conforms	
	App. A.1.l	Conforms	Condenser off gas systems are not a design feature of the AP600
	App. A.1.m	Conforms	
	App. A.1.n *(except A.1.n(14)(f))*	Conforms	Seal water boron recovery, shield cooling, refueling water storage tank heating, and equipment for establishing and maintaining subatmospheric pressures are not design features of the AP600.
	App. A.1.n(14)(f)	*Exception*	*A long term demonstration test of the MCR habitability system to m[illegible] the control environment is acceptable for the first plant only provided that the as-built information and design basis heat loads used as the assumptions in the heat sink capacity analysis for the MCR do not change for subsequent plants.*

FAX to DINO SCALETTI

May 9, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #750 (DSER 3.8.4.4-2)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #750 (DSER 3.8.4.4-2) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5070 on April 15, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We believe the information provided herewith has completed our action on Item #750. We request that NRC review the material we have provided concerning Item #750 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

1 of 6

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 750 And 750 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
750	NRR/ECGB	3.8.4.4-2	DSER-OI		Orr/ANSALDO	Closed	Action W	NSD-NRC-97-5070	

Westinghouse should provide for staff review the final design calculation for the shield building and the passive containment cooling water storage tank.

Methodology was presented to NRC in meeting on March 2. The final design calculation for the shield building and the passive containment cooling water storage tank were reviewed during the meeting in June 1995. Westinghouse should address comments identified in meeting notes.
Comments from meeting have been addressed and were discussed during meeting on March 7, 1996.
Closed: remaining issues are tracked under new RAI 230.100 transmitted by letter of April 5, 1996.
NRC to review design calculations during meeting in December.
NRC Status: Action W - Westinghouse will provide the design of the ring beam including torsional moment and axial tension. (12/16/96)
Action W - During the design calculation review conducted on December 9 through 13, 1996, the staff identified that the torsional moment due to the combined loads applied on the tension ring beam is significantly lower than that obtained from the staff's confirmatory analysis. Westinghouse should either justify the adequacy of the originally calculated torsional moment or perform a new analysis to calculate the torsional moment for the design.
Closed - Response provided by NSD-NRC-97-5070 of 4/15/97. jww

2de

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230 0355

NSD-NRC-97-5070
DCP/NRC0815
Docket No.: STN-52-003

April 15, 1997

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

ATTENTION: T. R. Quay

SUBJECT: SEISMIC AND STRUCTURAL OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR Sections 3.7 and 3.8. These responses are intended to resolve the identified open items related to this section.

The items addressed include the following:

OITS #	DSER Item or other	W Status
750	3.8.4.4-2	Closed
766	3.8.5-8	Audit N
5028	December, 1996 Meeting Item	Closed
5030	December, 1996 Meeting Item	Closed
5031	December, 1996 Meeting Item	Confirm W
5032	December, 1996 Meeting Item	Closed

The Westinghouse Status for the open item tracking system is noted for each of the items above. The NRC staff is requested to review the responses and confirm the status.

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

D. A. Lindgren / for
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml
Attachment

cc: D. Jackson, NRC (w/Attachment)
F. Fanous, Ames Lab (w/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

3-26

1194A

Open Item # 750 DSER Open Item 3.8.4.4-2

The open issues are summarized in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design" as follows:

OI # 750 "Westinghouse will provide the design of the ring beam including torsional moment and axial tension."

Item # 9 American Concrete Institute (ACI) Code Version for Torsional Design of Reinforced Concrete Members

The staff indicated that torsional design provisions in ACI 318-95 incorporate some significant recent developments and should be used. In response, Westinghouse expressed a concern that the use of different year versions of the same code for different design issues may be confusing and undesirable from consistency and quality control considerations. However, Westinghouse agreed to evaluate the practicality of incorporating ACI 318-95 torsional design provisions in AP600 structure design criteria.

Westinghouse response

Reference 1: Ames Laboratory, "3D Analysis of AP600 Standard Plant Shield Building Roof" Presentation 12/13/97, (attached to NRC letter of March 4, 1997)

Ames Lab presented resultant forces in the "tension ring" during the NRC/Westinghouse meeting in December, 1996. Table 1 shows a comparison of the Westinghouse results against those presented by Ames Lab during the meeting. These forces use the notation of the Ames Lab report and are the maximum at any location around the circumference.

Westinghouse has performed additional investigation subsequent to the meeting in December, 1996 using a more refined model. This investigation shows that the differences in the Ames Lab and Westinghouse results are due to the increased refinement of the Ames Lab model in the vicinity of the air inlet openings. While this additional refinement will be used in future analyses of the shield building roof, the higher magnitudes of moments in the Ames Lab results have little influence on the reinforcement required relative to that designed using the results from the Westinghouse analyses.

The design of the reinforcement for the tension ring is available for NRC review. The design is in accordance with ACI 349 and the methodology is summarized herein. The reinforcement for the shield building roof is designed for each section of the shield building roof. The tension ring is one of the sections. Design of the reinforcement uses the maximum member forces from the finite element analyses at any elevation or azimuth within the segment.

Hoop reinforcement for axial and flexural forces (A_t , A'_t)

The required hoop reinforcement considers unit height of the tension ring subject to hoop tension and bending about the vertical axis. Two cases are considered. One case uses the maximum hoop tension per unit height anywhere around the circumference (which occurs at the bottom of the tension ring above the center of an air inlet) plus the moment at the same location. The second case uses the maximum hoop compression and associated moment. The hoop reinforcement required for these cases is increased for in-plane shear and torsion loads as described below.

4 of 6

In-plane shear reinforcement (A_{xy})

The required reinfoıcement for in-plane shear is calculated using the maximum in-plane shear force which occurs above the edge of the air inlet opening. This required reinforcement is added to that required for flexure and may be provided on either face.

Out-of-plane shear reinforcement (A_v)

The required reinforcement for out-of-plane shear is calculated using the maximum out-of-plane shear force. Reinforcement is provided for the shear force exceeding V_c, the capacity of the concrete in shear. This reinforcement is provided as stirrups in the tension ring.

Torsion reinforcement (A_t)

The required reinforcement for torsion is calculated using the maximum torsional moment (M_{xy}) about the horizontal axis of the section at any location around the circumference. The required reinforcement is calculated as:

$$A_s = \frac{M_{xy}}{\phi \, f_y \, (d-d')}$$

The reinforcement is calculated using the maximum torsion per unit height around the circumference. In the tension ring longitudinal reinforcement is added for this torsional moment and stirrups are provided above the air inlet openings.

The ACI 349 Code permits torsion and out-of-plane shear reinforcement to be calculated at distance, d, from the face of the support. In the current reinforcement design based on the finite element analyses with a relatively coarse mesh, the design was based on the maximum forces anywhere around the circumference. In future analyses with a more refined mesh such as that used in the Ames Lab analyses the design will be based on out-of-plane shear and torsion at distance, d, from the support.

The principal differences in the torsional design provisions of ACI 318-95 relative to those of earlier codes are described in the commentary to ACI 318-95, paragraph 11.6. "In the 1995 code, the former elliptical interaction between the shear carried by concrete, V_c, and the torsion carried by the concrete has been eliminated. V_c remains constant at the value it has when there is no torsion, and the torsion carried by the concrete is always taken as zero." This approach is taken in the design of the out-of-plane shear and torsion reinforcement for the tension ring.

Meridional reinforcement

The calculation of the required meridional reinforcement is similar to that for the hoop reinforcement but uses the vertical membrane force and bending about the horizontal axis. In the tension ring itself the design loads are primarily axial and the moments are small since the overall moment is carried by the composite section including the webs of the precast panels. Thus vertical reinforcement between the air inlet openings is established by development of the reinforcement from the columns below the tension ring and the conical roof above the tension ring. Vertical reinforcement above the air inlets is established by the stirrups required for out-of-plane shear and torsion plus development of the reinforcement in the conical roof above the tension ring.

5db

2

Table 1 Member forces in tension ring

Force	N (kips)	Qz (kips)	Qx (kips)	Mx (k.ft)	T (k.ft)	Mz (k.ft)
Ames	2524	512	144	856	904	785
W	2554	755	252	323	380	454

Open Item # 766 DSER Open Item 3.8.5-8

The open issue is summarized in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 7, as follows:

"Validation of Initec's Post-Processing Computer Codes

The staff could not review the validation of these codes since the validation package provided for review was mostly in Spanish. Westinghouse agreed to provide an English version of the package for the staff to review."

Westinghouse response

An english version of the package is available for staff review

Open Item # 5028

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 4, as follows:

"Consideration of Out-of-Plane Seismic Load in the Design of Wall Panels

Out-of-plane seismic loads on the wall panels are caused by the inertia of the panel mass (local) as well as by the global behavior of the wall-floor system (which may cause panel support displacements resulting in out-of-plane moments and shears). But in designing the wall panels, Initec considered only the local out-of-plane bending and did not consider the global out-of-plane moments and shears that were calculated by Bechtel.

The staff discussed this with Initec, Westinghouse, and Bechtel personnel as a generic issue; it was agreed that the appropriate way of combining the local and global moments and shears and its impact in the design will be addressed by Westinghouse."

Westinghouse response

The out-of-plane seismic loads on the wall panels caused by the inertia of the panel mass have been considered in the design of the walls using hand calculations based on their mass and acceleration. This was done because generally there is only one element between floors. The out-of-plane moments shown in the finite element analysis results are primarily due to panel support displacements.

****
*** TX REPORT ***
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TRANSMISSION OK

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SUBADDRESS	
CONNECTION ID	
ST. TIME	05/14 09:11
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PGS.	6
RESULT	OK

FAX to DINO SCALETTI

May 9, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #750 (DSER 3.8.4.4-2)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #750 (DSER 3.8.4.4-2) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5070 on April 15, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We believe the information provided herewith has completed our action on Item #750. We request that NRC review the material we have provided concerning Item #750 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

FAX TO BILL HUFFMAN
May 14, 1997

Here is the information we provided for OITS item #2277:

A stuck open LTOP valve will result in reactor coolant leakage to the floor of a steam generator compartment. This leakage then goes through a floor drain to the containment sump. Abnormal operation procedures should require identification and correction of coolant leaks long before reaching a condition required for ERG implementation. If, however, the leakage continues long enough, system levels will drop enough to require initiation of the actions defined by SDG-1 of the ERGs as shown on Shutdown Safety Status Tree SDF-0.1. Step 6 in the step sequence of SDG-1 requires the operators to "Identify and Isolate any RCS Leakage."
The LTOP valves are addressed in subsection 2.3.6 of the ITAAC in Table 2.3.6-4 in Item 7.a). This item covers testing of the low temperature overpressure system, including the relief valves.

Jim

Jim Winters
412-374-5290

FAX to DINO SCALETTI

May 13, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Robert Tupper
Robin Nydes
Bob Vijuk
Brian McIntyre

OPEN ITEM #4117 (DSER 3.2.1)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages ~~for~~ open items that we believe are complete. Relevant documentation related to Open Item #4117 (DSER 3.2.1) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-4993 on February 21, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We request that NRC review the material we have provided concerning Item #4117 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 4117 And 4117 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
4117	NRR/HQMB	3.2.1	RAI-OI		RTNSS/Kloes	Closed	Action W	NSD-NRC-97-4993	

RAI# 260.84 Explain how quality assurance requirements for the regulatory treatment of nonsafety systems, systems, and components (RTNSS) which, Westinghouse has defined in Letter NSD-NRC-96-4670, dated March 26, 1996, are also sufficient to satisfy the regulatory requirements for seismic Category II, as described in RG 1.29, i.e., "all activities affecting the safety-related functions of those portions of structures, systems, and components covered under Regulatory Positions 2 and 3" of the RG?

Closed - See letter NSD-NRC-97-4993 dated 2/21/97.

2805

 Total Records: 1

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4993
DPC/NRC0747
Docket No.: STN-52-003

February 21, 1997

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

TO: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600.

Dear Mr. Quay:

Enclosed are three copies of the Westinghouse responses to open items on AP600 topics. Responses to nine RAIs are included in this transmittal. RAI 410.261 provides information on Section 9 of the SSAR. Responses to RAI 440.571, Revision 1, discusses the OSU Test Analysis Report. Responses to RAIs 260.83, 84, 85, 86, 87, 88, and 89 address questions on Section 3 of the SSAR.

The NRC technical staff should review these responses as a part of their review of the AP600 design. These responses close, from a Westinghouse perspective, the addressed questions. The NRC should inform Westinghouse of the status to be designated in the "NRC Status" column of the OITS.

Please contact Brian A. McIntyre on (412) 374-4334 if you have any questions concerning this transmittal.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Enclosures

cc: T. Kenyon, NRC - (w/o enclosures)
W. Huffman, NRC - (w/enclosures)
N. Liparulo, Westinghouse - (w/o enclosures)

XX90A

ENCLOSURE TO WESTINGHOUSE
LETTER NSD-NRC-97-4993

FEBRUARY 21,1997

1 of 5

X90A

Question 260.84

Re: OITS 4117

Explain how quality assurance requirements for the regulatory treatment of nonsafety systems, systems, and components (RTNSS) which, Westinghouse has defined in Letter NSD-NRC-96-4670, dated March 26, 1996, are also sufficient to satisfy the regulatory requirements for seismic Category II, as described in RG 1.29, i.e., "all activities affecting the safety-related functions of those portions of structures, systems, and components covered under Regulatory Positions 2 and 3" of the RG?

Response:

SSAR Subsection 3.2.1.1.2 contains the Westinghouse position that "the quality assurance requirements for Seismic Category II structures, systems and components are sufficient to provide that these components will meet the requirement to not cause unacceptable structural failure of or interaction with Seismic Category I structures." The quality assurance requirements defined in Letter NSD-NRC-96-4670 constitutes a Quality Assurance Program Plan used in a procurement contract to assure that the component will be designed, documented and delivered to the plant in a controlled manner. Design input, including seismic design requirements for Category II, shall be contained in design requirements documents or Equipment Specifications which are also part of a procurement contract.

SSAR Revision: NONE

FAX to DINO SCALETTI

May 13, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Robin Nydes
Bob Vijuk
Brian McIntyre

OPEN ITEM #4118 (DSER 3.2.1)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #4118 (DSER 3.2.1) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-4993 on February 21, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We request that NRC review the material we have provided concerning Item #4118 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 4118 And 4118 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
4118	NRR/HQMB	3.2.1	RAI-OI	RAI# 260.85 Please identify all RTNSS SSCs that would also satisfy the functional and design criteria of those portions of structures, systems, and components covered under Regulatory Positions 2 and 3 of RG 1.29. Response provided in NSD-NRC-97-4993 on 2/21	RTNSS/Tupper	Closed	Action W	NSD-NRC-97-4993	

Total Records: 1

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4993
DPC/NRC0747
Docket No.: STN-52-003

February 21, 1997

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

TO: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600.

Dear Mr. Quay:

Enclosed are three copies of the Westinghouse responses to open items on AP600 topics. Responses to nine RAIs are included in this transmittal. RAI 410.261 provides information on Section 9 of the SSAR. Responses to RAI 440.571, Revision 1, discusses the OSU Test Analysis Report. Responses to RAIs 260.83, 84, 85, 86, 87, 88, and 89 address questions on Section 3 of the SSAR.

The NRC technical staff should review these responses as a part of their review of the AP600 design. These responses close, from a Westinghouse perspective, the addressed questions. The NRC should inform Westinghouse of the status to be designated in the "NRC Status" column of the OITS.

Please contact Brian A. McIntyre on (412) 374-4334 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Enclosures

cc: T. Kenyon, NRC - (w/o enclosures)
W. Huffman, NRC - (w/enclosures)
N. Liparulo, Westinghouse - (w/o enclosures)

3090A

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ENCLOSURE TO WESTINGHOUSE
LETTER NSD-NRC-97-4993

FEBRUARY 21,1997

4 of 5

3090A

Question 260.85

Re: OITS 4118

Please identify all RTNSS SSCs that would also satisfy the functional and design criteria of those portions of structures, systems, and components covered under Regulatory Positions 2 and 3 of RG 1.29.

Response:

RG 1.29 Regulatory Position 2 imposes Category I design requirements and 10 CFR 50 Appendix B on "portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature" included in Category I. Regulatory Position 3 imposes Category I design requirements and 10 CFR 50 Appendix B on "portions of structures, systems, or components that form interfaces between Seismic Category I and non-Seismic Category I requirements."

AP600 does not have any identified RTNSS structures, systems, and components that fall under the definitions of Regulatory Position 2 or 3.

SSAR Revision: NONE

FAX to DINO SCALETTI

May 13, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Robert Tupper
Robin Nydes
Bob Vijuk
Brian McIntyre

OPEN ITEM #4119 (DSER 3.2.1)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #4119 (DSER 3.2.1) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-4993 on February 21, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We request that NRC review the material we have provided concerning Item #4119 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 4119 And 4119 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
4119	NRR/HQMB	3.2.1	RAI-OI		RTNSS/Tupper	Closed	Action W	NSD-NRC-97-4993	
				RAI# 260.86 How would RTNSS QA requirements as defined in NSD-NRC-96-4670 address interface design requirements identified in RP C.3.?					
				Response provided in NSD-NRC-97-4993 on 2/21					

2 of 5

 Total Records: 1

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4993
DPC/NRC0747
Docket No.: STN-52-003

February 21, 1997

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

TO: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600.

Dear Mr. Quay:

Enclosed are three copies of the Westinghouse responses to open items on AP600 topics. Responses to nine RAIs are included in this transmittal. RAI 410.261 provides information on Section 9 of the SSAR. Responses to RAI 440.571, Revision 1, discusses the OSU Test Analysis Report. Responses to RAIs 260.83, 84, 85, 86, 87, 88, and 89 address questions on Section 3 of the SSAR.

The NRC technical staff should review these responses as a part of their review of the AP600 design. These responses close, from a Westinghouse perspective, the addressed questions. The NRC should inform Westinghouse of the status to be designated in the "NRC Status" column of the OITS.

Please contact Brian A. McIntyre on (412) 374-4334 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Enclosures

cc: T. Kenyon, NRC - (w/o enclosures)
W. Huffman, NRC - (w/enclosures)
N. Liparulo, Westinghouse - (w/o enclosures)

3 of 5

3090A

ENCLOSURE TO WESTINGHOUSE
LETTER NSD-NRC-97-4993

FEBRUARY 21,1997

4 of 5

Question 260.86

Re: OITS 4119

How would RTNSS QA requirements as defined in NSD-NRC-96-4670 address interface design requirements identified in RP C.3.?

Response:

As indicated in the response to RAI 260.85, AP600 does not have any identified RTNSS structures, systems and components that "form interfaces between Seismic Category I and non-Seismic Category I requirements" identified in Reg. Guide 1.29, RP 3. QA requirements defined in NSD-NRC-96-4670 do not apply to any components in the Regulatory Position C.3 category.

SSAR Revision: NONE

FAX to DINO SCALETTI

May 8, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #628 (DSER 3.7.1.1-1)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 22 calendar days away (17 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #628 (DSER 3.7.1.1-) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5041 on March 26, 1997. The material from the letter has been incorporated into Revision 12 of the SSAR. A copy of this letter, with the pertinent attachments, and copies of the pages from the SSAR are included with this fax. We believe the information provided herewith has completed our action on Item #628 and that it should be closed separately from Items 5234, 5235 and 5236 which now are included in the Open Item Tracking System for Item 628. We request that NRC review the material we have provided concerning Item 628 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 628 And 628 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
628	NRR/ECGB	3.7.1.1-1	MTG-OI		Orr / NRCSEIS/SSARR	Action W	Action W	NSD-NRC-97-4956	

Westinghouse should commit, in the SSAR, that the potential plant site needs to meet the identified bounding parameters.

Closed - The shallow soil site (shear wave velocity < 1000 fps, depth to bedrock < 100 ft.) is excluded by the requirement that the shear wave velocity be greater than 1000 fps.
Clarification requested in NRC letter of 4/5/96 - Requirement for site specific time history and PSD criteria added in SSAR 2.5.4.5.5 which covers site specific seismic input.
Action W - Resolve the differences in NRC and Westinghouse position on site qualification. The shallow soil site (shear wave velocity < 1000 fps, depth to bedrock < 100 ft.) is excluded by the requirement that the shear wave velocity be greater than 1000 fps. COL must demonstrate that site is acceptable for the AP600 design. NRC position is that site specific analysis must use ZPA of 0.3 for SSE. Westinghouse position is that site specific earthquake should be used for sites outside the interface.
Discussed at NRC Management meeting 7-17-96 - further technical discussions required to clarify issue.
Westinghouse position identified in letter NSD-NRC-96-4804, dated 8/26/96.
NRC Status: Action W - The SSAR proposal presented at the December 1996 meeting does not satisfy the staff position. Westinghouse will respond. (12/16/96)
See NRC letter of 12/9/96.
Closed - Westinghouse provided a position on shallow soil sites in NSD-NRC-97-4956 of 1/28/97.
Action W - NRC Letter 1/31/97, The AP600 standard seismic design parameter is 0.3g peak ground acceleration with the response spectra shown in Figures 3.7.1-1 and 2.
The staff has reviewed the standard safety analysis report (SSAR), Revision 9. In SSAR Subsection 2.5.4.5.5, "Response of Soil and Rock to Dynamic Loading," Westinghouse states that for sites where the soil characteristics are outside the range considered in Appendix 2B.2 of the SSAR, site-specific soil-structure interaction analyses may be performed by the Combined Operating License (COL) applicant to demonstrate acceptability and that the analysis would use the site-specific soil conditions and site-specific safe shutdown earthquake. NRC staff has discussed this issue in the past and determined that this proposal was unacceptable. See NRC letter dated January 31, 1997, for further discussion on this issue. Westinghouse needs to revise the SSAR.
Action-N - Response provided in NSD-NRC-97-5041, March 26, 1997
Action W - respond to new RAIs, see open items 5234, 5235, 5236.

2
of
2

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5041
DCP/NRC0788
Docket No.: STN-52-003

March 26, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: SEISMIC AND SITE PARAMETER OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR section 2.5 and 3.7. Markup of proposed changes to SSAR Sections 2.5 and 3.7 and appendix 3F are also provided. These responses and markups are intended to resolve the identified open items related to these sections.

The changes in site parameters should be consistent with the interpretation of the regulations by the NRC staff. The approach includes information discussed between Westinghouse legal counsel and NRC Office of General Counsel and is similar to the approach previously approved for another ALWR design certification. The changes included in Section 3.7 are generally changes previously discussed with the staff.

This information will be discussed in a meeting between Westinghouse and NRC staff tentatively scheduled for April 14-18, 1997. 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
N. J. Liparulo, Westinghouse (w/o Attachment)

3129A

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The draft SSAR material provides a markup of Chapter 2, Section 3.7 and Appendix 3F. It responds to the following open items:

Open Item #	DSER Question	Response
547	2.5.4.3-2	Short and long term settlement are addressed in a revision to subsection 2.5.4.3.
623	3.7.1-2	SSAR subsection 3.7.1.1 and Appendix 3F are revised as shown in the markup. Damping for cable trays has been revised per agreement in January, 1997 meeting. Damping for configurations not similar to the tests referenced in the SSAR has been reduced to 10% matching the recommendations in a report by Brookhaven.
628	3.7.1.1-1	The shallow soil sites are excluded from the AP600 design basis due to the limitation on soil shear wave velocity. This is specifically shown in a revision to subsection 3.7.1.4.
649	3.7.2.4-7	Design information has been added in subsection 3.7.2.8 for the annex building demonstrating that there will be no unacceptable seismic interaction. The method of evaluation of seismic interaction for the radwaste building is also added in subsection 3.7.2.8.
662	3.7.2.8-5	Subsection 3.7.2.8 has been revised to show use of eccentric bracing or the turbine building.
664	3.7.2.8-7	Turbine building behavior is addressed for seismic margin in PRA Chapter 55. A draft copy was provided with letter NSD-NRC-97-5014, dated March 18, 1997.
670, 1885	3.7.2.16-1	Requirement has been added for COL applicant to reconcile seismic analyses of structures for as-built data.
769	3.8.5-11	SSAR subsection 2.5.4.5 has been revised to provide information on lateral soil variability and the requirements for site evaluation. This is also referenced from 3.8.5 in SSAR Rev 11.
791	3.9.3.1-6	Comments on ductwork were addressed in Appendix 3A in SSAR Rev 11. Comments on electrical raceways are addressed in Appendix 3F.
4997	RAI 231.34	Chapter 2 has been revised to use the term "site parameters".

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3.7.1.3 Critical Damping Values

Energy dissipation within a structural system is represented by equivalent viscous dampers in the mathematical model. The damping coefficients used are based on the material, load conditions, and type of construction used in the structural system. The safe shutdown earthquake damping values used in the dynamic analysis are presented in Table 3.7.1-1. The damping values are based on Regulatory Guide 1.61, ASCE Standard 4-86 (Reference 3), and 5 percent damping for piping, except for the damping value of the primary coolant loop piping, which is based on Reference 22, and conduits, cable trays and their related supports.

The damping values for conduits, cable trays and their related supports are shown in Table 3.7.1-1 and Figure 3.7.1-13. The damping value of conduit, empty cable trays, and their related supports is similar to that of a bolted structure, namely 7 percent of critical. The damping value of filled cable trays and supports increases with increased cable fill and level of seismic excitation. ~~The damping value f~~For cable trays and supports demonstrated to be similar to those tested, damping values of Figure 3.7.1-13 may be used. These are~~-is~~ based on test results (Reference 19).

For structures or components composed of different material types, the composite modal damping is calculated using the strain energy method. The strain energy dependent modal damping values are computed based on Reference 20. The modal damping values equal:

$$\beta_n = \sum_{i=1}^{nc} \frac{\{\phi_n\}^T \beta_i [K_t]_i \{\phi_n\}}{\{\phi_n\}^T [K_t] \{\phi_n\}}$$

where:

β_n = ratio of critical damping for mode n

nc = number of elements

$\{\phi_n\}$ = mode n (eigenvector)

$\left[K_t\right]_i$ = stiffness matrix of element i

β_i = ratio of critical damping associated with element i

$\left[K_t\right]$ = total system stiffness matrix

Strain-dependent damping values are used for the foundation material for rock sites in accordance with Reference 5 and 6 and for soil sites in accordance with Reference 33. The strain-dependent damping curves for the foundation materials are presented in Figures 3.7.1-14 and 3.7.1-15 for rock material and soil material, respectively. The strain-dependent soil material damping is limited to 15 percent of critical damping.

3.7.1.4 Supporting Media for Seismic Category I Structures

The seismic design basis for the AP600 is to provide design coverage for as many plant sites as practical. For the design of seismic Category I structures, a set of four design soil profiles of various shear wave velocities is established in Appendices 2A and 2B. The four design soil profiles include a hard rock site, a soft rock site, an upper bound soft-to-medium soil site and a soft-to-medium soil site. The shear wave velocity profiles and related governing parameters of the four sites considered are the following:

- For the hard rock site, an upper bound case for firm sites using fixed base seismic analysis.

- For the soft rock site, a shear wave velocity of 2400 feet per second at the ground surface, increasing linearly to 3200 feet per second at a depth of 240 feet, and base rock at the depth of 120 feet.

- For the soft-to-medium soil site, a shear wave velocity of 1000 feet per second at ground surface, increasing parabolically to 2400 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water at grade level.

- For the upper bound soft-to-medium soil site, a shear wave velocity of 1414 feet per second at ground surface, increasing parabolically to 3394 feet per second at 240 feet, base rock at the depth of 120 feet, and ground water at grade level. The initial soil shear modulus profile is twice that of the soft-to-medium soil site.

The strain-dependent shear modulus curves for the foundation materials, together with the corresponding damping curves, are shown in Figures 3.7.1-14 and 3.7.1-15 for rock material and soil material, respectively. The shear wave velocity profile for the design soil profiles, with variation of depth to base rock, is shown in Figure 3.7.1-17.

The AP600 design for the site parameters given in Table 2-1 is implemented using analyses for the four soil conditions described above. These four soil conditions consist of competent material. They do not include profiles consisting of one or more thin layers overlying competent material (Standard Review Plan 3.7.2). The shear wave velocity parameter in Table 2-1 excludes these shallow soil sites from consideration as a part of design certification.

The AP600 may also be suitable at sites outside the bounds of the site parameters. These sites may be demonstrated to be acceptable using the methodology in subsection 2.5.2.2. These site specific analyses demonstrating acceptability of the site would be submitted as part of the Combined License application. The evaluation of the suitability of these sites is not included in design certification.

The AP600 nuclear island consists of three seismic Category I structures founded on a common basemat. The three structures that make up the nuclear island are the coupled auxiliary and shield buildings, the steel containment vessel, and the containment internal structures. The nuclear island is shown in Figure 3.7.1-16. The foundation embedment depth, foundation size, and total height of the seismic Category I structures are presented in Table 3.7.1-2.

3.7.1.3 Critical Damping Values

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$$\beta_n = \sum_{i=1}^{nc} \frac{\{\phi_n\}^T \beta_i [K_i]_i \{\phi_n\}}{\{\phi_n\}^T [K_t] \{\phi_n\}}$$

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nc = number of elements

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A coupled nuclear island stick model and design soil profile finite element models are used in the three-dimensional soil-structure interaction analysis described in subsection 3.7.2.4.

FAX to DINO SCALETTI

May 8, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #662 (DSER 3.7.2.8-5)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 22 calendar days away (17 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #662 (DSER 3.7.2.8-5) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5041 on March 26, 1997. This material has been incorporated into Revision 12 of the SSAR. A copy of this letter, with the pertinent attachments, and the affected pages of the SSAR are included with this fax. We believe the information in this letter completed our action on Item #662 and request that NRC review the material we have provided and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 662 And 662 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
662	NRR/ECGB	3.7.2.8-5	DSER-OI		Orr / SCS/ NRCSEIS/S	Action N	Action W	NTD-NRC-95-4433	

Westinghouse should acceptably address two issues related to the design of bracing systems of structures adjacent to the NI structures.

Westinghouse to provide following additional information on turbine building:
Demonstrate that collapse of floors between the turbine building and the nuclear island in event of SSE will not impair safety functions of the nuclear island.
NRC Letter dated July 18, 1996 - B.3. Demonstrate that the turbine building frames can withstand with concentric bracing a seismic ground acceleration of 0.3g. Establish post-construction verification of structural members, connections, dimensions, etc to provide that these are consistent with the design. Demonstrate that the turbine building foundation will not pound the nuclear island wall at the foundation level.
Classification of turbine building is discussed in letter NSD-NRC-96-4854, dated 10/22/96. Further discussion is needed on behavior of K versus X bracing.
NRC Status: Action W - The staff reviewed the Westinghouse October 21, 1996, letter response. Westinghouse provided diagrams in the December 1996 meeting. A telephone conference call is necessary. Westinghouse needs to classify the turbine building as Category II or use an eccentric bracing system in the design. (12/16/96)
Action W - The staff reviewed the Westinghouse October 21, 1996, letter response. Westinghouse provided diagrams in the December 1996 meeting. A telephone conference call is necessary. Westinghouse needs to classify the turbine building as Category II or use an eccentric bracing system in the design.
Action W - In the SSAR, Westinghouse proposed to use the "X" type concentric bracing system together with the Uniform Building Code (UBC) requirements for the design of turbine building. Based on a series of test performed at the earthquake engineering research center of University of California at Berkeley, a steel frame structure with concentric bracing designed based on UBC requirements will result in a nonconservative design. To resolve this issue, the following three options were provided to Westinghouse:
a. The SSAR could commit to the eccentric bracing systems together with the same design requirements in stead of the "X" type concentric bracing systems for the design of the turbine building.
b. Westinghouse could provide test data by the industry to demonstrate that the "X" type concentric bracing systems designed for UBC requirements will not fail or collapse in the event of an SSE
c. Westinghouse could demonstrate by analysis to show that the collapse of the turbine building will not impair the integrity of seismic Category I structures, systems or components.
Westinghouse must revise the SSAR based on the option it decides to incorporate.
Action-N - Response provided in NSD-NRC-97-5041, March 26, 1997

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Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5041
DCP/NRC0788
Docket No.: STN-52-003

March 26, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
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Dear Mr. Quay:

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The changes in site parameters should be consistent with the interpretation of the regulations by the NRC staff. The approach includes information discussed between Westinghouse legal counsel and NRC Office of General Counsel and is similar to the approach previously approved for another ALWR design certification. The changes included in Section 3.7 are generally changes previously discussed with the staff.

This information will be discussed in a meeting between Westinghouse and NRC staff tentatively scheduled for April 14-18, 1997. If you have any questions please contact Donald A. Lindgren at (412) 374-4856.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment

cc: D. Jackson, NRC
N. J. Liparulo, Westinghouse (w/o Attachment)

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628	3.7.1.1-1	The shallow soil sites are excluded from the AP600 design basis due to the limitation on soil shear wave velocity. This is specifically shown in a revision to subsection 3.7.1.4.
649	3.7.2.4-7	Design information has been added in subsection 3.7.2.8 for the annex building demonstrating that there will be no unacceptable seismic interaction. The method of evaluation of seismic interaction for the radwaste building is also added in subsection 3.7.2.8.
662	3.7.2.8-5	Subsection 3.7.2.8 has been revised to show use of eccentric bracing or the turbine building.
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791	3.9.3.1-6	Comments on ductwork were addressed in Appendix 3A in SSAR Rev 11. Comments on electrical raceways are addressed in Appendix 3F.
4997	RAI 231.34	Chapter 2 has been revised to use the term "site parameters".

either a time history or a response spectrum analysis for each structure. The maximum displacement of the roof of the annex building is 1.6 inches in the east-west direction. The minimum clearance between the structural elements of the annex building above grade and the nuclear island is 4 inches.

3.7.2.8.2 Radwaste Building

The radwaste building is classified as nonseismic and is designed to the seismic requirements of the Uniform Building Code, Zone 2A with an Importance Factor of 1.25. As shown in the radwaste building general arrangement in Figure 1.2-22, it is a small steel framed building. If it were to impact the nuclear island or collapse in the safe shutdown earthquake, it would not impair the integrity of the reinforced concrete nuclear island. The minimum clearance between the structural elements of the radwaste building above grade and the nuclear island is 4 inches.

Three methods are used to demonstrate that a potential radwaste building impact on the nuclear island during a seismic event will not impair its structural integrity:

- The maximum kinetic energy of the impact during a seismic event considers the maximum radwaste building and nuclear island velocities. The total kinetic energy is considered to be absorbed by the nuclear island and converted to strain energy. The deflection of the nuclear island is less than 0.2". The shear forces in the nuclear island walls are less than the ultimate shear strength based on a minus one standard deviation of test data.

- Stress wave evaluation shows that the stress wave resulting from the impact of the radwaste building on the nuclear island has a maximum compressive stress less than the concrete compressive strength.

- An energy comparison shows that the kinetic energy of the radwaste building is less than the kinetic energy of tornado missiles for which the exterior walls of the nuclear island are designed.

3.7.2.8.3 Turbine Building

The turbine building is classified as nonseismic. As shown on the turbine building general arrangement in Figures 1.2-23 through 1.2-30, the major structure of the turbine building is separated from the nuclear island by approximately 18 feet. Floors between the turbine building main structure and the nuclear island provide access to the nuclear island. The floor beams are supported on the outside face of the nuclear island with a nominal horizontal clearance of 12 inches between the structural elements of the turbine building and the nuclear island. These beams are of light construction such that they will collapse if the differential deflection of the two buildings exceeds the clearance and will not jeopardize the two foot thick walls of the nuclear island. The roof in this area rests on the roof of the nuclear island and could slide relative to the roof of the nuclear island in a large earthquake. The seismic design is upgraded from Zone 2A, Importance Factor of 1.25, to Zone 3 with an Importance

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Factor of 1.0 in order to provide margin against collapse during the safe shutdown earthquake. The turbine building is ~~a concentrically~~ an eccentrically braced steel frame structure designed to meet the following criteria:

- The turbine building is designed in accordance with ACI-318 for concrete structures and with AISC for steel structures. Seismic loads are defined in accordance with the 1991 Uniform Building Code provisions for Zone 3 with an Importance Factor of 1.0. For an eccentrically braced structure the resistance modification factor is 10 (UBC-91, reference 1) using allowable stress design. When using allowable stress design, the allowable stresses are not increased by one third for seismic loads. The resistance modification factor is reduced to 7 for load and resistance factor design (ASCE 7-93, reference 35).

- The nominal horizontal clearance between the structural elements of the turbine building above grade and the nuclear island and annex building is 12 inches.

- ~~Steel structural bracing connections are designed with sufficient strength to develop tensile yield in the bracing before the connection fails.~~ The design of the lateral bracing system complies with the seismic requirements for eccentrically braced frames given in section 9.3 of the AISC Seismic Provisions for Structural Steel Buildings. (reference 34). Quality assurance is in accordance with ASCE 7-93 (reference 35) for the lateral bracing system.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Seismic model uncertainties due to, among other things, uncertainties in material properties, mass properties, damping values, the effect of concrete cracking, and the modeling techniques are accounted for in the widening of floor response spectra, as described in subsection 3.7.2.5. Stresses in the concrete structural elements due to the safe shutdown earthquake are below the tensile strength of the concrete. The effect of cracking of the concrete-filled structural modules inside containment due to thermal loads is discussed in subsection 3.8.3.4.2.

3.7.2.10 Use of Constant Vertical Static Factors

The vertical component of the safe shutdown earthquake is considered to occur simultaneously with the two horizontal components in the seismic analyses. Therefore, constant vertical static factors are not used for the design of seismic Category I structures.

3.7.2.11 Method Used to Account for Torsional Effects

The seismic analysis models of the nuclear island incorporate the mass and stiffness eccentricities of the seismic Category I structures and the torsional degrees of freedom. An accidental torsional moment is included in the design of the nuclear island structures. The accidental torsional moment due to the eccentricity of each mass is determined using the following:

minimum clearance between the structural elements of the annex building above grade and the nuclear island is 4 inches.

3.7.2.8.2 Radwaste Building

The radwaste building is classified as nonseismic and is designed to the seismic requirements of the Uniform Building Code, Zone 2A with an Importance Factor of 1.25. As shown in the radwaste building general arrangement in Figure 1.2-22, it is a small steel framed building. If it were to impact the nuclear island or collapse in the safe shutdown earthquake, it would not impair the integrity of the reinforced concrete nuclear island. The minimum clearance between the structural elements of the radwaste building above grade and the nuclear island is 4 inches.

Three methods are used to demonstrate that a potential radwaste building impact on the nuclear island during a seismic event will not impair its structural integrity:

- The maximum kinetic energy of the impact during a seismic event considers the maximum radwaste building and nuclear island velocities. The total kinetic energy is considered to be absorbed by the nuclear island and converted to strain energy. The deflection of the nuclear island is less than 0.2". The shear forces in the nuclear island walls are less than the ultimate shear strength based on a minus one standard deviation of test data.

- Stress wave evaluation shows that the stress wave resulting from the impact of the radwaste building on the nuclear island has a maximum compressive stress less than the concrete compressive strength.

- An energy comparison shows that the kinetic energy of the radwaste building is less than the kinetic energy of tornado missiles for which the exterior walls of the nuclear island are designed.

3.7.2.8.3 Turbine Building

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The turbine building is an eccentrically braced steel frame structure designed to meet the following criteria:

- The turbine building is designed in accordance with ACI-318 for concrete structures and with AISC for steel structures. Seismic loads are defined in accordance with the 1991 Uniform Building Code provisions for Zone 3 with an Importance Factor of 1.0. For an eccentrically braced structure the resistance modification factor is 10 (UBC-91, reference 1) using allowable stress design. When using allowable stress design, the allowable stresses are not increased by one third for seismic loads. The resistance modification factor is reduced to 7 for load and resistance factor design (ASCE 7-93, reference 35).

- The nominal horizontal clearance between the structural elements of the turbine building above grade and the nuclear island and annex building is 12 inches.

- The design of the lateral bracing system complies with the seismic requirements for eccentrically braced frames given in section 9.3 of the AISC Seismic Provisions for Structural Steel Buildings. (reference 34). Quality assurance is in accordance with ASCE 7-93 (reference 35) for the lateral bracing system.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Seismic model uncertainties due to, among other things, uncertainties in material properties, mass properties, damping values, the effect of concrete cracking, and the modeling techniques are accounted for in the widening of floor response spectra, as described in subsection 3.7.2.5. Stresses in the concrete structural elements due to the safe shutdown earthquake are below the tensile strength of the concrete. The effect of cracking of the concrete-filled structural modules inside containment due to thermal loads is discussed in subsection 3.8.3.4.2.

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3.7.2.11 Method Used to Account for Torsional Effects

The seismic analysis models of the nuclear island incorporate the mass and stiffness eccentricities of the seismic Category I structures and the torsional degrees of freedom. An accidental torsional moment is included in the design of the nuclear island structures. The accidental torsional moment due to the eccentricity of each mass is determined using the following:

- Horizontal mass properties of the building stick models shown in Figures 3.7.2-4, 3.7.2-5, and 3.7.2-6,

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FAX to DINO SCALETTI

May 8, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #649 (DSER 3.7.2.4-7)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 22 calendar days away (17 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #649 (DSER 3.7.2.4-7) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5041 on March 26, 1997. Material from the letter has been incorporated into Revision 12 of the SSAR. A copy of this letter, with the pertinent attachments, and the affected pages of the SSAR are included with this fax. We believe the information submitted herewith completes our action on Item #649 and request that NRC review the material we have provided and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 649 And 649 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
649	NRR/ECGB	3.7.2.4-7	DSER-OI		Orr / BPC / NRCSEIS/S	Action N	Action W	NSD-NRC-96-4825	

Westinghouse should evaluate the localized through-soil SSI effect of non-seismic Category I structures on the design of embedded seismic Category I walls and the potential for pounding between structures. Additional issues identified in NRC Letter dated July 18, 1996.

Results of the 2D SASSI analyses to determine the loads on the exterior walls below grade are included in SSAR Revision 7, Appendix 2C.
Potential for pounding between buildings is addressed in SSAR 3.7.2.8.
Action W- See meeting notes 7/18/96. Determine the effect of adjacent non-seismic Category I buildings on the lateral pressure on nuclear island walls below grade due to horizontal seismic ground motions; justify that existing analyses adequately represent the gap between the buildings.
Closed - Response provided in NSD-NRC-96-4825. Refined model has minimal effect on results.

Relative deflections from SASSI 2D analyses are available for review in December, 1996 meeting.

NRC Status: Action W - The staff reviewed the draft SSAR 3.7.2.8 in the December 1996 meeting. The draft is incomplete. Westinghouse does not have a means, such as a COL action item, to demonstrate that non-Category I structures adjacent to NI do not interact with NI. (12/16/96)
Action-N - Response provided in NSD-NRC-97-5041, March 26, 1997

8 of 2

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5041
DCP/NRC0788
Docket No.: STN-52-003

March 26, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: SEISMIC AND SITE PARAMETER OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR section 2.5 and 3.7. Markup of proposed changes to SSAR Sections 2.5 and 3.7 and appendix 3F are also provided. These responses and markups are intended to resolve the identified open items related to these sections.

The changes in site parameters should be consistent with the interpretation of the regulations by the NRC staff. The approach includes information discussed between Westinghouse legal counsel and NRC Office of General Counsel and is similar to the approach previously approved for another ALWR design certification. The changes included in Section 3.7 are generally changes previously discussed with the staff.

This information will be discussed in a meeting between Westinghouse and NRC staff tentatively scheduled for April 14-18, 1997. If you have any questions please contact Donald A. Lindgren at (412) 374-4856.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment

cc: D. Jackson, NRC
N. J. Liparulo, Westinghouse (w/o Attachment)

3129A

3 of 8

The draft SSAR material provides a markup of Chapter 2, Section 3.7 and Appendix 3F. It responds to the following open items:

Open Item #	DSER Question	Response
547	2.5.4.3-2	Short and long term settlement are addressed in a revision to subsection 2.5.4.3.
623	3.7.1-2	SSAR subsection 3.7.1.1 and Appendix 3F are revised as shown in the markup. Damping for cable trays has been revised per agreement in January, 1997 meeting. Damping for configurations not similar to the tests referenced in the SSAR has been reduced to 10% matching the recommendations in a report by Brookhaven.
628	3.7.1.1-1	The shallow soil sites are excluded from the AP600 design basis due to the limitation on soil shear wave velocity. This is specifically shown in a revision to subsection 3.7.1.4.
649	3.7.2.4-7	Design information has been added in subsection 3.7.2.8 for the annex building demonstrating that there will be no unacceptable seismic interaction. The method of evaluation of seismic interaction for the radwaste building is also added in subsection 3.7.2.8.
662	3.7.2.8-5	Subsection 3.7.2.8 has been revised to show use of eccentric bracing or the turbine building.
664	3.7.2.8-7	Turbine building behavior is addressed for seismic margin in PRA Chapter 55. A draft copy was provided with letter NSD-NRC-97-5014, dated March 18, 1997.
670, 1885	3.7.2.16-1	Requirement has been added for COL applicant to reconcile seismic analyses of structures for as-built data.
769	3.8.5-11	SSAR subsection 2.5.4.5 has been revised to provide information on lateral soil variability and the requirements for site evaluation. This is also referenced from 3.8.5 in SSAR Rev 11.
791	3.9.3.1-6	Comments on ductwork were addressed in Appendix 3A in SSAR Rev 11. Comments on electrical raceways are addressed in Appendix 3F.
4997	RAI 231.34	Chapter 2 has been revised to use the term "site parameters".

The modal responses of the response spectrum system structural analysis are combined using the double sum method shown in Section C of Regulatory Guide 1.92, Revision 1. When high frequency effects are significant, they are included using the procedure given in Appendix A to SRP 3.7.2. In the fixed base mode superposition time history analysis of the hard rock site, the total seismic response is obtained by superposing the modal responses within the analytical procedure so that further combination is not necessary.

A summary of the dynamic analyses performed and the combination techniques used are presented in Table 3.7.2-16.

3.7.2.8 Interaction of Seismic Category II and Nonseismic Structures with Seismic Category I Structures, Systems or Components

Nonseismic structures are evaluated to determine that their seismic response does not preclude the safety functions of seismic Category I structures, systems or components. This is accomplished by satisfying one of the following:

- The collapse of the nonseismic structure will not cause the nonseismic structure to strike a seismic Category I structure, system or component.
- The collapse of the nonseismic structure will not impair the integrity of seismic Category I structures, systems or components.
- The structure is classified as seismic Category II and is analyzed and designed to prevent its collapse under the safe shutdown earthquake.

The structures adjacent to the nuclear island are the annex building, the radwaste building, and the turbine building.

3.7.2.8.1 Annex Building

The annex building is classified as seismic Category II and is designed to prevent its collapse under the safe shutdown earthquake. The structural configuration is shown in Figure 3.7.2-19. The annex building is analyzed for the safe shutdown earthquake for the four sites described in subsection 3.7.1.4. Seismic input is defined by response spectra applied at the base of a dynamic model of the annex building. The horizontal spectra are obtained from the 2D SASSI analyses described in Appendix 2C and account for soil-structure and structure-soil-structure interaction. Input in the east-west direction uses the response spectra obtained from the two dimensional analyses for the annex building mat. Input in the north-south direction uses the response spectra obtained from the two dimensional analyses for the turbine building mat. Vertical input is obtained from 2D FLUSH finite element soil-structure interaction analyses. The envelope of the maximum building response acceleration values is applied as equivalent static loads to a more detailed static model.

The minimum space required between the annex building and the nuclear island to avoid contact is obtained by absolute summation of the deflections of each structure obtained from either a time history or a response spectrum analysis for each structure. The maximum displacement of the roof of the annex building is 1.6 inches in the east-west direction. The minimum clearance between the structural elements of the annex building above grade and the nuclear island is 4 inches.

3.7.2.8.2 Radwaste Building

The radwaste building is classified as nonseismic and is designed to the seismic requirements of the Uniform Building Code, Zone 2A with an Importance Factor of 1.25. As shown in the radwaste building general arrangement in Figure 1.2-22, it is a small steel framed building. If it were to impact the nuclear island or collapse in the safe shutdown earthquake, it would not impair the integrity of the reinforced concrete nuclear island. The minimum clearance between the structural elements of the radwaste building above grade and the nuclear island is 4 inches.

Three methods are used to demonstrate that a potential radwaste building impact on the nuclear island during a seismic event will not impair its structural integrity:

- The maximum kinetic energy of the impact during a seismic event considers the maximum radwaste building and nuclear island velocities. The total kinetic energy is considered to be absorbed by the nuclear island and converted to strain energy. The deflection of the nuclear island is less than 0.2". The shear forces in the nuclear island walls are less than the ultimate shear strength based on a minus one standard deviation of test data.

- Stress wave evaluation shows that the stress wave resulting from the impact of the radwaste building on the nuclear island has a maximum compressive stress less than the concrete compressive strength.

- An energy comparison shows that the kinetic energy of the radwaste building is less than the kinetic energy of tornado missiles for which the exterior walls of the nuclear island are designed.

3.7.2.8.3 Turbine Building

The turbine building is classified as nonseismic. As shown on the turbine building general arrangement in Figures 1.2-23 through 1.2-30, the major structure of the turbine building is separated from the nuclear island by approximately 18 feet. Floors between the turbine building main structure and the nuclear island provide access to the nuclear island. The floor beams are supported on the outside face of the nuclear island with a nominal horizontal clearance of 12 inches between the structural elements of the turbine building and the nuclear island. These beams are of light construction such that they will collapse if the differential deflection of the two buildings exceeds the clearance and will not jeopardize the two foot thick walls of the nuclear island. The roof in this area rests on the roof of the nuclear island and could slide relative to the roof of the nuclear island in a large earthquake. The seismic design is upgraded from Zone 2A, Importance Factor of 1.25, to Zone 3 with an Importance

Appendix A to SRP 3.7.2. In the fixed base mode superposition time history analysis of the hard rock site, the total seismic response is obtained by superposing the modal responses within the analytical procedure so that further combination is not necessary.

A summary of the dynamic analyses performed and the combination techniques used are presented in Table 3.7.2-16.

3.7.2.8 Interaction of Seismic Category II and Nonseismic Structures with Seismic Category I Structures, Systems or Components

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- The collapse of the nonseismic structure will not cause the nonseismic structure to strike a seismic Category I structure, system or component.
- The collapse of the nonseismic structure will not impair the integrity of seismic Category I structures, systems or components.
- The structure is classified as seismic Category II and is analyzed and designed to prevent its collapse under the safe shutdown earthquake.

The structures adjacent to the nuclear island are the annex building, the radwaste building, and the turbine building.

3.7.2.8.1 Annex Building

The annex building is classified as seismic Category II and is designed to prevent its collapse under the safe shutdown earthquake. The structural configuration is shown in Figure 3.7.2-19. The annex building is analyzed for the safe shutdown earthquake for the four sites described in subsection 3.7.1.4. Seismic input is defined by response spectra applied at the base of a dynamic model of the annex building. The horizontal spectra are obtained from the 2D SASSI analyses described in Appendix 2C and account for soil-structure and structure-soil-structure interaction. Input in the east-west direction uses the response spectra obtained from the two dimensional analyses for the annex building mat. Input in the north-south direction uses the response spectra obtained from the two dimensional analyses for the turbine building mat. Vertical input is obtained from 2D FLUSH finite element soil-structure interaction analyses. The envelope of the maximum building response acceleration values is applied as equivalent static loads to a more detailed static model.

The minimum space required between the annex building and the nuclear island to avoid contact is obtained by absolute summation of the deflections of each structure obtained from either a time history or a response spectrum analysis for each structure. The maximum displacement of the roof of the annex building is 1.6 inches in the east-west direction. The

minimum clearance between the structural elements of the annex building above grade and the nuclear island is 4 inches.

3.7.2.8.2 Radwaste Building

The radwaste building is classified as nonseismic and is designed to the seismic requirements of the Uniform Building Code, Zone 2A with an Importance Factor of 1.25. As shown in the radwaste building general arrangement in Figure 1.2-22, it is a small steel framed building. If it were to impact the nuclear island or collapse in the safe shutdown earthquake, it would not impair the integrity of the reinforced concrete nuclear island. The minimum clearance between the structural elements of the radwaste building above grade and the nuclear island is 4 inches.

Three methods are used to demonstrate that a potential radwaste building impact on the nuclear island during a seismic event will not impair its structural integrity:

- The maximum kinetic energy of the impact during a seismic event considers the maximum radwaste building and nuclear island velocities. The total kinetic energy is considered to be absorbed by the nuclear island and converted to strain energy. The deflection of the nuclear island is less than 0.2". The shear forces in the nuclear island walls are less than the ultimate shear strength based on a minus one standard deviation of test data.

- Stress wave evaluation shows that the stress wave resulting from the impact of the radwaste building on the nuclear island has a maximum compressive stress less than the concrete compressive strength.

- An energy comparison shows that the kinetic energy of the radwaste building is less than the kinetic energy of tornado missiles for which the exterior walls of the nuclear island are designed.

3.7.2.8.3 Turbine Building

The turbine building is classified as nonseismic. As shown on the turbine building general arrangement in Figures 1.2-23 through 1.2-30, the major structure of the turbine building is separated from the nuclear island by approximately 18 feet. Floors between the turbine building main structure and the nuclear island provide access to the nuclear island. The floor beams are supported on the outside face of the nuclear island with a nominal horizontal clearance of 12 inches between the structural elements of the turbine building and the nuclear island. These beams are of light construction such that they will collapse if the differential deflection of the two buildings exceeds the clearance and will not jeopardize the two foot thick walls of the nuclear island. The roof in this area rests on the roof of the nuclear island and could slide relative to the roof of the nuclear island in a large earthquake. The seismic design is upgraded from Zone 2A, Importance Factor of 1.25, to Zone 3 with an Importance Factor of 1.0 in order to provide margin against collapse during the safe shutdown earthquake.

FAX to DINO SCALETTI

May 8, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #664 (DSER 3.7.2.8-7)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 22 calendar days away (17 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #664 (DSER 3.7.2.8-7) is attached. Action was completed on this Item by our submittal of our letters NTD-NRC-97-5041 on March 26, 1997 and NTD-NRC-97-5014 on March 18, 1997. Information from the latter letter was incorporated into Revision 9 of the PRA. A copy of these letters, with the pertinent attachments, and the affected pages of the PRA are included with this fax. We believe this information completed our action on Item #664 and request that NRC review the material we have provided and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

1 of 15

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 664 And 664 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
664	NRR/ECGB	3.7.2.8-7	DSER-OI		Orr/Lapay	Closed	Action W	NTD-NRC-95-4464	

Westinghouse should demonstrate and document in the SSAR, for the evaluation of seismic margin, that both seismic Category II and non-seismic structures can withstand an earthquake up to 0.5g without collapse.

Seismic Category II and nonseismic structures are not directly included in the seismic margins assessment. The seismic design of these structures is described in Subsection 3.7.2.8 of the SSAR.
Seismic Category II building structures are designed for the safe shutdown earthquake using the same methods as are used for seismic Category I structures. The seismic Category II structures are the annex building and the stair tower to the shield building roof. These would have seismic capability similar to the seismic Category I structures. Therefore, it is expected that they will withstand an earthquake greater than 0.50g as shown in the seismic margin assessment for the seismic Category I structures in Appendix H of the PRA report, revison 1.
Nonseismic structures are generally analyzed and designed for seismic loads according to the Uniform Building Code requirements for Zone 2A with an Importance Factor of 1.25. The radwaste and turbine buildings are nonseismic structures. As described in Subsection 3.7.2.8, collapse of the radwaste building would not would impair the integrity of the reinforced concrete nuclear island.

As described in Subsection 3.7.2.8, the major structure of the turbine building is separated from the nuclear island by approximately eighteen feet and the seismic design of the turbine building has been upgraded to UBC Zone 3 with an Importance Factor of 1.0 in order to provide margin against collapse during the safe shutdown earthquake. The turbine building may not withstand the 0.5g earthquake without potential local collapse. However, it is separated from the nuclear island, and the equipment essential to safe shutdown is well protected by the thick concrete walls, floors and roof slab of the nuclear island. Hence the failure of the turbine building is not considered in the seismic margins assessment since its collapse is unlikely to impair the integrity of equipment essential to safe shutdown.

Staff update provided during 8/17/95 meeting: Statement made above that "collapse of the radwaste building would not impair the integrity of the reinforced concrete nuclear island" is a judgemental conclusion. Also, floors between T-building and NI may impact the safety of NI.
Discussion will be added in seismic margin report on why collapse of the nonseismic buildings is not expected to result in core damage.
Meeting 7/17-20/96 - Westinghouse evaluation of collapse of turbine building to be reviewed during seismic margins meeting.
Closed in chapter 3.7: transferred to seismic margins review
Action W - See NRC letter of 12/9/96.
Westinghouse to document that turbine building collapse does not lead to core damage.
Action-N - Response provided in NSD-NRC-97-5041, March 26, 1997 and NSD-NRC-97-5014, March 18, 1997.

2
of
5

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5041
DCP/NRC0788
Docket No.: STN-52-003

March 26, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: SEISMIC AND SITE PARAMETER OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR section 2.5 and 3.7. Markup of proposed changes to SSAR Sections 2.5 and 3.7 and appendix 3F are also provided. These responses and markups are intended to resolve the identified open items related to these sections.

The changes in site parameters should be consistent with the interpretation of the regulations by the NRC staff. The approach includes information discussed between Westinghouse legal counsel and NRC Office of General Counsel and is similar to the approach previously approved for another ALWR design certification. The changes included in Section 3.7 are generally changes previously discussed with the staff.

This information will be discussed in a meeting between Westinghouse and NRC staff tentatively scheduled for April 14-18, 1997. 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
N. J. Liparulo, Westinghouse (w/o Attachment)

3 of 15

The draft SSAR material provides a markup of Chapter 2, Section 3.7 and Appendix 3F. It responds to the following open items:

Open Item #	DSER Question	Response
547	2.5.4.3-2	Short and long term settlement are addressed in a revision to subsection 2.5.4.3.
623	3.7.1-2	SSAR subsection 3.7.1.1 and Appendix 3F are revised as shown in the markup. Damping for cable trays has been revised per agreement in January, 1997 meeting. Damping for configurations not similar to the tests referenced in the SSAR has been reduced to 10% matching the recommendations in a report by Brookhaven.
628	3.7.1.1-1	The shallow soil sites are excluded from the AP600 design basis due to the limitation on soil shear wave velocity. This is specifically shown in a revision to subsection 3.7.1.4.
649	3.7.2.4-7	Design information has been added in subsection 3.7.2.8 for the annex building demonstrating that there will be no unacceptable seismic interaction. The method of evaluation of seismic interaction for the radwaste building is also added in subsection 3.7.2.8.
662	3.7.2.8-5	Subsection 3.7.2.8 has been revised to show use of eccentric bracing or the turbine building.
664	3.7.2.8-7	Turbine building behavior is addressed for seismic margin in PRA Chapter 55. A draft copy was provided with letter NSD-NRC-97-5014, dated March 18, 1997.
670, 1885	3.7.2.16-1	Requirement has been added for COL applicant to reconcile seismic analyses of structures for as-built data.
769	3.8.5-11	SSAR subsection 2.5.4.5 has been revised to provide information on lateral soil variability and the requirements for site evaluation. This is also referenced from 3.8.5 in SSAR Rev 11.
791	3.9.3.1-6	Comments on ductwork were addressed in Appendix 3A in SSAR Rev 11. Comments on electrical raceways are addressed in Appendix 3F.
4997	RAI 231.34	Chapter 2 has been revised to use the term "site parameters".

4215

minimum clearance between the structural elements of the annex building above grade and the nuclear island is 4 inches.

3.7.2.8.2 Radwaste Building

The radwaste building is classified as nonseismic and is designed to the seismic requirements of the Uniform Building Code, Zone 2A with an Importance Factor of 1.25. As shown in the radwaste building general arrangement in Figure 1.2-22, it is a small steel framed building. If it were to impact the nuclear island or collapse in the safe shutdown earthquake, it would not impair the integrity of the reinforced concrete nuclear island. The minimum clearance between the structural elements of the radwaste building above grade and the nuclear island is 4 inches.

Three methods are used to demonstrate that a potential radwaste building impact on the nuclear island during a seismic event will not impair its structural integrity:

- The maximum kinetic energy of the impact during a seismic event considers the maximum radwaste building and nuclear island velocities. The total kinetic energy is considered to be absorbed by the nuclear island and converted to strain energy. The deflection of the nuclear island is less than 0.2". The shear forces in the nuclear island walls are less than the ultimate shear strength based on a minus one standard deviation of test data.

- Stress wave evaluation shows that the stress wave resulting from the impact of the radwaste building on the nuclear island has a maximum compressive stress less than the concrete compressive strength.

- An energy comparison shows that the kinetic energy of the radwaste building is less than the kinetic energy of tornado missiles for which the exterior walls of the nuclear island are designed.

3.7.2.8.3 Turbine Building

The turbine building is classified as nonseismic. As shown on the turbine building general arrangement in Figures 1.2-23 through 1.2-30, the major structure of the turbine building is separated from the nuclear island by approximately 18 feet. Floors between the turbine building main structure and the nuclear island provide access to the nuclear island. The floor beams are supported on the outside face of the nuclear island with a nominal horizontal clearance of 12 inches between the structural elements of the turbine building and the nuclear island. These beams are of light construction such that they will collapse if the differential deflection of the two buildings exceeds the clearance and will not jeopardize the two foot thick walls of the nuclear island. The roof in this area rests on the roof of the nuclear island and could slide relative to the roof of the nuclear island in a large earthquake. The seismic design is upgraded from Zone 2A, Importance Factor of 1.25, to Zone 3 with an Importance Factor of 1.0 in order to provide margin against collapse during the safe shutdown earthquake.

The turbine building is an eccentrically braced steel frame structure designed to meet the following criteria:

- The turbine building is designed in accordance with ACI-318 for concrete structures and with AISC for steel structures. Seismic loads are defined in accordance with the 1991 Uniform Building Code provisions for Zone 3 with an Importance Factor of 1.0. For an eccentrically braced structure the resistance modification factor is 10 (UBC-91, reference 1) using allowable stress design. When using allowable stress design, the allowable stresses are not increased by one third for seismic loads. The resistance modification factor is reduced to 7 for load and resistance factor design (ASCE 7-93, reference 35).

- The nominal horizontal clearance between the structural elements of the turbine building above grade and the nuclear island and annex building is 12 inches.

- The design of the lateral bracing system complies with the seismic requirements for eccentrically braced frames given in section 9.3 of the AISC Seismic Provisions for Structural Steel Buildings. (reference 34). Quality assurance is in accordance with ASCE 7-93 (reference 35) for the lateral bracing system.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Seismic model uncertainties due to, among other things, uncertainties in material properties, mass properties, damping values, the effect of concrete cracking, and the modeling techniques are accounted for in the widening of floor response spectra, as described in subsection 3.7.2.5. Stresses in the concrete structural elements due to the safe shutdown earthquake are below the tensile strength of the concrete. The effect of cracking of the concrete-filled structural modules inside containment due to thermal loads is discussed in subsection 3.8.3.4.2.

3.7.2.10 Use of Constant Vertical Static Factors

The vertical component of the safe shutdown earthquake is considered to occur simultaneously with the two horizontal components in the seismic analyses. Therefore, constant vertical static factors are not used for the design of seismic Category I structures.

3.7.2.11 Method Used to Account for Torsional Effects

The seismic analysis models of the nuclear island incorporate the mass and stiffness eccentricities of the seismic Category I structures and the torsional degrees of freedom. An accidental torsional moment is included in the design of the nuclear island structures. The accidental torsional moment due to the eccentricity of each mass is determined using the following:

- Horizontal mass properties of the building stick models shown in Figures 3.7.2-4, 3.7.2-5, and 3.7.2-6,

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5014
DCP/NRC0774
Docket No.: STN-52-003

March 18, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: DRAFT SEISMIC MARGIN ANALYSIS (PRA CHAPTER 55) AND RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Dear Mr. Quay:

Enclosure 1 of this letter provides a copy of draft AP600 PRA Chapter 55, Seismic Margin Analysis. Chapter 55 will be included in Revision 9 to the PRA, which is scheduled for early April 1997. No changes are expected between the draft copy enclosed with this letter and the Chapter 55 that will be included in the next PRA revision, except that a change bar will be added in the margin since this is a new chapter to the PRA. If there are any changes, they will be clearly identified to the staff.

Enclosure 2 provides Westinghouse responses to NRC requests for additional information, follow-on questions, and meeting open items pertaining to seismic margin analysis and the high-confidence, low probability of failure (HCLPF) calculations.

The NRC is expected to review draft PRA Chapter 55 and the responses provided in Enclosure 2 as preparation for a meeting between Westinghouse and NRC. The purpose of the meeting will be for the NRC to perform their review of the final AP600 HCLPF calculations. To maintain the schedule outlined in SECY-97-051, Westinghouse believes the HCLPF review meeting with the NRC must occur no later than April 11. Westinghouse also requests that the meeting occur in our Monroeville office to allow the support engineering staff to be available to answer potential questions the NRC may have during their review of the HCLPF calculations. Cindy Haag will contact Diane Jackson to set the meeting dates.

Enclosure 3 provides Westinghouse responses to seven NRC requests for additional information pertaining to the AP600 internal fire analysis which is documented in Chapter 57 of the AP600 PRA, and one RAI on the shutdown PRA.

A listing of the NRC requests for additional information and open items responded to in this letter is

7/

contained in Attachment A.

The Westinghouse status of the OITS for six of the internal fire analysis RAIs will be changed to "Confrm-W" while the Westinghouse status for the other responses provided in this letter will be changed to "Closed." The NRC should review these responses and inform Westinghouse of the status to be designated in the "NRC Status" column of the OITS.

Enclosure 4 provides markup page changes to the AP600 PRA. Specifically, there are changes to PRA pages 11-10, 11-13, 14-2, 37-5, 38-5, 41-5, 49-8, and 49-11. These changes will be included in the next revision to the PRA. The staff reviewers should include these markup pages with their PRA report.

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

Enclosures

cc: J. Sebrosky, NRC (Attachment and Enclosures 1, 2, 3 and 4)
D. Jackson, NRC (Attachment and Enclosures 1 and 2)
N. J. Liparulo, Westinghouse (w/o attachment/enclosures)

8of

structures, as well as adequate anchorage load transfer and structural ductility. The seismic margin evaluation provides a means of identifying specific equipment or structures that are vulnerable beyond design basis seismic events.

55.2.2.5 Verification of Equipment Fragility Data

The AP600 safety-related equipment is designed to meet the safe shutdown earthquake requirements defined in Chapter 3 of the AP600 SSAR. This seismic margin evaluation has focused on demonstrating that the design of the nuclear island structures, safety-related equipment, and equipment supports can carry the loads induced by the review level earthquake discussed here. This evaluation incorporates as-specified equipment data. After the plant has been built, it will be necessary to perform a verification of the seismic margin assessment for the installed conditions.

55.2.2.6 Turbine Building Seismic Interaction

As part of the seismic margin assessment, the seismic interaction between the Turbine Building and the Nuclear Island was evaluated (subsection 55.5.8). It was determined that:

- The adjacent Auxiliary Building structural integrity will not be lost with the failure of the Turbine Building.

- It is not likely that the size and energy of debris from the Turbine Building will be large enough to result in penetration through the Auxiliary Building roof structure.

Even though it is not likely that Turbine Building debris could be large enough or have sufficient energy for penetration through the Auxiliary Building roof structure, this event was evaluated. The consequences of damage to the safety-related equipment in the Auxiliary Building was investigated. It was determined from this investigation that, should an event occur that causes the failure of equipment in the upper elevations of the Auxiliary Building, the results of the seismic margin assessment analysis, the plant HCLPF value, and the insights derived from the seismic margin assessment would not be affected. Moreover, according to the AP600 focused PRA results, steam line break events that would result from damage to equipment in the upper elevations are not dominant contributors to the core damage frequency. Further, any loss of equipment in the upper elevations would not affect the passive safety systems used to put the plant in a safe shutdown condition should an event occur.

55.3 Seismic Margin Model

This section provides the AP600 risk-based seismic margin model. Based on this model, the cutset, system, and plant HCLPFs are calculated in Section 55.4. The risk-based seismic margin model meets the PRA-based seismic margin requirement as specified in SECY-93-087 (Reference 55-1).

Also, the secondary side safety relief valves should not fail open by the seismic event.

The main steam line isolation, feedwater isolation, PORV block and relief valves are located in Rooms 12406 and 12404. These rooms have a HCLPF value of 0.81g.

A seismic failure of any one of these valves is assumed to fail the SLB isolation, with a HCLPF of 0.81g. Moreover, the failure of the main steam line pressure transmitters also fails the closure of MSIVs, with a HCLFP value of 0.88g. Finally, failure of PMS or Class 1E dc power also fails the automatic actuation, with HCLPF values of 0.74g and 0.51g respectively.

Thus, the HCLPF value for SLB isolation is 0.81g, without the support systems; it becomes 0.51g if support systems are considered.

A successful SLB isolation reduces the possibility of a consequential SGTR event, following a large SLB.

2. For an unisolated SLB, a success path with the PRHR system and the CMTs operational is defined in the focused PRA, although this success path is not used in the SMA. If this path is used, then the HCLPF for the failure of this path would be:

Seismically induced SLB occurs:	HCLPF = 0.81g
PRHR fails:	HCLPF = 0.60g
CMTs fail:	HCLPF = 0.63g

The failure of this success path has a HCLPF value of 0.81g. Thus, the HCLPF is the same as the one calculated for the more conservative success path used for SLB event in the SMA model.

55.5.8 Seismic Interaction Between Turbine and Auxiliary Buildings

Introduction

The AP600 seismic margin analysis shows the plant HCLPF is 0.50g. One of the model assumptions is:

".. only the Nuclear Island is considered for the SMA model; the interaction between the other buildings and the Nuclear Island is assumed to have no detrimental effect on the Nuclear Island structures."

The Turbine Building is designed to the Uniform Building Code requirements. It is taller than the Auxiliary Building, which is a Seismic Category I structure. The Auxiliary Building contains important safety-related equipment. The Turbine Building is adjacent to the north-end wing of the Auxiliary Building, the wing containing the main control room (MCR) and the shutdown panel, as well as I&C rooms and I&C penetration rooms. The main structure of the Turbine Building is separated from the Nuclear Island by an access bay, which is approximately 19 feet wide. The consequences of the potential Turbine Building collapse onto and falling debris penetrating the Auxiliary Building is evaluated here.

The conclusions of the assessment of structural integrity consequences of the Turbine Building potential collapse are as follows:

1. The horizontal impact of the Turbine Building on the Auxiliary Building will not result in the loss of structural integrity of the Auxiliary Building.

2. It is not likely that the size and the energy of the debris would be large enough to result in a vertical impact on the roof of the Auxiliary Building to cause penetration.

Even though it is not likely that penetration of Turbine Building debris could be large and energetic enough for penetration into the roof of the north end of the Auxiliary Building, this event was evaluated and the consequences of damage to the safety-related equipment in the Auxiliary Building investigated.

Evaluation

The rooms at different levels of the north end of the Auxiliary Building are listed below. These rooms are labeled by 123NN, 124NN, and 125NN, where NN are integers (for example, the MCR is 12401). The integers 3, 4, and 5 in the room numbers are indicators of various levels, 3 being a lower level than 4, etc. Some of the rooms located at lower levels contain equipment important to the plant control functions.

The roof and upper floors of the Auxiliary Building include a 15-inch concrete slab at elevation 153' for the roof, a 24-inch concrete slab at elevation 135'3" over the main control room, and a 24-inch concrete slab at elevation 117'6" on the floor of the main control room and MSIV areas. The 24-inch thick concrete floors would not allow penetration of debris into the other rooms in lower levels of the same building because the debris would not have sufficient energy to do so. Thus the damage from the impacting debris is confined to upper levels of the structure, namely rooms with numbers 124NN (except the MCR) and 125NN. These rooms are:

12404	Lower MSIV compartment B	SGS containment isolation valves, instrumentation and controls
12405	Lower VBS B and D equipment room	VWS/PXS/CAS containment isolation valves
12406	Lower MSIV compartment A	SGS containment isolation valves, instrumentation and controls

12504	Lower MSIV compartment B	SGS containment isolation valves, instrumentation and controls
12505	Upper VBS B and D equipment room	VWS/PXS/CAS containment isolation valves
12506	Lower MSIV compartment A	SGS containment isolation valves, instrumentation and controls

The other rooms at io. - levels contain MCR, I&C penetration rooms, I&C rooms, valve and piping penetration rooms, and the remote shutdown panel. It is important that these rooms (labeled with room numbers 123NN, plus the MCR) would not be penetrated, since their penetration may potentially fail plant control through either failure of MCR and shutdown panel; or failure of multiple I&C division connections.

Given that the only credible failures may occur in the 124NN and 125NN rooms (except the MCR), the most credible event to occur is a steam line break (SLB) transient; this event would be coupled with a loss of offsite power (LOSP) event expected to occur at low acceleration levels. During such a seismically induced LOSP event, nonsafety-related systems are assumed to fail (they are not credited in the SMA model in plant HCLPF calculations), leading to a station blackout event.

Upon penetration of the Turbine Building debris through the Auxiliary Building roof, damage to some containment isolation valves outside of the containment (in the rooms listed above) is possible; but each containment penetration has a second isolation valve inside the containment, unaffected by the Turbine Building collapse. Thus, containment isolation failures are not credible.

The following SLB event scenarios are discussed:

1. SLB-TURB-LOWG (seismic event less than 0.50g)
2. SLB-TURB-HIGHG (seismic event greater than 0.50g)

SLB-TURB-LOWG (seismic event less than 0.50g)

This scenario is described as follows:

i. A low-level seismic event occurs; the plant safety systems are not seismically affected (e.g., the effect of the seismic event on the plant structures and components is less than the 0.50g plant HCLPF).

ii. A loss of offsite power event occurs and the control rods are inserted due to loss of power; nonsafety-related systems (including diesel generators) are assumed to have failed due to the event.

iii. The Turbine Building fails by the seismic event and large, energetic debris from this failure falls over the north end of the Auxiliary Building, penetrating the roof. A steam line break event occurs due to the damage to pipes or valves. This break is conservatively assumed to be unisolable.

This event is very similar to the SLB-V event modeled and quantified in the focused PRA. The event mitigation and success paths are the same; the only difference is the cause of the initiating event. This event would not affect the plant HCLPF calculated in the SMA since it only has mixed cutsets, and has no seismically induced failures of safety-related systems. However, it is discussed below for a better understanding of its consequences.

The success path for the above scenario is that the PRHR and CMT systems automatically actuate and perform their functions. Since both systems have air-operated valves that open and actuate the systems upon loss of power, air, or signal, the system actuation does not require any support systems, such as PMS signal, or dc power. Thus, the event mitigation is truly passive and highly reliable.

In the highly unlikely case that the PRHR or the CMTs fail, a second success path is considered in the focused PRA. This path takes credit for successful actuation of ADS, IRWST injection, and recirculation.

The SLB-V event has an initiating event frequency of 1.21E-03/year (from the focused PRA) and has a core damage frequency of 3.66E-09/year. This is about 0.5-percent contribution to the core damage frequency from internal events. The conditional probability of core damage for this event (given that the SLB occurred) is 3.0E-06. This conditional probability is also applicable to the seismically induced SLB scenario described above, since it considers only safety-related systems for the event mitigation.

The low probability of core damage given that an unisolable SLB occurs in scenario SLB-TURB-LOWG, coupled with the fact that the frequency of such seismically initiated events would be small, indicates that this scenario is not a significant risk contributor.

SLB-TURB-HIGHG (seismic event greater than 0.50g)

This scenario is described as follows:

i. A high-level seismic event occurs; the plant safety systems may be seismically affected (e.g., the effect of the seismic event on the plant structures and components is greater than the 0.50g plant HCLPF).

ii. A loss of offsite power event occurs and the control rods are inserted due to loss of power; nonsafety-related systems (including the diesel generators) are assumed to have failed due to the event.

iii. The Turbine Building fails by the seismic event and large, energetic debris from this failure falls over the north end of the Auxiliary Building, penetrating the roof. A steam line break event occurs due to the damage to pipes or valves. This break is conservatively assumed to be unisolable.

This SLB event is considered in the SMA model. The current scenario is very similar to it except the initiating event HCLPF is much lower than the one used in the SMA model. Thus, the success path described earlier is valid.

Since the HCLPF value of the Turbine Building collapse event for this scenario is less than 0.50g, the seismic core damage sequence HCLPFs would be determined by the failure of the PRHR or CMT system failures (success path 1) discussed in the previous scenario. As discussed before, both systems have air-operated valves that open without the need for PMS or dc power. Both systems have high HCLPF values (much higher than the plant HCLPF value of 0.50g), as demonstrated in the SMA model. Thus, the results of the SMA and its insights are not affected by this scenario.

Conclusions

Although the north end of the Auxiliary Building contains important safety-related control and I&C equipment, the main control room, and the remote shutdown panel, a design feature prevents the damage to this important equipment from a seismically induced Turbine Building failure. This design feature, a two-foot concrete floor (barrier) over this equipment and control rooms, is an important factor in excluding control failures. This limits the potential failures from the Turbine Building debris to steam lines and some containment isolation valves.

The potential event scenarios from the seismic interaction of the turbine and the north end of the Auxiliary Building are studied. Two remotely credible, but highly unlikely scenarios involving unisolable steam line breaks below and above 0.50g level are identified and are examined from their impact on SMA. The conclusion is that the SMA results and insights would not be affected by the seismic interaction between the Turbine Building and the north end of the Auxiliary Building.

55.9 SMA Results and Insights

55.9.1 AP600 SMA Results

The AP600 seismic margin analysis has demonstrated that for structures, systems, and components required for safe shutdown the high confidence, low probability of failures (HCLPF) magnitudes are equal to or greater than 0.50g. This HCLPF is determined by the seismically induced failure of the fuel in the reactor vessel.

Thus, the AP600 design meets the requirement set forth in the AP600 SSAR that the plant meet or exceed a review level earthquake of 0.5g.

The success paths used for the SMA are taken conservatively in many cases, and no credit for operator actions for events at 0.50g review level earthquake have been included. Thus, the results are valid without operator intervention, which indicates a strong point of the AP600 design ability to mitigate seismically induced core damage and large release sequences.

All the SMA sequences are evaluated with loss of offsite power and loss of onsite ac power leading to a station blackout event. The plant design is shown to be robust against seismic event sequences each of which contain station blackout coupled with other seismic or random failures.

The dominant seismic sequences for plant HCLPF are the same for core damage and the large fission product release. These sequences, and their dominant contributors are as follows:

FAX to DINO SCALETTI

May 9, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #5032 (DSER 3.8.5)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #5032 (DSER 3.8.5) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5070 on April 15, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We request that NRC review the material we have provided concerning Item #5032and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 5032 And 5032 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
5032	NRR/ECGB	3.8.5	MTG-OI		Orr	Closed	Action W	NSD-NRC-97-5032	

Meeting December, 1996 - NRC Letter March 4, 1997 - Design Adequacy of 18-inch Thick Part of the Basemat in the Elevator Pit.
The staff reviewed Calculation No. 1010-CCC-005, Rev. 1 by Initec and observed the following:
a) The soil reaction (= 19 ksf) was obtained from the nonlinear ANSYS analysis of the NI.
b) The basemat panel containing the pit (6 ft. x 9 ft.) was subjected to soil reaction load and was analyzed using a finite element model.
c) The shear force at the support (= 51k/ft) was read approximately from a contour map. Shear adequacy check was made at a distance t/2 + d, where t is the wall thickness and d is the slab effective thickness. At this location, the 51 k/ft support shear was reduced to a very low negative value. However, if the 6 ft x 9 ft pit slab is separately modeled as supported at its edges (by the wall and the 6 ft. thick basemat slab), a simple two-way slab analysis shows that the end reaction is about 48 k/ft (versus 51 k/ft from the panel finite element analysis) and the shear at the critical section (at distance d from the edge) is about 28 kips. This is much larger than what has been used in the design. This discrepancy raised a generic concern about the use of shear force computed from the contour at the edge (i.e. 51 k/ft in this case) and then to assume this value at the center of a very wide support (=36 inches), which artificially reduces the shear at the critical section. This issue was discussed with Westinghouse.

Closed - Response provided by NSD-NRC-97-5070 of 4/15/97. jww

4 of 2

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5070
DCP/NRC0815
Docket No.: STN-52-003

April 15, 1997

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

ATTENTION: T. R. Quay

SUBJECT: SEISMIC AND STRUCTURAL OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR Sections 3.7 and 3.8. These responses are intended to resolve the identified open items related to this section.

The items addressed include the following:

OITS #	DSER Item or other	W Status
750	3.8.4.4-2	Closed
766	3.8.5-8	Audit N
5028	December, 1996 Meeting Item	Closed
5030	December, 1996 Meeting Item	Closed
5031	December, 1996 Meeting Item	Confirm W
5032	December, 1996 Meeting Item	Closed

The Westinghouse Status for the open item tracking system is noted for each of the items above. The NRC staff is requested to review the responses and confirm the status.

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

D. A. Lindgren /For
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml
Attachment

cc: D. Jackson, NRC (w/Attachment)
F. Fanous, Ames Lab (w/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

3714

2158A

Open Item # 5032

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 15, as follows:

"Design Adequacy of 18-inch Thick Part of the Basemat in the Elevator Pit.

The staff reviewed Calculation No. 1010-CCC-005, Rev. 1 by Initec and observed the following:

a) The soil reaction (= 19 ksf) was obtained from the nonlinear ANSYS analysis of the NI.

b) The basemat panel containing the pit (6 ft. x 9 ft.) was subjected to soil reaction load and was analyzed using a finite element model.

c) The shear force at the support (= 51k/ft) was read approximately from a contour map. Shear adequacy check was made at a distance t/2 + d, where t is the wall thickness and d is the slab effective thickness. At this location, the 51 k/ft support shear was reduced to a very low negative value. However, if the 6 ft x 9 ft pit slab is separately modeled as supported at its edges (by the wall and the 6 ft. thick basemat slab), a simple two-way slab analysis shows that the end reaction is about 48 k/ft (versus 51 k/ft from the panel finite element analysis) and the shear at the critical section (at distance d from the edge) is about 28 kips. This is much larger than what has been used in the design. This discrepancy raised a generic concern about the use of shear force computed from the contour at the edge (i.e. 51 k/ft in this case) and then to assume this value at the center of a very wide support (=36 inches), which artificially reduces the shear at the critical section. This issue was discussed with Westinghouse."

Westinghouse response

The shear force at the edge of the pit was read correctly from the ANSYS contour plot. It was then used as though it occurred at the center of the wall in a manner similar to the design for shear on other basemat panels without the elevator pit. For such cases shear adequacy is checked at a distance t/2 + d from the center line of the wall. For the elevator pit where the shear force was read at the edge of the pit, shear adequacy should be checked at a distance, d, from the support. The calculation has been revised.

FAX to DINO SCALETTI

May 9, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #5030 (DSER 3.8.5)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #5030 (DSER 3.8.5) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5070 on April 15, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We request that NRC review the material we have provided concerning Item #5030and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 5030 And 5030 Sorted by Type

Item No	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
5030	NRR/ECGB	3.8.5	MTG-OI	December 1996 meeting - NRC Letter March 4, 1997 - The Effects of In-plane Shear in the Basemat Design The staff's review of the basemat design calculation showed that large in-plane shears in the basemat elements (as evidenced in Initec's nonlinear ANSYS analysis) were not considered. Since such large shears can potentially induce principal tension in a non-orthogonal direction, consideration of these shear forces may result in additional reinforcements. This issue was discussed with Westinghouse, Bechtel and Initec personnel. However, no definitive response was given by Westinghouse. Closed - Response provided by NSD-NRC-97-5070 of 4/15/97. jww	Orr	Closed	Action W	NSD-NRC-97-5070	

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Total Records: 1

Westinghouse Electric Corporation
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Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5070
DCP/NRC0815
Docket No.: STN-52-003

April 15, 1997

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

ATTENTION: T. R. Quay

SUBJECT: SEISMIC AND STRUCTURAL OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR Sections 3.7 and 3.8. These responses are intended to resolve the identified open items related to this section.

The items addressed include the following:

OITS #	DSER Item or other	W Status
750	3.8.4.4-2	Closed
766	3.8.5-8	Audit N
5028	December, 1996 Meeting Item	Closed
5030	December, 1996 Meeting Item	Closed
5031	December, 1996 Meeting Item	Confirm W
5032	December, 1996 Meeting Item	Closed

The Westinghouse Status for the open item tracking system is noted for each of the items above. The NRC staff is requested to review the responses and confirm the status.

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

D. A. Lindgren /for
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml
Attachment

3d4

cc: D. Jackson, NRC (w/Attachment)
F. Fanous, Ames Lab (w/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

3158A

The methodology for design of the walls has been revised to consider the out-of-plane seismic loads caused by both the inertia of the panel mass (local) as well as by the global behavior of the wall-floor system. The design out-of-plane moments are taken from the results of the finite element response spectrum analysis. When the finite element model includes representation of the wall mass between floors, these moments include both the global and local behavior. Where the finite element model does not include an intermediate node between floors, these moments only include the component due to the global behavior. For such cases, the out-of-plane seismic loads on the wall panels caused by the inertia of the panel mass are added using hand calculations based on their mass and acceleration.

Open Item # 5030

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 11, as follows:

"The Effects of In-Plane Shear in the Basemat Design

The staff's review of the basemat design calculation showed that large in-plane shears in the basemat elements (as evidenced in Initec's nonlinear ANSYS analysis) were not considered. Since such large shears can potentially induce principal tension in a non-orthogonal direction, consideration of these shear forces may result in additional reinforcements. This issue was discussed with Westinghouse, Bechtel and Initec personnel. However, no definitive response was given by Westinghouse."

Westinghouse response

The effects of in-plane shears were considered in the basemat design by calculating the principal in-plane stresses in critical elements. If the resulting principal stress was tensile it was used as an axial tension together with the moments to calculate the required reinforcing steel.

Open Item # 5031

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 14, as follows:

"Correction in Figure 3.7.2-18 of SSAR Rev. 9

The staff indicated that Elevation 85.5 ft., shown in Figure 3.7.2-18, seems to be inconsistent with Elevation 82.5 ft., shown in Figure 3.7.2-6. Also, Figure 3.7.2-18 seems to lack a lateral support at Elevation 82.5 ft. Westinghouse agreed to examine these apparent discrepancies and update the SSAR accordingly."

Westinghouse Response

The figure will be revised in SSAR revision 12. Draft changes were provided in letter NSD-NRC-97-5041, dated March 27, 1997.

FAX to DINO SCALETTI

May 9, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #766 (DSER 3.8.5-8)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #766 (DSER 3.8.5-8) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5070 on April 15, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We request that NRC review the material we have provided concerning Item #766 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 766 And 766 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
766	NRR/ECGB	3.8.5-8	DSER-OI		Orr	Audit N	Action W	NSD-NRC-97-5070	

Westinghouse should provide the validation package of INITEC's in-house computer codes for review and should verify the adequacy of the post-processed results obtained from these codes.

Closed - Validation package is available for review in proposed basemat meeting in December, 1996. Comparisons of results of computer code versus hand calaculation are included in revised documentation.
Action W - Westinghouse will provide technical information and final design calculations for the staff to review in meeting in December.
NRC Status: Action W - Westinghouse has not provided (in English) the ARMA computer code validation and verification package. (12/16/96)
Audit N - Response provided by NSD-NRC-97-5070 of 4/15/97. jww

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Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5070
DCP/NRC0815
Docket No.: STN-52-003

April 15, 1997

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

ATTENTION: T. R. Quay

SUBJECT: SEISMIC AND STRUCTURAL OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR Sections 3.7 and 3.8. These responses are intended to resolve the identified open items related to this section.

The items addressed include the following:

OITS #	DSER Item or other	W Status
750	3.8.4.4-2	Closed
766	3.8.5-8	Audit N
5028	December, 1996 Meeting Item	Closed
5030	December, 1996 Meeting Item	Closed
5031	December, 1996 Meeting Item	Confirm W
5032	December, 1996 Meeting Item	Closed

The Westinghouse Status for the open item tracking system is noted for each of the items above. The NRC staff is requested to review the responses and confirm the status.

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

D. A. Lindgren /for

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml
Attachment

3rd

cc: D. Jackson, NRC (w/Attachment)
F. Fanous, Ames Lab (w/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

1158A

Table 1 Member forces in tension ring

Force	N (kips)	Qz (kips)	Qx (kips)	Mx (k.ft)	T (k.ft)	Mz (k.ft)
Ames	2524	512	144	856	904	785
W	2554	755	252	323	380	454

Open Item # 766 DSER Open Item 3.8.5-8

The open issue is summarized in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 7, as follows:

"Validation of Initec's Post-Processing Computer Codes

The staff could not review the validation of these codes since the validation package provided for review was mostly in Spanish. Westinghouse agreed to provide an English version of the package for the staff to review."

Westinghouse response

An english version of the package is available for staff review

Open Item # 5028

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 4, as follows:

"Consideration of Out-of-Plane Seismic Load in the Design of Wall Panels

Out-of-plane seismic loads on the wall panels are caused by the inertia of the panel mass (local) as well as by the global behavior of the wall-floor system (which may cause panel support displacements resulting in out-of-plane moments and shears). But in designing the wall panels, Initec considered only the local out-of-plane bending and did not consider the global out-of-plane moments and shears that were calculated by Bechtel.

The staff discussed this with Initec, Westinghouse, and Bechtel personnel as a generic issue; it was agreed that the appropriate way of combining the local and global moments and shears and its impact in the design will be addressed by Westinghouse."

Westinghouse response

The out-of-plane seismic loads on the wall panels caused by the inertia of the panel mass have been considered in the design of the walls using hand calculations based on their mass and acceleration. This was done because generally there is only one element between floors. The out-of-plane moments shown in the finite element analysis results are primarily due to panel support displacements.

FAX to DINO SCALETTI

May 9, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #5028 (DSER 3.8.5)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #5028 (DSER 3.8.5) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5070 on April 15, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We request that NRC review the material we have provided concerning Item #5028 and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 5028 And 5028 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
5028	NRR/ECGB	3.8.5	MTG-OI		Orr	Closed	Action W	NSD-NRC-97-5070	

December 1996 Meeting - NRC Letter March 4, 1997 - Consideration of Out-of-Plane Seismic Load in the Design of Wall Panels
Out-of-plane seismic loads on the wall panels are caused by the inertia of the panel mass (local) as well as by the global behavior of the wall-floor system (which may cause panel support displacements resulting in out-of-plane moments and shears). But in designing the wall panels, Initec considered only the local out-of-plane bending and did not consider the global out-of-plane moments and shears that were calculated by Bechtel. The staff discussed this with Initec, Westinghouse, and Bechtel personnel as a generic issue, it was agreed that the appropriate way of combining the local and global moments and shears and its impact in the design will be addressed by Westinghouse.

Closed - Response provided by NSD-NRC-97-5070 of 4/15/97. jww

205

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5070
DCP/NRC0815
Docket No.: STN-52-003

April 15, 1997

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

ATTENTION: T. R. Quay

SUBJECT: SEISMIC AND STRUCTURAL OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR Sections 3.7 and 3.8. These responses are intended to resolve the identified open items related to this section.

The items addressed include the following:

OITS #	DSER Item or other	W Status
750	3.8.4.4-2	Closed
766	3.8.5-8	Audit N
5028	December, 1996 Meeting Item	Closed
5030	December, 1996 Meeting Item	Closed
5031	December, 1996 Meeting Item	Confirm W
5032	December, 1996 Meeting Item	Closed

The Westinghouse Status for the open item tracking system is noted for each of the items above. The NRC staff is requested to review the responses and confirm the status.

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

D. A. Lindgren /for
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml
Attachment

3.15

cc: D. Jackson, NRC (w/Attachment)
F. Fanous, Ames Lab (w/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

1198A

Table 1 Member forces in tension ring

Force	N (kips)	Qz (kips)	Qx (kips)	Mx (k.ft)	T (k.ft)	Mz (k.ft)
Ames	2524	512	144	856	904	785
W	2554	755	252	323	380	454

Open Item # 766 DSER Open Item 3.8.5-8

The open issue is summarized in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 7, as follows:

"Validation of Initec's Post-Processing Computer Codes

The staff could not review the validation of these codes since the validation package provided for review was mostly in Spanish. Westinghouse agreed to provide an English version of the package for the staff to review."

Westinghouse response

An english version of the package is available for staff review

Open Item # 5028

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 4, as follows:

"Consideration of Out-of-Plane Seismic Load in the Design of Wall Panels

Out-of-plane seismic loads on the wall panels are caused by the inertia of the panel mass (local) as well as by the global behavior of the wall-floor system (which may cause panel support displacements resulting in out-of-plane moments and shears). But in designing the wall panels, Initec considered only the local out-of-plane bending and did not consider the global out-of-plane moments and shears that were calculated by Bechtel.

The staff discussed this with Initec, Westinghouse, and Bechtel personnel as a generic issue; it was agreed that the appropriate way of combining the local and global moments and shears and its impact in the design will be addressed by Westinghouse."

Westinghouse response

The out-of-plane seismic loads on the wall panels caused by the inertia of the panel mass have been considered in the design of the walls using hand calculations based on their mass and acceleration. This was done because generally there is only one element between floors. The out-of-plane moments shown in the finite element analysis results are primarily due to panel support displacements.

The methodology for design of the walls has been revised to consider the out-of-plane seismic loads caused by both the inertia of the panel mass (local) as well as by the global behavior of the wall-floor system. The design out-of-plane moments are taken from the results of the finite element response spectrum analysis. When the finite element model includes representation of the wall mass between floors, these moments include both the global and local behavior. Where the finite element model does not include an intermediate node between floors, these moments only include the component due to the global behavior. For such cases, the out-of-plane seismic loads on the wall panels caused by the inertia of the panel mass are added using hand calculations based on their mass and acceleration.

Open Item # 5030

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 11, as follows:

"The Effects of In-Plane Shear in the Basemat Design

The staff's review of the basemat design calculation showed that large in-plane shears in the basemat elements (as evidenced in Initec's nonlinear ANSYS analysis) were not considered. Since such large shears can potentially induce principal tension in a non-orthogonal direction, consideration of these shear forces may result in additional reinforcements. This issue was discussed with Westinghouse, Bechtel and Initec personnel. However, no definitive response was given by Westinghouse."

Westinghouse response

The effects of in-plane shears were considered in the basemat design by calculating the principal in-plane stresses in critical elements. If the resulting principal stress was tensile it was used as an axial tension together with the moments to calculate the required reinforcing steel.

Open Item # 5031

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 14, as follows:

"Correction in Figure 3.7.2-18 of SSAR Rev. 9

The staff indicated that Elevation 85.5 ft., shown in Figure 3.7.2-18, seems to be inconsistent with Elevation 82.5 ft., shown in Figure 3.7.2-6. Also, Figure 3.7.2-18 seems to lack a lateral support at Elevation 82.5 ft. Westinghouse agreed to examine these apparent discrepancies and update the SSAR accordingly."

Westinghouse Response

The figure will be revised in SSAR revision 12. Draft changes were provided in letter NSD-NRC-97-5041, dated March 27, 1997.

FAX to DINO SCALETTI

May 8, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Richard Orr
Don Lindgren
Bob Vijuk
Brian McIntyre

OPEN ITEM #5031 (DSER 3.7.2)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 22 calendar days away (17 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #5031 (DSER 3.7.2.) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5070 on April 15, 1997 and incorporation of revisions to Figure 3.7.2-18 of Revision 12 of the SSAR. A copy of this letter, with the pertinent attachments, and the revised pages of the SSAR are included with this fax. We believe this information completed our action on these open items and request that NRC review the material we have provided and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

1 of 6

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 5031 And 5031 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
5031	NRR/ECGB	3.7.2	MTG-OI		Orr	Confirm-W	Action W	NSD-NRC-97-5070	

Meeting December, 1996, - NRC Letter March 4, 1997 - Correction in Figure 3.7.2-18 of SSAR Rev. 9
The staff indicated that Elevation 85.5 ft., shown in Figure 3.7.2-18, seems to be inconsistent with Elevation 82.5 ft., shown in Figure 3.7.2-6. Also, Figure 3.7.2-18 seems to lack a lateral support at Elevation 82.5 ft. Westinghouse agreed to examine these apparent discrepancies and update the SSAR accordingly.

Confirm W - Response provided by NSD-NRC-97-5070 of 4/15/97. jww

2 of 6

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230 0355

NSD-NRC-97-5070
DCP/NRC0815
Docket No.: STN-52-003

April 15, 1997

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

ATTENTION: T. R. Quay

SUBJECT: SEISMIC AND STRUCTURAL OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR Sections 3.7 and 3.8. These responses are intended to resolve the identified open items related to this section.

The items addressed include the following:

OITS #	DSER Item or other	W Status
750	3.8.4.4-2	Closed
766	3.8.5-8	Audit N
5028	December, 1996 Meeting Item	Closed
5030	December, 1996 Meeting Item	Closed
5031	December, 1996 Meeting Item	Confirm W
5032	December, 1996 Meeting Item	Closed

The Westinghouse Status for the open item tracking system is noted for each of the items above. The NRC staff is requested to review the responses and confirm the status.

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

D. A. Lindgren for
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml
Attachment

cc: D. Jackson, NRC (w/Attachment)
F. Fanous, Ames Lab (w/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

3 of 6

1158A

The methodology for design of the walls has been revised to consider the out-of-plane seismic loads caused by both the inertia of the panel mass (local) as well as by the global behavior of the wall-floor system. The design out-of-plane moments are taken from the results of the finite element response spectrum analysis. When the finite element model includes representation of the wall mass between floors, these moments include both the global and local behavior. Where the finite element model does not include an intermediate node between floors, these moments only include the component due to the global behavior. For such cases, the out-of-plane seismic loads on the wall panels caused by the inertia of the panel mass are added using hand calculations based on their mass and acceleration.

Open Item # 5030

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 11, as follows:

"The Effects of In-Plane Shear in the Basemat Design

The staff's review of the basemat design calculation showed that large in-plane shears in the basemat elements (as evidenced in Initec's nonlinear ANSYS analysis) were not considered. Since such large shears can potentially induce principal tension in a non-orthogonal direction, consideration of these shear forces may result in additional reinforcements. This issue was discussed with Westinghouse, Bechtel and Initec personnel. However, no definitive response was given by Westinghouse."

Westinghouse response

The effects of in-plane shears were considered in the basemat design by calculating the principal in-plane stresses in critical elements. If the resulting principal stress was tensile it was used as an axial tension together with the moments to calculate the required reinforcing steel.

Open Item # 5031

The open issue is identified in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 14, as follows:

"Correction in Figure 3.7.2-18 of SSAR Rev. 9

The staff indicated that Elevation 85.5 ft., shown in Figure 3.7.2-18, seems to be inconsistent with Elevation 82.5 ft., shown in Figure 3.7.2-6. Also, Figure 3.7.2-18 seems to lack a lateral support at Elevation 82.5 ft. Westinghouse agreed to examine these apparent discrepancies and update the SSAR accordingly."

Westinghouse Response

The figure will be revised in SSAR revision 12. Draft changes were provided in letter NSD-NRC-97-5041, dated March 27, 1997.

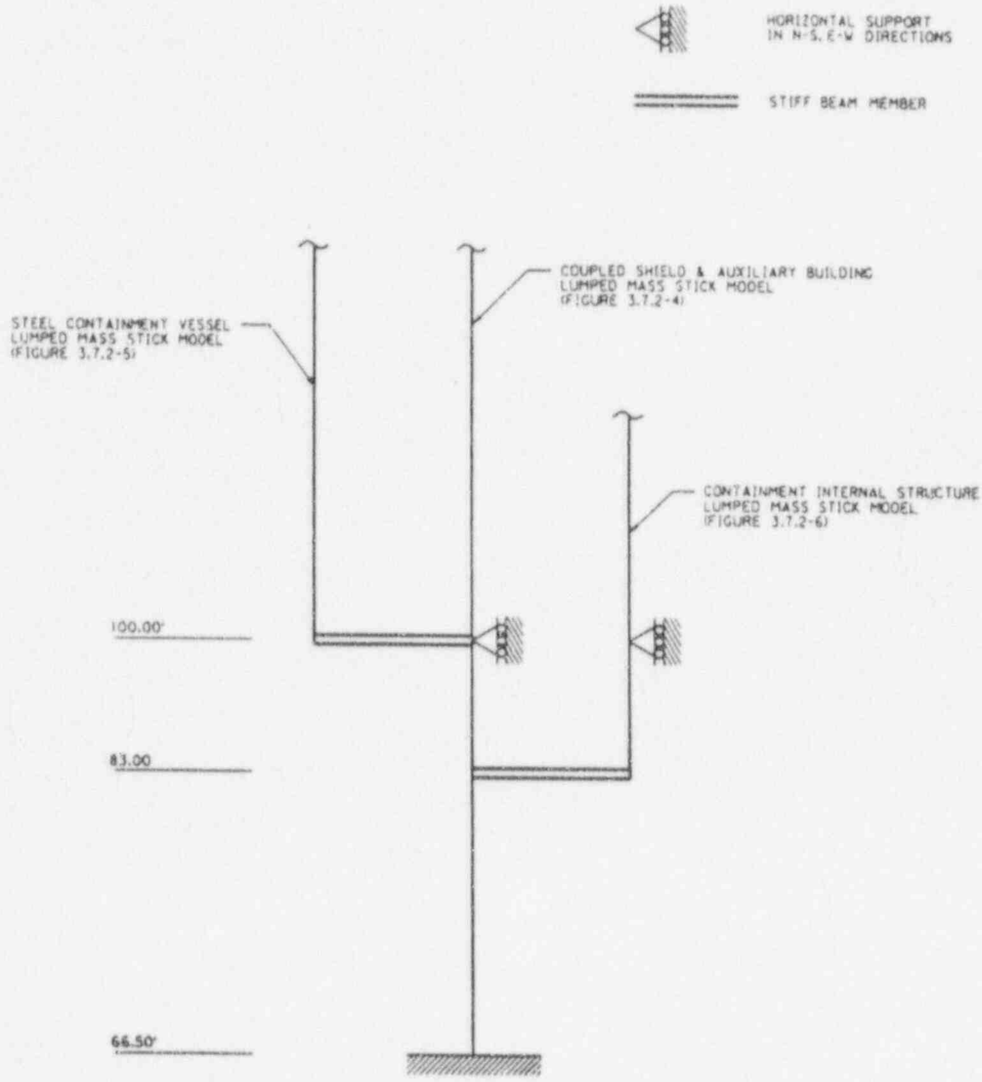

Figure 3.7.2-18 (Sheet 1 of 2)

Connection Between Lumped Mass Stick Model - Fixed Base Analysis

5 of 6

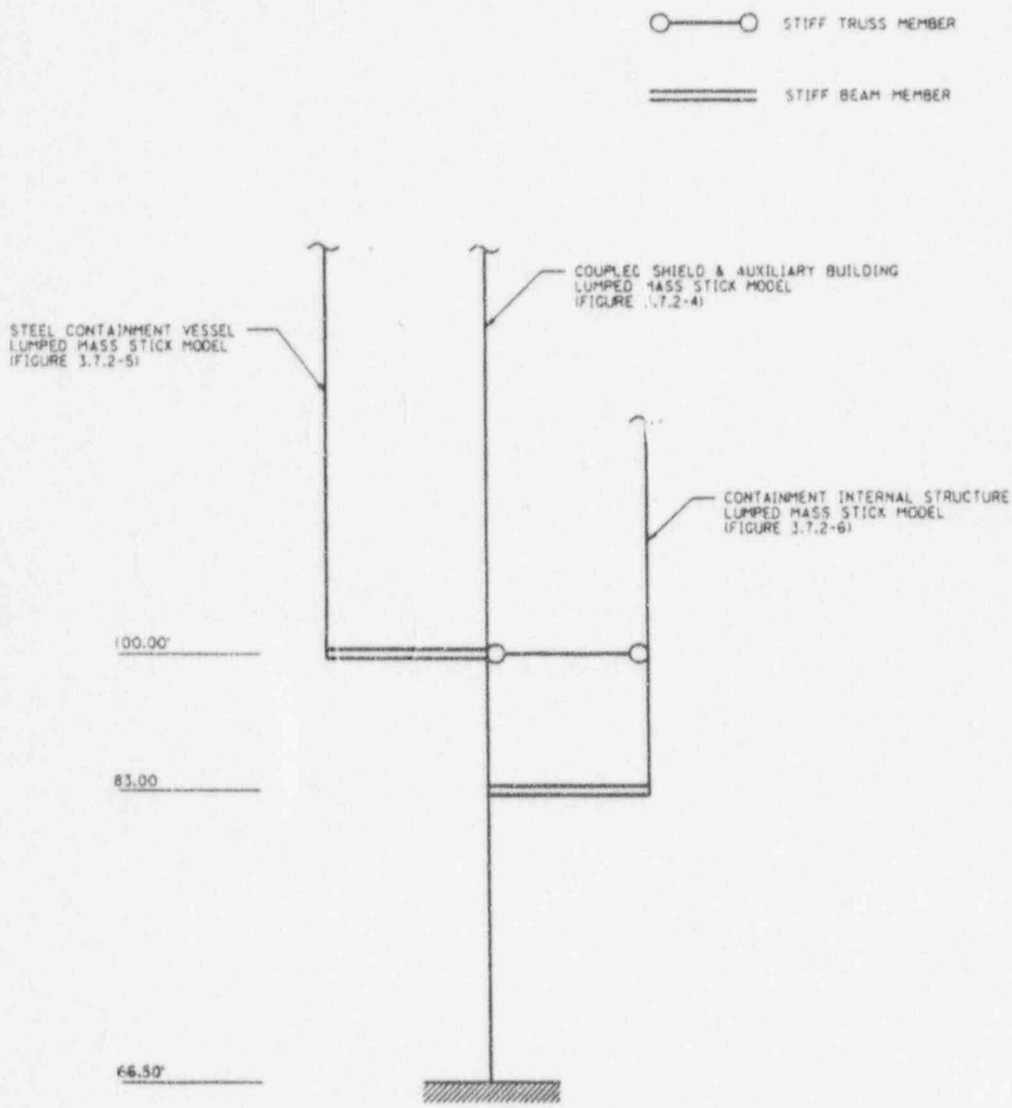

Figure 3.7.2-18 (Sheet 2 of 2)

Connection Between Lumped Mass Stick Model - Sassi Analysis

6 of 6

FAX to DINO SCALETTI

May 8, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #668 (DSER 3.7.2.12-1)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 22 calendar days away (17 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #668 (DSER 3.7.2.12-1) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-97-5105 on May 2, 1997. A copy of this letter, with the pertinent attachments, is included with this fax. We believe the information in this letter completed our action on Item #668 and request that NRC review the material we have provided and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 668 And 668 Sorted by Type

Item No	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No /	Date
668	NRR/ECGB	3.7.2.12-1	DSER-OI		Orr/BPC	Action N	Action W		

Westinghouse should compare the results from the response spectrum analysis method to those of the modal time-history analysis method.

Preliminary comparison of results from the response spectrum analysis method vs. the modal time-history analysis was presented in meeting with NRC on 6/13/95.
Comparison is included in SSAR Revision 7. Editorial changes included in SSAR revision 9.
Reopened in telecon 11/26/96. Westinghouse to provide additional comparisons in meeting in December, 1996.
NRC Status: Action W - Westinghouse to reconcile differences in time history and response spectrum results. (12/16/96)

Action W - In Revision 9 of SSAR Section 3.7.2.12, Westinghouse stated that the 3D lumped mass fixed-base stick model of the nuclear island was analyzed by modal superposition time history analysis and by the response spectrum analysis method for the hard rock site condition. The staff's review found that the maximum absolute nodal accelerations calculated by the response spectrum analysis are consistently higher than those from the modal superposition time history analysis. At some locations, the accelerations from these analyses are deviated by 30 percent in the East-West direction and 40 percent in the vertical direction. The staff's concern is that if the maximum nodal accelerations calculated by modal time history analyses are always lower than those obtained from response spectrum analyses, it implies that the floor response spectra generated based on the floor time histories may not be conservative for the design of subsystems such as piping. Westinghouse must justify the adequacy of the final design floor response spectra documented in the SSAR.
Response provided in letter NSD-NRC-97-5105, dated 5/2/97

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230 0355

NSD-NRC-97-5105
DCP/NRC0844
Docket No.: STN-52-003

May 2, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: Response to DSER Open Item 3.7.2.12-1

During a review of the structural design of the AP600 conducted by the NRC staff and consultants the week of December 9-13, 1996 a question was raised about the differences between the results of response spectra analyses and time history analyses for the nuclear island. This item is associated with DSER open item 3.7.2.12-1, OITS# 668. The NRC question on this issue was included in your letter dated March 4, 1997, Summary of Meeting to Discuss Westinghouse AP600 Structural Design.

The Westinghouse response for this item is attached. This response completes the Westinghouse action on this item except for formal revision of the SSAR. The Westinghouse status of this item in the Open Item Tracking System will be Confirm-W pending inclusion of the draft changes in the SSAR.

Please contact Donald A. Lindgren at (412) 374-4856 with any questions.

Brian A. McIntyre

Enclosure

cc: D. Jackson, NRC w/att.
T. Cheng, NRC w/att.
N. J. Liparulo, Westinghouse w/o att.

0121H

Open Item # 668 DSER Open Item 3.7.2.12-1

The open issue is summarized in the NRC letter of March 4, 1997, "Summary of Meeting to discuss Westinghouse AP600 structural design", item # 6, as follows:

"Difference between Response Spectra and Time-History Analyses

While making the comparison, the staff observed that: a) the maximum absolute nodal accelerations from response spectrum analysis are consistently higher, sometimes by a very large margin; b) because of item (a) above, the floor response spectra generated from floor time-histories may not be conservative; c) member forces from response spectrum and time history analyses do not always follow the same trend as the nodal accelerations. Sometime the trend is reversed, in which case the design forces and moments from the 3-D finite element response spectrum analysis may not be conservative.

Based on these observations, the staff asked Westinghouse to justify the adequacy of both the final design floor response spectra and the final design forces and moments for structural members."

Westinghouse response

1.0 Introduction

SSAR Tables 3.7.2-17, 3.7.2-18, and 3.7.2-19 show comparisons of maximum response accelerations, member forces and member moments between results from modal time history (TH) and response spectrum analyses (RSA) of the AP600 multiple stick fixed-base model. The differences between the results of the two analytical methods are discussed below. The AP600 final design floor response spectra and the final design forces and moments are adequate. Responses of the steel containment vessel are discussed in section 2, of the coupled auxiliary and shield building in section 3, and of the containment internal structures in section 4.

The time history and the response spectrum analyses use the same fixed-base, multiple-stick model consisting of the nuclear island (NI) stick, the containment internal structures (CIS) stick, the steel containment vessel (SCV) stick and the reactor coolant loop (RCL). The NI stick, representing the coupled auxiliary and shield buildings including the embedded exterior walls and basemat, is fixed at the basemat elevation and is supported horizontally at floor elevations of 82.5' and 100' to represent embedment in Hard Rock. The reactor coolant loop model, RCL, is entirely supported by the CIS model. The CIS model is attached to the NI model at elevation 82.5'. The CIS model is also supported horizontally by the NI model at elevation 100'. The SCV model is attached to the NI model at elevation 100'.

In the modal time history analysis, a uniform time step of 0.005 seconds and a cut-off frequency of 33 hertz are used with the three AP600 input acceleration time histories applied in the three global coordinates. The responses due to the three directions of input are therefore combined by algebraic summation.

In the response spectrum analysis, the "smooth" ground response spectra for AP600 are applied in the three global directions. The cut-off frequency and ZPA are 33 hertz and 0.3 g. The "double sum" modal combination method, as described in the Standard Review Plan, is used including "missing mass" from high frequency modes. Responses due to the three directional inputs are combined by the SRSS method.

2.0 Steel Containment Vessel (SCV)

2.1 Accelerations

For the SCV, accelerations calculated by time history analysis are used in design while those calculated by the response spectrum analysis are used for comparison purpose only. The maximum accelerations from these analyses are compared in SSAR Table 3.7.2-17 (sheet 2 of 3) where it is shown that response accelerations calculated by the response spectrum analysis are generally higher than those calculated by the time history analysis with the maximum difference occurring in the vertical direction at the top elevation of the SCV model.

70 modes are considered in the response spectrum analyses with a cut-off frequency at 33 hertz as shown in SSAR Table 3.7.2-4. The "double sum" modal combination method is used where:

$$R = \left\{ \sum_{k=1}^{N} \sum_{s=1}^{N} \left| R_k \cdot R_s \right| \cdot \varepsilon_{ks} \right\}^{1/2}.$$

Where

$\varepsilon_{ks} = \{ 1 + [(\omega_k' - \omega_s') / (\beta_k' \omega_k + \beta_s' \omega_s)]^2 \}^{-1} > 1.0$,
$\beta_k' = \beta_k + (2 / t_d \omega_k)$
$\omega_k' = \omega_k (1 - \beta_k^2)^{1/2}$
ω_k = frequency of the *k*th mode
β_k = damping of the *k*th mode
t_d = earthquake duration.

From the above equation for the "double sum" modal combination method, in each of the three global input directions, the response is equal to the summation of a total of 70^2 (or 4,900) terms where each of these terms is defined to be positive. This "absolute summation" of a large number of terms can add large conservatism in analyses using this method of combination. The method may be particularly conservative for a model such as the AP600 multiple sticks where there are large differences in the magnitude of the masses. The masses on the NI stick are two orders of magnitude larger than the mass at the top of the SCV stick. The light mass may occur in a number of closely spaced modes with opposite phasing of the light mass. In time history analyses the responses in these modes tend to cancel out whereas the response spectrum analysis combines the maximum responses absolutely.

Both the combined CIS-RCL stick model and the SCV stick model are supported at floor elevation 100'. The support points at elevation 100' are themselves well supported, vertically by walls down to the fixed basemat and horizontally by floor slabs outward to the horizontally supported exterior wall. Stick to stick interactions are therefore expected to be small. For comparison purposes to evaluate the effect of the large nuclear island masses, the SCV was re-analyzed without the NI, CIS and RCL stick models that are above elevation 100'. This was accomplished by re-running the response spectrum analysis using the same multiple stick model with masses above elevation 100' on the other sticks removed while retaining other parameters. This modified model has 10 modes within the 33 hertz cut-off frequency. The number of terms in the "double sum" summation reduces from 4,900 to 100. Results from this modified model show reduced acceleration responses which generally are lower in magnitude than those of the original time history analysis. The vertical acceleration responses at the top of the SCV stick, where maximum differences occur between the two analysis methods, are summarized below.

		Vertical Acceleration Responses (g)		
Node #	elevation	T/H (SSAR)	RSA (SSAR)	RSA (Single stick)
3115	256'	1.49	2.91	1.62
3114	248'	1.20	2.03	1.07

This comparison demonstrates that the smaller number of modes in the modified single stick model reduces significantly the number of terms in the "double sum" summation and thus reduces the conservatism of this combination.

Table 1 compares the maximum responses from the modal time history analysis (from SSAR) with those from the "single modified SCV stick". These comparisons support the use of the accelerations and the floor response spectra obtained from the time history analysis for design of the AP600.

2.2 Member Forces

Maximum member forces are compared in SSAR Table 3.7.2-18 (sheets 2 and 3). In the vertical direction, forces calculated by the response spectrum analysis are consistently higher than those calculated by time history. This is also caused by the conservatism of the "double sum" method as discussed for the accelerations. In the two horizontal directions, the shear forces from the response spectrum analysis are generally in agreement with those by time history analysis at elevations below 200'.

The SCV is designed using the accelerations from the time history analysis discussed in section 2.1 above. The calculated member forces in the SCV stick are not used in design.

3.0 Coupled Auxiliary and Shield Buildings

3.1 Accelerations

The maximum accelerations of the Coupled Auxiliary and Shield Buildings (NI) are compared in SSAR Table 3.7.2-17 (sheet 1 of 3). Accelerations calculated by the modal time history analysis are used in developing the design floor response spectra while those calculated by the response spectrum analysis are used for comparison purpose only. Table 2 shows that response accelerations from the response spectrum analysis are generally in agreement with those from time history analysis except at elevations 200', 220', and 241' for vertical responses and elevations 200', 220', 241', 272', and 284' for east-west responses where accelerations from the response spectrum analysis are higher than those from the time history analysis.

In Table 3, response accelerations at the elevations where the responses differ most are separated by the direction of input. For single direction seismic input, acceleration responses in the input direction compare well between response spectrum and time history analyses. However, large responses occur in directions orthogonal to the input direction in the response spectrum analysis but not in the time history analysis.

Modal properties are shown in Table 4 for modes 49 and 50 for responses in the east-west and vertical directions. For these two closely spaced modes (19.34 hertz and 19.70 hertz), the modal participation factors (Γ) and the modeshapes (Φ) in the global Y (east-west) and Z (vertical) directions are shown with their sign. When calculating the cross terms in these modes, east-west responses due to vertical input and vertical responses due to east-west input, the product of ($\Gamma\Phi$) for modes 49 and 50 have opposite signs and will "cancel" each other if the two modes are combined by algebraic sum. However, the cross terms conservatively add together if the modal combination is based on some form of "absolute" summation such as the "double sum" method.

Both time history and response spectrum analyses were re-run excluding modes 49 and 50 and the response accelerations are compared in Table 5. The comparison of time history responses shows that east-west responses are largely unaffected by inclusion or exclusion of these two modes. The comparison of response spectrum analysis responses shows that east-west responses have meaningful reduction when these two modes are excluded. The east-west responses from the response spectrum analysis excluding modes 49 and 50 are similar to those from the time history analysis except at elevation 200' where the time history acceleration is larger. It is concluded that the differences in the east-west accelerations are caused by the conservatism of the "double sum" combination in the response spectrum analysis. The design accelerations and design floor response spectra in the east-west direction calculated by time history analysis are adequate for AP600 design.

The differences in the vertical response accelerations are also caused by the conservatism of the "double sum" combination of the response spectrum analysis. The vertical responses due to vertical input for modes 49 and 50 have the same sign and do not "cancel" each other even when combined by algebraic summation. As a result the time history results and the response spectrum results show a reduction in response when modes 49 and 50 are excluded. The vertical response due to east-west input for modes 49 and 50 "cancel" out in the time history analysis but not in the response spectrum analysis.

The above comparison has demonstrated that the "absolute" modal combination used in the "double sum" method introduces additional conservatism in the response spectrum analysis responses. The design accelerations and the design floor response spectra, which are calculated by modal time history analysis, are appropriate for the design of the AP600.

3.2 Member Forces

Maximum member forces are compared in SSAR Table 3.7.2-18 (sheet 1). For the Coupled Auxiliary and Shield Buildings (NI) stick model, maximum member forces from the time history analysis were used to develop the SSI factor for member forces from other soil cases. This SSI factor was then used to amplify the results of the finite element fixed base response spectrum analysis to give the final design forces for the AP600. Member forces calculated by response spectrum analysis of the stick model were used for comparison only. Comparison of these maximum member forces in Table 6 shows that axial (vertical) forces from the time history analysis are higher than those from the response spectrum analysis at elevations below elevation 241'. In both horizontal directions, time history and response spectrum analysis forces are close except at sections close to grade where time history forces are higher.

Maximum vertical forces are re-calculated by (1) time history analysis with only vertical input time history applied in the vertical direction, and (2) response spectrum analysis with only the acceleration response spectra of the vertical input time history applied in the vertical direction. Axial forces from the two modal time history analyses, the original which has three directional input and the new time history analysis which has only vertical input, are summarized in Table 7 and compare well. Maximum axial forces from the two re-analyzed cases, both with only vertical input, are also compared in Table 7. They compare well except at sections close to grade at elevation 100' where time history forces are higher.

Tables 6 and 7 show that the maximum member forces calculated by time history compare well with those calculated by response spectrum analysis except at sections close to grade where the time history forces are higher than the response spectrum analysis forces. The member forces are used in the AP600 design to develop the SSI factor using comparable results of the soil structure interaction analyses using SASSI. Since SASSI also calculates the time history of member forces with algebraic summation of the three directions of input, the member forces from the time history analysis are appropriate for use in the AP600 design.

4.0 Containment Internal Structures

4.1 Accelerations

The maximum accelerations of the Containment Internal Structures (CIS) are compared in SSAR Table 3.7.2-17 (sheet 3 of 3). The Containment Internal Structures are nearly rigid in the vertical direction and below elevation 107' in the horizontal directions where the maximum accelerations are close to the ZPA of the input motion. Between elevations 107' and 158', the maximum NS and EW accelerations calculated by response spectrum analysis are higher than those calculated by time history (Table 8).

Table 9 shows response accelerations when the NS and EW input motions are applied separately. The response accelerations in the primary directions (NS response due to NS input, or EW response due to EW input) calculated by the time history analysis are generally higher than those by the response spectrum analysis. In the cross coupled directions (NS response due to EW input, or EW response due to NS input), responses calculated by the response spectrum analysis are approximately twice as high as those calculated by time history. When responses in the primary directions are combined with those from the other direction, the combined responses from the response spectrum analysis exceed those from time history.

Modal properties for selected dominant horizontal modes are presented in Table 10. The term $\Gamma\phi$, the product of the modal participation factor and the modeshape, as shown in the table indicates that (1) this product is generally positive in the dominant modes in the primary directions; and (2) this product is much more evenly distributed into positive and negative in the cross coupled directions. Relative to the time history modal superposition, the use of "absolute" summation in the double sum modal combination of the response spectrum analysis leads to:

- similar magnitudes of the responses in the primary directions since most of the individual terms are already positive;
- additional conservatism in responses in the cross coupled direction since many terms have opposite participation and are changed into positive values before being combined.

This is shown at the bottom of Table 10 where the summation of the $\Gamma\phi$ terms for east-west and north -south response are given from left to right for: (1) algebraic summation of all modes, (2) algebraic sum of modes 34, 41,and 45 only, (3) SRSS of modes 34, 41,and 45 only, and (4) absolute sum of modes 34, 41, and 45 only.

Response acceleration in the primary directions are shown to be similar between the time history and response spectrum analyses (Table 9). The "double sum" modal combination introduces additional conservatism in the response spectrum analysis accelerations in the cross coupled directions. Therefore, the design accelerations and design floor response spectra calculated by the time history analysis are appropriate for use in the AP600 design.

4.2 Member Forces

Maximum member forces are compared in SSAR Table 3.7.2-18 (sheet 4). For the Containment Internal Structures (CIS) stick model, maximum member forces from the response spectrum analysis were used to develop the SSI factor for member forces from other soil cases. This SSI factor was then used to amplify the results of the finite element fixed base response spectrum analysis to give the final design forces for the AP600.

Member forces calculated by time history and response spectrum analyses are compared in Table 11. The CIS is "rigid" in the vertical direction and most of its vertical mass does not participate in the time history analysis with a cut off frequency of 33 hertz. Therefore, axial forces from the time history analysis are much smaller than the axial forces from the response spectrum analysis which include the high frequency "missing mass".

In the response spectrum analysis, the "missing mass" is multiplied by the ZPA of 0.3 g and combined with member forces for modes below 33 hertz by the SRSS method. For comparison with the time history results, the "missing mass" of the CIS is summarized and equivalent forces for 0.3 g static acceleration are calculated in Table 12. These additional forces due to the high frequency "missing mass" are combined with the time history analysis forces by SRSS method. The combined response forces are shown in Table 13 and are similar to those from the response spectrum analysis.

The design member forces for the CIS are developed from the finite element model amplified by the SSI factor. This factor is obtained using the member forces from the stick model results of the response spectrum analysis. These member forces from the stick model are appropriate for the design of the AP600.

5.0 SSAR Revision - Add summary of comparison in subsection 3.7.2.12 as shown below:

The three-dimensional lumped mass fixed base stick model of the nuclear island was analyzed by mode superposition time history analysis and by the response spectrum analysis method for the hard rock site condition. Tables 3.7.2-17, 3.7.2-18, and 3.7.2-19 compare the maximum absolute nodal accelerations, member forces, and moments, respectively. Both analyses considered vibration modes up to 33 hertz. In the response spectrum analyses, the combination of modal responses used the double sum method and included high frequency effects as discussed in subsection 3.7.2.7 and summarized in Table 3.7.2-16. The two methods of analysis give similar results with the response spectrum analysis being generally more conservative. Investigations of the two analyses showed that the conservatism in the response spectrum analyses is due to cross coupling of the directions in the multistick model. The double sum modal combination method used in the response spectrum analyis is very conservative when there are closely spaced modes some of which are out-of-phase.

Table 1: SCV - Maximum response acceleration

Node #	Elev. (feet)	(1) TH, SSAR NS	(1) TH, SSAR EW	(1) TH, SSAR V	(2) RSA NS	(2) RSA EW	(2) RSA V	(3) Ratio (T/H) / (RSA) NS	(3) Ratio (T/H) / (RSA) EW	(3) Ratio (T/H) / (RSA) V
		-------- (g) --------			-------- (g) --------					
3115	256	0.94	1.21	1.49	1.03	1.21	1.62	0.92	1.00	0.92
3114	248	0.90	1.17	1.20	0.98	1.15	1.07	0.92	1.02	1.12
3113	240	0.87	1.13	1.04	0.93	1.10	0.86	0.94	1.03	1.22
3112	229	0.83	1.07	0.84	0.86	1.02	0.64	0.97	1.05	1.32
3111	218	0.78	1.01	0.77	0.79	0.95	0.56	0.98	1.07	1.38
3110	205	0.72	0.93	0.75	0.72	0.86	0.53	1.00	1.08	1.41
3109	190	0.65	0.82	0.70	0.63	0.75	0.48	1.03	1.09	1.45
3108	170	0.56	0.68	0.64	0.53	0.61	0.41	1.06	1.12	1.56
3107	162	0.51	0.62	0.60	0.48	0.55	0.38	1.06	1.13	1.58
3106	144									
3105	132									
3104	116									
3103	112									

Notes: (1) Maximum acceleration calculated by modal time history analyses as reported in SSAR.
(2) Maximum acceleration with single SCV stick.
(3) Ratio of maximum accelerations, time history over response spectrum analyses, (1)/(2) = (3)

Conclusion: From the ratios shown above, most of the maximum accelerations calculated by modal time history analyses are conservative (ratio > 1.0) except at a few places where it is no more than 8% below those calculated by RSA of the modified model (SCV alone).

Table 2: Maximum response accelerations at NI (from SSAR)

Node #	Elev. (feet)	(1) T/H, SSAR NS	(1) T/H, SSAR EW	(1) T/H, SSAR V	(2) RSA, SSAR NS	(2) RSA, SSAR EW	(2) RSA, SSAR V	(3) Ratio (T/H) / (RSA) NS	(3) Ratio (T/H) / (RSA) EW	(3) Ratio (T/H) / (RSA) V
		-------- (g) --------			-------- (g) --------					
3016	306	1.44	1.47	0.90	1.43	1.49	0.88	1.01	0.99	1.02
3015	297	1.32	1.27	0.90	1.31	1.39	0.88	1.01	0.91	1.02
3014	284	1.20	0.98	0.89	1.16	1.25	0.87	1.03	**0.78**	1.02
3013	272	1.09	0.94	0.88	1.09	1.16	0.86	1.00	**0.81**	1.02
3011	241	0.82	0.78	0.55	0.85	0.97	0.75	0.96	**0.80**	**0.73**
3010	220	0.73	0.69	0.53	0.75	0.89	0.70	0.97	**0.78**	**0.76**
3009	200	0.63	0.67	0.49	0.69	0.77	0.62	0.91	**0.87**	**0.79**
3008	180	0.51	0.60	0.45	0.59	0.61	0.47	**0.86**	0.98	0.96
3007	161	0.44	0.54	0.42	0.48	0.56	0.37	0.92	0.96	1.14
3006	153	0.42	0.51	0.40	0.44	0.54	0.33	0.95	0.94	1.21
3005	135	0.38	0.41	0.37	0.33	0.45	0.30	1.15	0.91	1.23
3004	117	0.34	0.34	0.35	0.30	0.30	0.30	1.13	1.13	1.17
3003	100	0.30	0.30	0.32	0.30	0.30	0.30	1.00	1.00	1.07

Table 3 : Maximum Nodal Accelerations due to seismic input in individual direction (T/H vs. RSA)

		Nouth-South Input			East-West Input			Vertical Input			
Node #	Elev. (feet)	NS	EW ft/sec^2	V	NS	EW ft/sec^2	V	NS ft/sec^2	EW	V	
3014	284	----	0.59	----	----	32.75	----	----	4.52	----	
3013	272	----	0.74	----	----	30.30	----	----	3.08	----	**Modal Time History**
3011	241	----	0.77	1.27	----	25.52	1.70	----	2.65	17.26	
3010	220	----	0.84	1.24	----	22.90	1.63	---	1.75	16.49	
3009	200	----	0.94	1.19	----	21.88	1.58	----	1.45	15.44	
3008	180	16.04	----	----	0.43	----	----	0.55	----	----	

Node #	Elev. (feet)	NS	EW ft/sec^2	V	NS	EW ft/sec^2	V	NS ft/sec^2	EW	V	
3014	284	----	2.55	----	----	37.90	----	----	13.01	----	
3013	272	----	3.77	----	----	35.99	----	----	8.91	----	
3011	241	----	4.41	1.93	----	26.76	8.93	----	15.40	22.32	**RSA**
3010	220	----	4.33	1.81	----	23.35	8.42	----	16.01	20.85	
3009	200	----	4.47	1.62	----	21.48	7.49	----	11.60	18.33	
3008	180	18.22	----	----	4.41	----	----	2.32	----	----	

Table 4 : Modal Participation Factors and Modeshapes in the Y & Z (EW & Vertical) directions for Modes 49 & 50.

Mode #	49		50	
Freq.	19.34 hz.		19.70 hz	
Direct.	Y	Z	Y	Z
P.F.	10.25	20.73	8.84	-17.41
Node				
3014	1.80E-02	-4.80E-03	1.40E-02	3.70E-03
3013	1.30E-02	-4.20E-03	1.20E-02	3.30E-03
3011	-1.90E-02	2.90E-02	-2.70E-02	-2.30E-02
3010	-2.10E-02	2.70E-02	-2.90E-02	-2.20E-02
3009	-1.40E-02	2.40E-02	-1.90E-02	-1.90E-02

Table 5a

Node #	Elev. (feet)	(a) T/H -49,50 NS	(a) EW	(a) V	(b) T/H, SSAR NS	(b) EW	(b) V	(c) = (a)/(b) NS	(c) EW	(c) V
		-------- (g) --------			-------- (g) --------					
3014	284		0.99			0.98			1.01	
3013	272		0.96			0.94			1.02	
3011	241		0.77	0.44		0.78	0.55		0.99	0.80
3010	220		0.68	0.42		0.69	0.53		0.99	0.80
3009	200		0.65	0.40		0.67	0.49		0.97	0.82
3008	180	0.51			0.51			1.00		

Table 5b

Node #	Elev. (feet)	(d) RSA -49,50 NS	(d) EW	(d) V	(e) RSA, SSAR NS	(e) EW	(e) V	(f) = (d)/(e) NS	(f) EW	(f) V	(g) = TH(a) / RSA(d) NS	(g) EW	(g) V
		-------- (g) --------			-------- (g) --------								
3014	284		1.10			1.25			0.88			0.90	
3013	272		0.98			1.16			0.85			0.97	
3011	241		0.77	0.27		0.97	0.75		0.80	0.36		1.00	1.63
3010	220		0.60	0.25		0.89	0.70		0.68	0.35		1.13	1.72
3009	200		0.48	0.21		0.77	0.62		0.62	0.34		1.36	1.92
3008	180	0.58			0.59			0.98			0.88		

Table 5a, comparison of max EW accelerations from Modal Time History analyses with & without modes 49 and 50. The comparison shows "no change" in EW acceleration.

Table 5b, (1)comparison of max EW accelerations from RSA with & without modes 49 and 50, and
(2)comparison of EW acceleraions from T/H and RSA, both without modes 49 and 50.

The comparisons show reduced EW acceleration in RSA when modes 49 & 50 are excluded. These nodal accelerations become comparable to those calculated by Modal Time History analyses, with and without modes 49 and 50. This shows the conservatism built into the "double sum" modal combination method as stated in SRP where all of the terms combined used absolute summation.

Table 6: NI - Maximum member forces

Element No.		Elev.	(1) TH, SSAR			(2) RSA, SSAR			(3) Ratio (TH1 / RSA2)		
			Axial	NS	EW	Axial	NS	EW	Axial	NS	EW
gp #2	gp #1	(feet)	---- (1,000 kips) ----			---- (1,000 kips) ----					
		306									
---	14		1.45	2.46	2.43	1.48	2.40	2.51	0.98	1.03	0.97
		297									
---	13		3.40	4.47	4.36	3.46	4.40	4.63	0.98	1.02	0.94
		284									
---	12		7.65	8.30	7.67	7.76	8.26	8.80	0.99	1.00	0.87
		272									
---	11		11.54	12.52	10.57	11.66	12.39	13.22	0.99	1.01	0.80
		241									
---	10		15.44	16.43	15.68	12.78	16.41	17.03	1.21	1.00	0.92
		220									
---	9		18.05	18.72	18.32	14.24	18.64	19.44	1.27	1.00	0.94
		200									
8	8		20.43	20.68	20.32	15.77	20.21	21.32	1.30	1.02	0.95
		180									
7	7		23.40	23.28	23.03	17.62	22.11	23.18	1.33	1.05	0.99
		161									
6	6		25.45	25.51	25.17	18.90	23.32	24.48	1.35	1.09	1.03
		153									
5	5		28.14	28.82	28.40	20.80	25.11	26.57	1.35	1.15	1.07
		135									
4	4		31.92	34.03	33.57	23.54	27.82	29.96	1.36	1.22	1.12
		117									
3	3		34.96	37.54	37.59	26.04	29.79	32.85	1.34	1.26	1.14
		100									

Note: Maximum forces from SSAR (1) by TH analyses, and (2) by RSA, and .
(3) Ratio of (TH forces) / (RSA forces)

Table 7: NI - Maximum axial forces

Element No. gp #2	Element No. gp #1	Elev. (feet)	(a) TH Axial (1,000 kips)	(b) TH Axial (1,000 kips)	(c) ratio (a) / (b)	(d) RSA Axial (1000k)	(e) ratio (b) / (d)
		306					
---	14		1.45	1.50	0.97	1.60	0.94
		297					
---	13		3.40	3.50	0.97	3.73	0.94
		284					
---	12		7.65	7.84	0.98	8.36	0.94
		272					
---	11		11.54	11.80	0.98	12.57	0.94
		241					
---	10		15.44	13.72	1.13	13.67	1.00
		220					
---	9		18.05	15.58	1.16	14.99	1.04
		200					
8	8		20.43	17.51	1.17	16.35	1.07
		180					
7	7		23.40	20.58	1.14	18.02	1.14
		161					
6	6		25.45	23.13	1.10	19.17	1.21
		153					
5	5		28.14	26.94	1.04	20.86	1.29
		135					
4	4		31.92	33.10	0.96	23.01	1.44
		117					
3	3		34.96	38.62	0.91	25.64	1.51
		100					

Note: Maximum member forces

(a) Axial forces calculated by modal time history analysis, taken from SSAR.
(b) modal time history analysis with only vertical input time history
(c) Ratio of max time history forces, 3 directional input vs. vertical input. (c) =(a) / (b)
(d) response spectrum analysis, input only ARS of vertical input time history
(e) ratio of max acceleration with only vertical input, (e) = TH(b) / RSA(d)

Table 8: Maximum Response Accelerations at CIS

Node No	Elev. (feet)	(1) TH, SSAR NS	(1) TH, SSAR EW	(1) TH, SSAR V	(2) RSA, SSAR NS	(2) RSA, SSAR EW	(2) RSA, SSAR V	(3) Ratio, TH(1) / RSA(2) NS	(3) Ratio, TH(1) / RSA(2) EW	(3) Ratio, TH(1) / RSA(2) V
		---- g ----			---- g ----					
		West SG Compartment								
3207	158	0.79	0.65	0.30	0.84	0.82	0.30	0.94	0.79	1.00
3206	148	0.73	0.58	0.31	0.76	0.70	0.30	0.96	0.83	1.03
----	135									
		East SG Compartment								
3205	148	0.69	0.54	0.32	0.77	0.67	0.30	0.90	0.81	1.07
----	135									
		Below Elevation 135								
3204	135	0.61	0.52	0.30	0.53	0.57	0.30	1.15	0.91	1.00
3203	107	0.32	0.30	0.30	0.30	0.30	0.30	1.07	1.00	1.00
3202	103	0.31	0.30	0.30	0.30	0.30	0.30	1.03	1.00	1.00
	100	0.30	0.30	0.30						

Note: Maximum response accelerations in (1) and (2) are taken from SSAR.
Response acceleration in the vertical and response acceleration below elevation 135' are closed to the zpa of 0.30 g which reflects "rigid" condition.

Table 9: Maximum Accelerations at CIS (single direction input motion)

Node No.	Elev. (feet)	(4) NS input, TH NS	(4) NS input, TH EW	(5) EW input, TH NS	(5) EW input, TH EW	(6) NS input, RSA NS	(6) NS input, RSA EW	(7) EW input, RSA NS	(7) EW input, RSA EW
		----g ----		----g ----		----g ----		----g ----	
3207	158	0.76	0.20	0.26	0.68	0.65	0.40	0.56	0.71
3206	148	0.70	0.19	0.25	0.61	0.57	0.40	0.51	0.58
3205	148	0.67	0.18	0.27	0.56	0.68	0.43	0.37	0.51
3204	135	0.60	0.15	0.14	0.54	0.44	0.34	0.32	0.46

Node No.	Elev. (feet)	(8) Ratio, TH/RSA NS/NS (4)/(6)	(8) Ratio, TH/RSA EW/EW (5)/(7)	(9) Ratio, TH/RSA EW/NS (4)/(6)	(9) Ratio, TH/RSA NS/EW (5)/(7)	SRSS(NS,EW) input (10) TH NS	SRSS(NS,EW) input (10) TH EW	SRSS(NS,EW) input (11) RSA NS	SRSS(NS,EW) input (11) RSA EW	Ratio of SRSS (12) = (10) / (11) NS	Ratio of SRSS (12) = (10) / (11) EW
		(direct. of input)		(90 from input)							
3207	158	1.17	0.96	0.51	0.47	0.80	0.71	0.86	0.82	0.94	0.87
3206	148	1.22	1.05	0.46	0.48	0.74	0.63	0.77	0.71	0.97	0.90
3205	148	0.99	1.09	0.43	0.71	0.72	0.59	0.77	0.67	0.93	0.88
3204	135	1.36	1.17	0.45	0.43	0.61	0.56	0.54	0.57	1.13	0.98

Notes: Maximum accelerations due to NS Time History input and EW Time History input are shown in (4) and (5), while those due to NS ARS input and EW ARS input and calculated by RSA are shown in (6) and (7).
(8) = ratio of maximum acceleration due to input in the same direction, (TH responses)/(RSA responses)
(9) = ratio of maximum acceleration due to input 90 degrees away.
(10), and (11) = square root of sum of squares of responses due to NS and EW input for TH and RSA, respectively. For RSA, SRSS is the exact method of combination; for TH, SRSS gives an approximation.

Table 10: Modal Properties at CIS

Mode No.		33		34		36		37		39		41		45	
Freq (cps)		12.85		12.99		13.64		14.36		14.846		15.103		17.126	
Modal Part. F.		-1.15	-25.57	-4.94	9.50	7.52	-0.76	-6.66	0.50	-5.40	-3.72	9.52	3.01	11.30	-4.78
Node	Elev.	NS	EW	NS	EW	NS	EW	NS	EW	NS	EW	NS	EW	NS	EW
No.	(feet)	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ	ϕ
3204	135	.0014	-.0050	-.0137	.0413	.0022	-.0002	-.0059	-.0005	-.0106	-.0129	.0336	.0264	.0335	-.0148
3205	148	-.0025	-.0053	.0227	.0424	.0034	-.0004	-.0076	-.0005	-.0001	-.0162	.0097	.0338	.0914	-.0229
3206	148	.0031	-.0065	-.0304	.0530	.0023	-.0002	-.0071	-.0007	-.0178	-.0153	.0533	.0312	.0260	-.0155
3207	158	.0032	-.0083	-.0310	.0693	.0026	-.0001	-.0080	-.0090	-.0195	-.0149	.0590	.0294	.0341	-.0073

$\Gamma\phi$ (NS=NS responses due to NS input, EW=EW response due to EW input)

3204	*(.0016)*	*.1284*	*.0677*	*.3925*	*.0166*	*.0002*	*.0393*	*(.0003)*	*.0574*	*.0479*	*.3203*	*.2111*	*.3779*	*.0705*
3205	*.0029*	*.1348*	*(.1123)*	*.4027*	*.0259*	*.0003*	*.0506*	*(.0003)*	*.0007*	*.0602*	*.0919*	*.2711*	*1.0321*	*.1094*
3206	*(.0036)*	*.1649*	*.1504*	*.5038*	*.0172*	*.0002*	*.0470*	*(.0003)*	*.0961*	*.0570*	*.5078*	*.2501*	*.2932*	*.0743*
3207	*(.0036)*	*.2130*	*.1531*	*.6586*	*.0195*	*.0000*	*.0536*	*(.0045)*	*.1052*	*.0556*	*.5614*	*.2353*	*.3856*	*.0348*

$\Gamma\phi$ (NS=NS responses due to EW input, EW=EW response due to NS input)

3204	*(.0350)*	*.0058*	*(.1301)*	*(.2041)*	*(.0017)*	*(.0019)*	*(.0030)*	*.0033*	*.0395*	*.0695*	*.2695*	*.2509*	*(.1598)*	*(.1668)*
3205	*.0649*	*.0061*	*.2158*	*(.2095)*	*(.0026)*	*(.0029)*	*(.0038)*	*.0034*	*.0005*	*.0874*	*.0773*	*.3222*	*(.4363)*	*(.2588)*
3206	*(.0802)*	*.0074*	*(.2892)*	*(.2621)*	*(.0017)*	*(.0017)*	*(.0035)*	*.0045*	*.0662*	*.0827*	*.4272*	*.2972*	*(.1240)*	*(.1756)*
3207	*(.0811)*	*.0096*	*(.2944)*	*(.3425)*	*(.0020)*	*(.0004)*	*(.0040)*	*.0600*	*.0724*	*.0807*	*.4723*	*.2797*	*(.1630)*	*(.0823)*

$\Sigma(\Gamma\phi)$:

	NS	EW	Modes 34, 41, 45 only			Modes 34, 41, 45 (SRSS)			Modes 34, 41, 45 (ABS)		
3204	.8776	.8502	3204	.7658	.6741	3204	.5000	.4512	3204	7658	6741
3205	1.0919	.9782	3205	1.0117	.7832	3205	1.0423	.4976	3205	1.2362	.7832
3206	1.1081	1.0499	3206	.9515	.8282	3206	.6054	.5674	3206	.9515	.8282
3207	1.2748	1.1928	3207	1.1002	.9287	3207	.6981	.7002	3207	1.1002	.9287
3204	(.0206)	(.0434)	3204	(.0204)	(.1201)	3204	.3392	.3639	3204	5593	.6218
3205	(.0842)	(.0520)	3205	(.1432)	(.1461)	3205	.4929	.4633	3205	7295	.7904
3206	(.0052)	(.0475)	3206	.0141	(.1405)	3206	.5306	.4334	3206	.8404	.7349
3207	.0002	.0047	3207	.0149	(.1451)	3207	.5800	.4498	3207	9298	7045

(Γ= Modal Participation Factor)

Table 11: Maximum Member Forces at CIS

ment No. gp #4	gp #3	Elev. (feet)	(1) TH, SSAR Axial	NS	EW	(2) RSA, SSAR Axial	NS	EW	(3) Ratio (RSA2 / TH1) Axial	NS	EW
			---- (1,000 kips) ----			---- (1,000 kips) ----					
			West SG Compartment								
		158									
8	11		0.02	0.16	0.13	0.05	0.16	0.15	2.50	1.00	1.15
		153									
8	10		0.02	0.28	0.22	0.05	0.29	0.28	2.50	1.04	1.27
		148									
7	9		0.08	0.81	0.65	0.24	0.81	0.76	3.00	1.00	1.17
		135									
			East SG Compartment								
		148									
6	8		0.03	0.31	0.24	0.13	0.31	0.27	4.33	1.00	1.13
		135									
			Below Elevation 135								
		135									
5	7		0.32	6.14	6.09	1.99	5.73	5.98	6.22	0.93	0.98
		121									
5	6		0.32	6.24	6.16	1.99	5.83	6.07	6.22	0.93	0.99
		107									
4	5		0.67	7.30	6.34	4.07	7.02	6.90	6.07	0.96	1.09
		103									
3	4		0.86	7.35	6.37	6.55	7.65	7.54	7.62	1.04	1.18
		100									

Note: Maximum acceleration calculated (1) by modal time history analyses and
(2) by Response Spectrum Analysis, both as reported in SSAR.
(3) Ratio of (RSA forces) / (Time History forces), as reported in SSAR.

Table 12: CIS - Equivalent Member Forces for "Missing Mass"

Node No	Element No. gp #4	Element No. gp #3	Elev. (feet)	(1) Model Mass X	(1) Model Mass Y	(1) Model Mass Z	(2) "Missing" Forces @ 0.3 g ZPA X (percent)	Y (percent)	Z (percent)	X (1,000 kips)	Y (1,000 kips)	Z (1,000 kips)	(3) Cumulative, Shear X (1,000 kips)	Y (1,000 kips)	Z (1,000 kips)
				West SG Compartment											
3207			158	5.75	5.75	5.75	----	----	-99.36	0.00	0.00	0.06			
	8	11											0.00	0.00	0.06
----			153	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00			
	8	10											0.00	0.00	0.06
3206			148	21.48	21.48	21.48	----	----	-99.46	0.00	0.00	0.21			
	7	9											0.00	0.00	0.26
----			135												
				East SG Compartment											
3205			148	12.60	12.60	12.60	----	----	-100.00	0.00	0.00	0.12			
	6	8											0.00	0.00	0.12
----			135												
				Below Elevation 135											
3204			135	258.31	258.31	187.31	-0.14	-0.32	-99.77	0.00	0.01	1.81			
	5	7											0.00	0.01	2.19
----			121	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00			
	5	6											0.00	0.01	2.19
3203			107	304.57	304.57	238.62	-94.94	-96.03	-99.99	2.79	2.83	2.30			
	4	5											2.80	2.83	4.49
3202			103	136.46	136.46	273.40	-98.37	-98.76	-99.98	1.30	1.30	2.64			
	3	4											4.09	4.14	7.13
			100	----	----	----									
			total =	739.17	739.17	739.16									

Notes: (1) = Mass at CIS stick model.

(2) = Percent of nodal mass not already considered at cut-off frequency of 33 hertz, and force equivalent of "missing mass" at acceleration of 0.30 g.

(3) = Cumulative of "missing" force = equivalent shear forces.

Table 13: CIS - Maximum Member Forces, Adjusted for "Missing Mass".

Node No.	Element No. gp #4	Element No. gp #3	Elev. (feet)	(4) TH, SSAR Axial	(4) TH, SSAR NS	(4) TH, SSAR EW	(5) SRSS {(TH,SSAR (4)), + (Missing Mass (3))} Axial	(5) NS	(5) EW	(6) RSA, SSAR Axial	(6) NS	(6) EW	(7) Ratio (6) / (5) Axial	(7) NS	(7) EW
				---- (1,000 kips) ----			---- (1,000 kips) ----			---- (1,000 kips) ----					
				West SG Compartment											
3207			158												
	8	11		0.02	0.16	0.13	0.06	0.16	0.13	0.05	0.16	0.15	0.85	1.00	1.15
----			153												
	8	10		0.02	0.28	0.22	0.06	0.28	0.22	0.05	0.29	0.28	0.85	1.04	1.27
3206			148												
	7	9		0.08	0.81	0.65	0.27	0.81	0.65	0.24	0.81	0.76	0.88	1.00	1.17
----			135												
				East SG Compartment											
3205			148												
	6	8		0.03	0.31	0.24	0.13	0.31	0.24	0.13	0.31	0.27	1.04	1.00	1.13
----			135												
				Below Elevation 135											
3204			135												
	5	7		0.32	6.14	6.09	2.21	6.14	6.09	1.99	5.73	5.98	0.90	0.93	0.98
----			121												
	5	6		0.32	6.24	6.16	2.21	6.24	6.16	1.99	5.83	6.07	0.90	0.93	0.99
3203			107												
	4	5		0.67	7.30	6.34	4.54	7.82	6.94	4.07	7.02	6.90	0.90	0.90	0.99
3202			103												
	3	4		0.66	7.35	6.37	7.19	8.41	7.59	6.55	7.65	7.54	0.91	0.91	0.99
			100												

FAX to DINO SCALETTI

May 8, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #1885 (DSER 3.7.2.16-1)
#670 (DSER 3.7.2.16-1)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 22 calendar days away (17 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Items #1885 (DSER 3.7.2.16-1) and #670 (DSER 3.7.2.16-1) is attached. Action was completed on these Items by our submittal of letter NTD-NRC-97-5041 on March 26, 1997. Material from this letter was incorporated into Revision 12 of the SSAR. A copy of this letter, with the pertinent attachments, and the affected page from the SSAR are included with this fax. We believe this information completed our action on these open items and request that NRC review the material we have provided and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 1885 And 1885 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1885	NRR/ECGB	3.7.2.16-1	DSER-COL		Orr/SSARREV	Action N	Action W	NSD-NRC-97-4956	

3.7.2.16-1 The COL applicant should perform an analysis and evaluation using the design basis earthquake ground motion and plant specific site conditions to confirm the design adequacy of the AP600 design.

SSAR Subsection 2.5 provides the information requirements for the COL applicant. Site-specific soil structure interaction analyses may be performed by the Combined License applicant to demonstrate acceptability by comparison of floor response spectra. These analyses would use the site specific soil conditions and safe shutdown earthquake.
The COL applicant requirement is included in SSAR Section 2.5.4.5.5.
See open item 3.7.1.1-1 for details and issue B.1 of NRC Letter dated July 18, 1996.

NRC Status: Action W - This COL action is connected to DSER 3.7.1.1-1 (OITS# 628). The current SSAR proposal presented at the December 1996 meeting does not satisfy the staff position. (12/16/96)
Closed - Westinghouse provided a position on shallow soil sites in NSD-NRC-97-4956 of 1/28/97.
Action W - NRC Letter of 1/31/97 did not accept position this item will be closed with item #628
Action-N - Response provided in NSD-NRC-97-5041, March 26, 1997

2 of 7

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/8/97

Selection: [item no] between 670 And 670 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
670	NRR/ECGB	3.7.2.16-1	DSER-OI		Orr/BPC/SSARREV	Closed	Action N	NTD-NRC-95-4433	

Westinghouse should commit in the SSAR that the COL applicant should perform an analysis and evaluation using the design basis earthquake ground motion and plant-specific site conditions to confirm the adequacy of the AP600 design.

SSAR Subsection 2.5.4 provides the information requirements for the COL applicant. Site-specific soil structure interaction analyses may be performed by the Combined License applicant to demonstrate acceptability by comparison of floor response spectra. These analyses would use the site specific soil conditions and safe shutdown earthquake.
NRC Letter dated July 18, 1996, NRC/DCP0525 NRC position is that a minimum 0.3g ground motion should be used at ALL sites regardless of actual conditions. See OI # 3.7.1.1-1.
NRC Status: Action N - The staff will review SSAR 2.5 for an adequate reconciliation evaluation with as-built condition. (12/16/96)
Westinghouse provided a position on shallow soil sites in NSD-NRC-97-4956 of 1/28/97.
Action-N - Response provided in NSD-NRC-97-5041, March 26, 1997

3 of 7

Total Records: 1

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5041
DCP/NRC0788
Docket No.: STN-52-003

March 26, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: SEISMIC AND SITE PARAMETER OPEN ITEMS

Dear Mr. Quay:

Attached are responses for open items related to SSAR section 2.5 and 3.7. Markup of proposed changes to SSAR Sections 2.5 and 3.7 and appendix 3F are also provided. These responses and markups are intended to resolve the identified open items related to these sections.

The changes in site parameters should be consistent with the interpretation of the regulations by the NRC staff. The approach includes information discussed between Westinghouse legal counsel and NRC Office of General Counsel and is similar to the approach previously approved for another ALWR design certification. The changes included in Section 3.7 are generally changes previously discussed with the staff.

This information will be discussed in a meeting between Westinghouse and NRC staff tentatively scheduled for April 14-18, 1997. If you have any questions please contact Donald A. Lindgren at (412) 374-4856.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment

cc: D. Jackson, NRC
N. J. Liparulo, Westinghouse (w/o Attachment)

3129A

4 of 7

The draft SSAR material provides a markup of Chapter 2, Section 3.7 and Appendix 3F. It responds to the following open items:

Open Item #	DSER Question	Response
547	2.5.4.3-2	Short and long term settlement are addressed in a revision to subsection 2.5.4.3.
623	3.7.1-2	SSAR subsection 3.7.1.1 and Appendix 3F are revised as shown in the markup. Damping for cable trays has been revised per agreement in January, 1997 meeting. Damping for configurations not similar to the tests referenced in the SSAR has been reduced to 10% matching the recommendations in a report by Brookhaven.
628	3.7.1.1-1	The shallow soil sites are excluded from the AP600 design basis due to the limitation on soil shear wave velocity. This is specifically shown in a revision to subsection 3.7.1.4.
649	3.7.2.4-7	Design information has been added in subsection 3.7.2.8 for the annex building demonstrating that there will be no unacceptable seismic interaction. The method of evaluation of seismic interaction for the radwaste building is also added in subsection 3.7.2.8.
662	3.7.2.8-5	Subsection 3.7.2.8 has been revised to show use of eccentric bracing or the turbine building.
664	3.7.2.8-7	Turbine building behavior is addressed for seismic margin in PRA Chapter 55. A draft copy was provided with letter NSD-NRC-97-5014, dated March 18, 1997.
670, 1885	3.7.2.16-1	Requirement has been added for COL applicant to reconcile seismic analyses of structures for as-built data.
769	3.8.5-11	SSAR subsection 2.5.4.5 has been revised to provide information on lateral soil variability and the requirements for site evaluation. This is also referenced from 3.8.5 in SSAR Rev 11.
791	3.9.3.1-6	Comments on ductwork were addressed in Appendix 3A in SSAR Rev 11. Comments on electrical raceways are addressed in Appendix 3F.
4997	RAI 231.34	Chapter 2 has been revised to use the term "site parameters".

5 of 7

3.7.5 Combined License Information

3.7.5.1 Seismic Analysis of Dams

Combined License applicants referencing the AP600 certified design will evaluate dams whose failure could affect the site interface flood level specified in subsection 2.4.1.2. The evaluation of the safety of existing and new dams will use the site-specific safe shutdown earthquake.

3.7.5.2 Post-Earthquake Procedures

Combined License applicants referencing the AP600 certified design will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and the cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. The procedures and the data from the seismic instrumentation system will provide sufficient information to guide the operator on a timely basis to determine if the level of earthquake ground motion requiring shutdown has been exceeded. The procedures will follow the guidance of EPRI Reports NP-5930 (Reference 1), TR-100082 (Reference 17), and NP-6695 (Reference 18), as modified by the NRC staff (Reference 32).

3.7.5.3 Seismic Interaction Review

The seismic interaction review will be updated by the Combined License applicant. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition.

3.7.5.4 Reconciliation of Seismic Analyses of Nuclear Island Structures

The Combined License applicant will reconcile the seismic analyses described in subsection 3.7.2 for detail design changes such as those due to as-procured equipment information. If it is necessary to update the soil structure interaction analyses, these analyses should be performed with site specific soil properties using seismic input defined by the response spectra given in Figures 3.7.1-1 and 3.7.1-2.

3.7.6 References

1. EPRI Report NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake," July 1988.

2. Uniform Building Code, 1991.

3. ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers, September 1986.

4. ASME B&PV Code, Code Case N-411.

guide the operator on a timely basis to determine if the level of earthquake ground motion requiring shutdown has been exceeded. The procedures will follow the guidance of EPRI Reports NP-5930 (Reference 1), TR-100082 (Reference 17), and NP-6695 (Reference 18), as modified by the NRC staff (Reference 32).

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1. EPRI Report NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake," July 1988.

2. Uniform Building Code, 1991.

3. ASCE Standard 4-86, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers, September 1986.

4. ASME B&PV Code, Code Case N-411.

5. H. B. Seed, and I. M. Idriss, "Soil Moduli and Damping Factors for Dynamic Response Analysis," Report No. EERC-70-14, Earthquake Engineering Research Center, University of California, Berkeley, 1970.

6. H. B. Seed, R. T. Wong, I. M. Idriss, and K. Tokimatsu, "Moduli and Damping Factors for Dynamic Analysis of Cohesionless Soils," Report No. UCB/EERC-8914, Earthquake Engineering Research Center, University of California, Berkeley, 1984.

7. Bechtel Corporation, "User's and Theoretical Manual for Computer Program BSAP (CE800)," Revision 12, 1991.

8. Bechtel Corporation, "Theoretical, Validation and User's Manuals for Computer Program SASSI (CE994)," 1988.

Post-it® Fax Note 7671

To DIANE JACKSON		From JIM WINTERS	
Co./Dept. US NRC		Co. WESTINGHOUSE	
Phone #		Phone # 412-374-5290	
Fax #		Fax #	
Date 5/13/97	# of pages ▶ 1		

AP600 Open Item Tracking System

Project Status Report

Selection: Full Selection

Status as of : [5/13/97]

Open Item Type	Resolution Status (W / NRC)								
	Dropped	**Confrm-W**	**Confrm-N**	**Audit N**	**Action W**	**Action N**	**Resolved**	**Closed**	**Total**
DSER - OI	0 / 0	16 / 23	12 / 2	2 / 0	77 / 205	21 / 311	5 / 604	1119 / 107	1252 / 1252
DSER - Confirmatory	0 / 0	0 / 1	3 / 0	0 / 0	0 / 22	1 / 6	0 / 49	76 / 2	80 / 80
DSER - COL	0 / 0	0 / 5	0 / 0	0 / 0	0 / 7	1 / 36	0 / 105	164 / 12	165 / 165
DSER - OI50	62 / 62	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	62 / 62
RAI - OI	0 / 0	2 / 3	8 / 2	0 / 0	636 / 929	24 / 529	0 / 182	1229 / 254	1899 / 1899
RAI -Confirmatory	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0	0 / 0
Meeting - OI	1 / 1	3 / 6	4 / 2	0 / 0	208 / 334	19 / 160	1 / 165	639 / 207	875 / 875
Teleconference - OI	0 / 0	0 / 0	4 / 0	0 / 0	7 / 25	3 / 8	0 / 13	41 / 9	55 / 55
Key Issue	0 / 0	0 / 0	0 / 0	0 / 0	35 / 45	2 / 0	0 / 0	8 / 0	45 / 45
Total:	63 / 63	21 / 38	31 / 6	2 / 0	963 / 1567	71 / 1050	6 / 1118	3276 / 591	4433 / 4433

FAX to DINO SCALETTI

May 9, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Bob Vijuk
Brian McIntyre

OPEN ITEM #820 (DSER 3.11.3.2-2)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 21 calendar days away (16 business days). The relevant documentation related to Open Item #820 (DSER 3.11.3.2-2) is in the SSAR in Sub-Section 3D.4.3 (supplied to you some months ago). The pertinent page of the SSAR is attached. It is requested NRC review this material and provide definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed."

Jim

Jim Winters
412-374-5290

1 of 3

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/9/97

Selection: [item no] between 820 And 820 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
820	NRR/SPLB	3.11.3.2-2	DSER-OI	Westinghouse should acceptably address issues pertaining to qualification of electronic equipment. Closed - Subsection 3D.4.3 identifies the radiation limits for mild environment for electronic and other equipment.	Miller	Closed	Action W		

2of3

Total Records: 1

addresses considerations for cable field splices and connections, guidance for their qualification is taken from IEEE 572 and Regulatory Guide 1.156.

Regulatory Guide 1.156, "Environmental Qualification of Connection Assemblies for Nuclear Power Plants" - The guide endorses IEEE 572-1985. The AP600 equipment qualification program employs the recommendations of Regulatory Guide 1.156 in specifying the qualification program plans where this guide supplements the guidance of IEEE 572 to demonstrate conformance with the guidance of IEEE 323.

Regulatory Guide 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants" - The guide endorses IEEE 535-1986. The AP600 equipment qualification program employs the recommendations of Regulatory Guide 1.158 in specifying the qualification program plans where this guide supplements the guidance of IEEE 535 to demonstrate conformance with the guidance of IEEE 323.

3D.4.2 Definitions

Definitions of terms used in this appendix are contained in the referenced standards and IEEE 100, "IEEE Dictionary of Electrical and Electronic Terms." Subsection 3D.4.5 clarifies the definitions of "life" (that is, design, shelf, and qualified life) as used in this methodology. The terms "design life" and "qualified life" have the meanings set forth in IEEE 323 and are used in the context of that standard.

3D.4.3 Mild versus Harsh Environments

Qualification requirements differ for equipment located in mild and harsh environments.

IEEE 323 defines a mild environment as an environment expected as a result of normal service conditions and the extremes of abnormal service conditions where a safe shutdown earthquake is the only design basis event of consequence or conditions where thresholds of material degradation are reached. The following limits are established as the delimiting environmental parameter values for mild and harsh environments.

Typically a mild environment conforms with the environmental parameter limits of Table 3D.4-1, though others may apply to specific equipment applications or locations.

The scope of 10 CFR 50.49 is limited exclusively to equipment located in a harsh environment. The AP600 equipment qualification program conforms with the requirements of 10 CFR 50.49 for the qualification of harsh environment equipment. The "radiation-harsh" environment is a significant subset of the harsh environment category. A radiation-harsh environment is defined for equipment designed to operate above certain radiation thresholds where other environmental parameters remain bounded by normal or abnormal conditions. Any equipment that is above 10^4 rads gamma (10^3 for electronics) will be evaluated to determine if a sequential test which includes aging, radiation, and the applicable seismic event is required or if sufficient documentation exists to preclude such a test.

Westinghouse FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MAY 12 1997	NAME:	Jim Winters
TO:	Diane Jackson	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:	8-1-301-415-2300		

Cover + Pages 1 + 2

The following pages are being se.. from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

Diane,

This provides a trail to the resolution of Open Item 363. It references Open Item 358 which was answered by letter NSO-NRC-97-5089 of April 25. We will now change W Status for 363 to "Action N" and await your direction to change NRC Status.

Jim

cc: Lindgren
McIntyre
Hutchings
Chase
Winters

AP600 Open Item Tracking System Database: Project Management Summary

Date: 5/12/97

Selection: [item no] between 363 And 363 Sorted by Item #

Item No.	Branch	DSER Section Question	Type	Coord/Resp Engineer Title/Description Issue Closure Path Status Detail	Res Est (hrs)	(W) Status	NRC Status	Schedule ICP	Schedule Draft	Schedule Review	Schedule Transmit
363	NRR/SPLB	10.2	MTG-OI	Lindgren, D. / Winters, J.	1	Action W	Action W		A	5/23/95 S	

M10.2-6 (TURBINE GENERATOR DESIGN BASIS) The responses to Q410.139, Q410.143, and Q410.144 were received after the DSER was prepared, and are under staff review. Open items and questions may be developed as a result of the review of those responses.

This item is included in DSER open item 10.2.10-1 (OITS 1142)
DISCUSSED AT 12/13/94 MEETING BETWEEN WESTINGHOUSE AND NRC PLANT SYSTEMS BRANCH. SEE NRC MEETING SUMMARY FOR SPECIFICS SUPPORTING RESULTING STATUS.
4/28/95 status: Will be closed when Item Nos. 358, 359, 360, and 361 are closed.

Closed(Q410.139) - The staff found response to RAI 410.139, TG / RCS compatability, acceptable per 12/13/94 meeting. (No SSAR revision required.)
Closed(Q410.144) - The staff found response to RAI 410.144, compliance with URD, acceptable per 12/13/94 meeting. (No SSAR revision required.)
Closed(Q410.143) - The staff found response to RAI 410.143, compliance with Standard Review Plan, partially acceptable per 12/13/94 meeting. Remaining concerns are detailed in meeting items M10.2-1 through M10.2-4 (OITS 358, 359, 360, 361)

NRC - Action W - correct table; Ch. 10 and Ch. 4 are inconsistent on MWth.

Closed - NSSS power, as shown in SSAR Chapter 10, includes core heat + RCP heat added - RCS heat loss. SSAR Chapter 4 covers core design. In general, the core design should only discuss the 1933 Mwt rating. SSAR Chapter 10 discusses steam and power conversion and it is appropriate to use the NSSS rating of 1940 Mwt. The use of 1933 in Chapter 4 and 1940 in Chapter 10 is consistent. Correcting the table in Chapter 4 would be incorrect. This is the basis for the Non-ILOCA analyses for Chapter 15 and for preparing Chapter 4 and in the core design.

Action W - Pending resolution of OITS# 358.

AP600 Open Item Tracking System Database: Project Management Summary

Date: 5/12/97

Selection: [item no] between 358 And 358 Sorted by Item #

Item No.	Branch	DSER Section Question	Type	Coord/Resp Engineer / Title/Description / Issue Closure Path / Status Detail	Res Est (hrs)	(W) Status	NRC Status	Schedule ICP	Schedule Draft	Schedule Review	Schedule Transmit
358	NRR/SPLB	10.2	MTG-OI	Lindgren, D. / Winters, J.	1	Action N	Action W		A	5/23/95	S

M10.2-1 (TURBINE OVERSPEED TRIP) The AP600 turbine generator does not have a mechanical overspeed trip device as described in SRP Section 10.2, Paragraph III.2.c. The applicant should provide the bases for not having a mechanical overspeed trip device. Specifically, the concern of diversity and common mode failure needs to be addressed.

This item is included in DSER open item 10.2.4-1 (OITS 1134)
The AP600 SSAR references an electronic overspeed trip for the MTS in Section 10.2.2.3.2. This trip system includes 2 out of 3 logic for individual analog speed signals, redundant channels for the speed control unit, and a turbine trip if both channels fail. This represents different technology than referenced in the SRP, but reflects modern controls and provides redundant features to prevent turbine overspeed.
The electronic trip syst [illegible] provide reliability equal or better than systems described in the SRP.
A design review meeti[illegible] the turbine overspeed protective system has been completed. An electronic overspeed trip mechanism is as reliable as a mechanical trip during operation and mo[illegible] reliable during testing. SSAR revision will address redundancy and common mode failure of components.

Closed - SSAR revision (10.2.2.5.3) provides requested justification for electronic trip devices per 12/13/94 meeting agreement.
Related item, Closed - Concerns related to turbine missiles (OITS 2030) were discussed at Westinghouse/NRC Senior Management Meetings.

NRC - Action W - Will send qualitative discussion and demonstrate quantitatively that diversity is equal or better when calculation is complete.
Action N - Response provided by NSD-NRC-97-5089 of 4/25/97. jww

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MAY 12, 1997	NAME:	James Winters
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 3

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

DIANE

THIS MARKUP SHOULD RESOLVE OPEN ITEM 3967. THIS IS THE SAME AS NEW ITEM 11-5-2 FROM OUR 11/5/96 TELECONFERENCE. IT WILL GO INTO SAR REV. 13 UNLESS WE HEAR FROM YOU.

cc: LINDGREN
MCINTYRE
RONVITUIC
CUMMINS
WINTERS
HUTCHINGS
JEANNE EVANS

Table 9.3.1-2 provides design information for the main components associated with the instrument air subsystem.

Service Air Subsystem

The service air subsystem consists of two air compressor trains. Each compressor train consists of a multistage, low-pressure, rotary screw, air compressor package, and a desiccant dryer with a prefilter and afterfilter. A common air receiver is provided for the two trains. Each compressor package includes an intake filter, rotary screw compressor elements, silencer, intercooler, aftercooler, moisture separators, bleed-off cooler, oil cooler, oil reservoir, automatic load controls, relief valves, and a discharge air check valve. Each compressor train produces oil-free air.

The common service air receiver functions as a storage device for compressed air. This air receiver is equipped with an automatic condensate drain valve and a pressure relief valve.

Two air dryer assemblies are provided for the service air subsystem. Each dryer assembly consists of a desiccant-filled, twin tower design. One tower may be used to dry air while the other tower goes through regeneration. When instrumentation senses a high dew point, the towers switch. The former operating tower then undergoes regeneration while the regenerated tower drys the service air.

Each dryer assembly includes a coalescing prefilter that removes oil aerosols and moisture droplets, as well as an afterfilter to remove desiccant dust.

Table 9.3.1-3 provides design information for the main components associated with the service air subsystem.

High-Pressure Air Subsystem

The high-pressure air subsystem utilizes an air-cooled, oil-lubricated, four-stage, reciprocating-air compressor with an integral air purification system to produce oil-free air for high-pressure applications. The compressor train includes an intake filter, air-cooled intercoolers, interstage oil/water separators, an air-cooled aftercooler, a final oil/water separator, relief valves, an air purification system, discharge check valves, and a high-pressure receiver.

The high-pressure air subsystem supplies ANSI/CGA G-7.1 Quality Verification Level E air. See Table 9.3.1-4 for the design parameters for this system. 6

INSERT A

Breathing air of a Quality Verification Level E is supplied from the integral high-pressure air purification system in accordance with the requirements of ANSI/CGA G-7.1. This integral air purification system utilizes a series of replaceable cartridge-type filters to produce breathing quality air. Carbon monoxide is controlled by a catalytic conversion to carbon dioxide within the package.

The onsite standby power system (diesel generators) provides an alternate source of electrical power for the high-pressure air compressor.

9.3.1.3 Safety Evaluation

The compressed and instrument air system has no safety-related function other than containment isolation and therefore requires no nuclear safety evaluation. Containment isolation functions are described in subsection 6.2.3.

The compressed and instrument air system is required for normal operation and startup of the plant. Air-operated valves that are essential for safe shutdown and accident mitigation are designed to actuate to the fail-safe position upon loss of air pressure. These air-operated valves utilize safety-related solenoid valves to control the air supply.

The instrument and service air subsystems are classified as moderate-energy systems. There are no adverse effects on safety-related components associated with a postulated failure of the instrument and service air piping.

The high-pressure air subsystem is classified as a high-energy system. The high-pressure compressor and receiver are located in the turbine building, which contains no safety-related, equipment or structures. Air piping routed in safety-related areas is 1 inch or less in diameter and the dynamic consequences of a rupture are not required to be analyzed. The high-pressure air subsystem is not required to operate following a design basis accident nor is it used for safe shutdown of the plant.

9.3.1.4 Tests and Inspections

System components, such as the air compressors and air dryers, are inspected or tested prior to installation. The installed compressed air system is inspected, tested, and operated to verify that it meets its performance requirements, including operational sequences and alarm functions.

Air compressors and associated components on standby are checked and operated periodically. Desiccant in the air dryers is changed when required.

Sample points are provided downstream of the air dryers in both the instrument and service air subsystems and downstream of the purifier in the high-pressure air subsystem. Periodic checks are made to ensure high quality instrument air as specified in the ANSI/ISA S-7.3 standard.

INSERT A

Breathing air connections of the high pressure air subsystem are incompatible with the breathing air connections of the service air subsystem.

FAX to DINO SCALETTI

May 12, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Gene Piplica
Bob Vijuk
Brian McIntyre

OPEN ITEM #2271 (DSER 15.)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 19 calendar days away (15 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #2271 (DSER 15.) is attached. Action was completed on this Item by our submittal of letter NTD-NRC-96-4800 on August 13, 1996. This letter/answer references Sub-Section 14.2.10.3.6 of the SSAR which describes natural circulation testing. A copy of this letter, with the pertinent attachments, is included with this fax, together with a copy of the referenced Sub-Section of the SSAR. We request that NRC review the material we have attached and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim

Jim Winters
412-374-5290

1 of 6

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/12/97

Selection: [item no] between 2271 And 2271 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2271	NRR/SRXB	15.	MTG-OI	APKIL 19, 1995 (HSII) DISCUSSION ITEMS 14. Reactor Vessel Head Vent SSAR Chapter 15): d. Branch Technical Position RSB 5-1 in SRP Section 5.4.7 states that the initial test program should include tests with supporting analysis to confirm that (1) adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions, and (2) the cooldown under NC conditions can be achieved within the limits specified in the emergency operating procedures. Discuss how the AP600 satisfies this position. Closed - Response provided via Westinghouse letter NSD-NRC-96-4800, dated August 13, 1996	Chapter 14	Closed	Action W		

2 of 2

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4800
DCP/NRC0580
Docket No.: STN-52-003

August 13, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: SSAR CHAPTER 14 - INITIAL TEST PROGRAM, RESPONSES TO RAIS AND OPEN ITEMS

Dear Mr. Quay:

The enclosure to this letter provides responses to requests for additional information and open items contained in the November 1994 Draft Safety Evaluation Report on the AP600. These responses reflect the recent revision to Chapter 14 of the AP600 SSAR (Revision 9). Attachment 1 identifies the open items and RAIs addressed by this transmittal. We request that these responses be included in the ongoing review of the Chapter 14 revision.

Please contact John C. Butler on (412) 374-5268 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Enclosures
Attachment

cc: T. Kenyon, NRC (w/o Enclosures/Attachments)
J. Sebrosky, NRC (1A, 1E)
J. Peralta, NRC (1A, 1E)
R. Gruel, PNL (1A, 1E)
N. J. Liparulo, Westinghouse (w/o Enclosures/Attachments)

2894A

3 of 6

OITS 2271

Re: SSAR Chapter 14, Sections 14.2.10.3.6 & 14.2.10.4.12

APRIL 19, 1995 (HSII) DISCUSSION ITEMS

14. Reactor Vessel Head Vent SSAR Chapter 15):
   d. Branch Technical Position RSB 5-1 in SRP Section 5.4.7 states that the initial test program should include tests with supporting analysis to confirm that (1) adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions, and (2) the cooldown under NC conditions can be achieved within the limits specified in the emergency operating procedures. Discuss how the AP600 satisfies this position.

Response:

Subsection 14.2.10.3.6 describes natural circulation testing, which includes simulation of reactor decay heat and specifies that data characterizing the plant under natural circulating conditions be obtained. This data is sufficient to support analysis to confirm that adequate mixing of borated water can be achieved under natural circulation conditions.

The capability to cooldown the reactor using active systems is demonstrated during the hot functional portion of preoperational testing of the reactor coolant system, steam generator system, startup feedwater system, normal residual heat removal system, and plant cooling water systems; prior to fuel loading. This testing verifies that each component performs in accordance with its design requirements. Additional testing of the capability to remove heat via the steam generators in a controlled manner is described in subsection 14.2.10.4.12, steam dump control system testing. These component performance capabilities are part of the calculational basis for the cooldown limits which are specified in the operating procedures.

Westinghouse Status: Closed

Prerequisites

- The reactor is critical and the neutron flux level is within the range for low-power physics testing
- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- Instrumentation and equipment used to measure and compute reactivity is installed and operational, with input flux signals representative of the core average neutron flux level

Test Method

- One of the following methods will be used to measure the worth of all of the individual control rod banks:
  - A bank is stepwise inserted into the core from fully withdrawn and the worth is measured using the reactivity computer
  - Exchange bank with another bank measured as above, with the worth determined from the critical positions and the worth of the reference bank

Performance Criteria

- The measured value for the individual bank worth is consistent with the design value within specified limits
- The sum of the measured bank worth is consistent with the design value within the assumed uncertainty used in the shutdown margin calculation

14.2.10.3.6 Natural Circulation (First Plant Only)

Objective

Demonstrate that core decay heat can be removed by the steam generators under the conditions of natural circulation (no reactor coolant pumps operating).

Prerequisites

- The reactor is critical, and the neutron flux level is within the range for low-power physics testing
- The neutron flux level and reactor coolant system boron concentration and temperature are stable, and the controlling rod bank is positioned in such a way that an increase in core power level to approximately 3 percent can be achieved by rod motion alone

- Reactor coolant pumps are operating

- The reactivity computer is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level

- Instrumentation and data collection equipment is operational and available for logging plant data

- Special instrumentation is available to measure vessel ΔT with high precision at low-power levels

Test Method

- Because this test is performed at beginning of life when the core fission product density is low, decay heat is simulated by reactor power

- By control rod motion, increase reactor power to approximately 3 percent of full power based on predictions of vessel ΔT at full power

- With reactor coolant pumps running, obtain data for correlating nuclear flux level and loop temperatures with power

- Trip all reactor coolant pumps. Maintain core power at approximately 3 percent by control rod motion while cold leg temperatures remain relatively constant.

- Verify natural circulation by observing the response of the hot leg temperature in each loop. The plant is stable under natural circulation at this power level when hot leg temperature is constant.

- Obtain data characterizing the plant under natural circulation conditions

- Restart reactor coolant pumps only after the reactor is shut down and isothermal conditions are re-established

Performance Criterion

The measured average vessel ΔT under natural circulation conditions is equal to or less than limiting design predictions for the measured reactor power level.

14.2.10.4 Power Ascension Tests

After low power testing is completed, testing is performed at specified elevated power levels to demonstrate the facility operates in accordance with design during normal steady-state operations, and to the extent practical, during and following anticipated transients. During power ascension, tests are performed to obtain operational data and to demonstrate the operational capabilities of the plant.

FAX to DINO SCALETTI

May 12, 1997

CC: Sharon or Dino, please make copies for: D. Jackson
Ted Quay

Don Lindgren
Richard Orr
Bob Vijuk
Brian McIntyre

OPEN ITEM #1792 (DSER 3.9.2.1-4)

To meet the SECY-97-051 schedule of "Applicant Submits Final SSAR Revisions & Documentation" by 5/97, we believe that NRC must acknowledge receipt of all Westinghouse submittals by May 30, 1997. This is just 19 calendar days away (15 business days). In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I have been providing background packages for open items that we believe are complete. Relevant documentation related to Open Item #1792 (DSER 3.9.2.1-4) is attached. Action was completed on this Item by our submittal of letters NTD-NRC-96-4800 dated August 13,1996 and NTD_NRC-4857 dated October 23,1996. Additionally, the material has been incorporated into the SSAR, Sub-Section 14.2.9.1.7. A copy of these letters, with the pertinent attachments, is included with this fax, together with the a copy of Sub-Section 14.2.9.1.7 of the SSAR. We request that NRC review the material we have attached and provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed." Thank you.

Jim Winters
412-374-5290

1 of 10

AP600 Open Item Tracking System Database: Executive Summary

Date: 5/12/97

Selection: [item no] between 1792 And 1792 Sorted by Type

Item No.	Branch	DSER Section Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
1792	NRR/EMEB	3.9.2.1-4	DSER-CN		Wills	Confirm-N	Action W	NSD-NRC-96-4857

3.9.2.1-4 Westinghouse should revise the SSAR as noted in Section 3.9.2.1 of this report. (Incorporate SSAR revision from RAI 210.57, acceptance standard for alternating stress intensity.)

Closed - Response provided via Westinghouse letter NSD-NRC-96-4800, dated August 13, 1996.
Action W - The Draft Revision 9 for Chapter 14 replaced Section 14.2.8.1.78 with Section 14.2.9.1.7. However the response to RAI# 210.57 does not appear to be in this Draft.
Resolved - Letter NSD-NRC-96-4857, dated October 23, 1996 provided a draft SSAR revision (14.2.9.1.7) to resolve this issue.
Confirm-N - SSAR Revision 10 included changes required to address this issue.

2 of 10

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4800
DCP/NRC0580
Docket No.: STN-52-003

August 13, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: SSAR CHAPTER 14 - INITIAL TEST PROGRAM, RESPONSES TO RAIS AND OPEN ITEMS

Dear Mr. Quay:

The enclosure to this letter provides responses to requests for additional information and open items contained in the November 1994 Draft Safety Evaluation Report on the AP600. These responses reflect the recent revision to Chapter 14 of the AP600 SSAR (Revision 9). Attachment 1 identifies the open items and RAIs addressed by this transmittal. We request that these responses be included in the ongoing review of the Chapter 14 revision.

Please contact John C. Butler on (412) 374-5268 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Enclosures
Attachment

cc: T. Kenyon, NRC (w/o Enclosures/Attachments)
J. Sebrosky, NRC (1A, 1E)
J. Peralta, NRC (1A, 1E)
R. Gruel, PNL (1A, 1E)
N. J. Liparulo, Westinghouse (w/o Enclosures/Attachments)

2894A

3 of 10

OITS 1792 DSER 3.9.2.1-4

Re: SSAR Chapter 14, Section 14.2.9.1.7 Question 210.57

3.9.2.1-4 Westinghouse should revise the SSAR as noted in Section 3.9.2.1 of this report. (Incorporate SSAR revision from RAI 210.57, acceptance standard for alternating stress intensity.)

Response:

Subsection 14.2.9.1.7 on expansion, vibration, and dynamic effects testing, has been revised to include reference to SSAR subsection 3.9.2, which delineates the acceptance criteria for alternating stress intensity due to vibration.

Westinghouse Status: Closed

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4857
DCP/NRC0635
Docket No.: STN-52-003

October 23, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: RESPONSES TO NRC MECHANICAL ENGINEERING BRANCH QUESTIONS INCLUDING HIGH-ENERGY LINE BREAK

Dear Mr. Quay:

Attached are responses to several open items discussed in NRC letter dated August 20, 1996. These items were Mechanical Engineering Branch questions in the ECGB scope of view. The synopsis of the NRC position comes from the NRC letter. The questions are identified by the numbers from the letter, DSER open item, and OITS number. The questions addressed in the attachment include questions related to high-energy line break, piping supports, and the initial test program.

This submittal will permit completion of the staff review for items included and preparation of the Final Safety Evaluation Report.

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

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Attachment

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

Enclosed Responses to NRC Request for Additional Information
Letter NSD-NRC-96-4857

From NRC letter dated August 20, 1998

Question 13. - Open Item 3.6.2-1
Question 15. - RAI# 210.40
Question 18. - Open Item 3.6.2.3-5
Question 19. - Open Item 3.9.2.1-1
Question 26. - Open Item 3.9.3.3-1
Question 27. - Open Item 3.9.3.3-2

Confirmatory Items
Item 7. - DSER CN 3.9.2.1-4
Item 8. - DSER CN 3.9.2.3-1

6 of 10

Confirmatory Items

7. DSER CN 3.9.2.1-4 (1792) - Reference SSAR 3.9.2.1.1 in 14.2.7.8.1.78 - RAI # 210.57

The Draft Revision 9 for Chapter 14 replaced Section 14.2.8.1.78 with Section 14.2.9.1.7. However, the response to RAI# 210.57 does not appear to be in this Draft.

Response:

Subsection 14.2.9.1.7, paragraphs b) and c) will be modified as follows:

b) Vibration testing is performed on safety-related and high-energy system piping and components during both cold and hot conditions to demonstrate that steady-state vibrations are within acceptable limits. **See subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to vibration.** This testing includes visual observation and local and remote monitoring in critical steady-state operating modes. Results are acceptable when visual observations show no signs of excessive vibration and when measured vibration amplitudes are within acceptable limits.

c) Testing for significant dynamic events is conducted during hot functional testing and may be performed as part of other specified preoperational tests. This testing is conducted to verify that stress analysis of safety-related and high energy system piping under transient conditions are acceptable. **See subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to vibration.** These tests are performed to verify that the dynamic effects are within expected values during transients such as pump starts and stops, valve stroking, and significant process flow changes.

Deflection measurements during various plant transients are recorded and compared to acceptance limits.

This item is Resolved pending incorporation in a formal SSAR revision.

8. DSER CN 3.9.2.3-1 (1793) - Flow-induced vibration tests for all plants - RAI# 210.58

SSAR Revision 4 revised Section 3.9.2.4 to state that reactor internals of AP600 plants subsequent to the first plant will perform hot functional tests and post test inspection to ensure structural integrity and operability of the internals. This agrees with the response to RAI# 210.58 and is acceptable. However, Section 14.2.8.1.77 was replaced by proposed Section 14.2.9.1.9 in the Draft Revision 9 for Chapter 14, dated July, 1996. This proposed section only addresses the prototype plant (first plant only) tests to comply with that portion of RG 1.20. There should be another section in Chapter 14 to provide the same commitment as that in Section 3.9.2.4. Therefore, this issue remains open.

7 of 10

d) The ability of the emergency air supply to maintain the main control room at the proper positive pressure is demonstrated, verifying proper operation of the main control room pressure relief dampers.

e) The ability of the habitability system to maintain the main control room environment as well as temperatures in the protection and safety monitoring system cabinet and emergency switchgear rooms during a long term loss of the nuclear island nonradioactive ventilation system may be verified with a limited duration test. **This verification is only required for the first plant.**

14.2.9.1.7 Expansion, Vibration and Dynamic Effects Testing

Purpose

The purpose of the expansion, vibration and dynamic effects testing is to verify that the safety-related, high energy piping and components are properly installed and supported such that expected movement due to thermal expansion during normal heatup and cooldown, and as a result of transients; thermal stratification and thermal cycling; as well as vibrations or dynamic effects during steady-state and transients do not result in excessive stress or fatigue to safety-related plant systems and equipment, as described in Section 3.9.

Prerequisites

The construction testing and preoperational testing of the reactor coolant system at cold conditions has been successfully completed. Required portions of the chemical and volume control system, passive core cooling system, normal residual heat removal system, main feedwater system, startup feedwater system, steam generator system, and steam generator blowdown system are operational. Piping and components within the reactor coolant system and steam generator system pressure boundaries and their associated supports and restraints have been inspected and determined to be installed as designed. Permanently installed support devices have been verified to be in their expected cold, static positions and temporary restraining devices such as hanger locking pins have been removed. The instrumentation required for this testing is installed.

General Test Method and Acceptance Criteria

During hot functional testing, verifications that ASME Code Class 1, 2, and 3 high-energy piping system components, piping, support, and restraint deflections are unobstructed and within design basis functional requirements. These tests verify that thermal movements for safety-related piping supports with system operating temperature above 250°F are within design specifications. The high-temperature portions of the following systems are considered for inclusion in this test:

- Reactor coolant system
- Chemical and volume control system
- Passive core cooling system

- Steam generator system (including the safety-related portions of main steam system, main and startup feedwater systems, and steam generator blowdown system)

- Normal residual heat removal system

a) Thermal expansion testing during the preoperational testing phase consists of displacement measurements on the above systems during heatup and cooldown of the reactor coolant system and associated systems (including heatup and cooldown of the passive core cooling system). The testing is performed in accordance ASME OM Standard, Part 7 as discussed subsection 3.9.2.1.2 and consists of a combination of visual inspections and local and remote displacement measurements. This testing includes the inspection and measurement of deflection data associated with support thermal movements to verify support swing clearance at specified heatup and cooldown intervals; that there is no evidence of blocking of the thermal expansion of any piping or components, other than by installed supports, restraints, and hangers; that spring hanger movements remain within the hot and cold setpoints; that moveable supports do not become fully retracted or extended; and that piping and components return to their approximate baseline cold positions.

b) Vibration testing is performed on safety-related and high-energy system piping and components during both cold and hot conditions to demonstrate that steady-state vibrations are within acceptable limits. See Subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to vibration. This testing includes visual observation and local and remote monitoring in critical steady-state operating modes. Results are acceptable when visual observations show no signs of excessive vibration and when measured vibration amplitudes are within acceptable levels.

c) Testing for significant dynamic events is conducted during hot functional testing and may be performed as part of other specified preoperational tests. This testing is conducted to verify that stress analyses of safety-related and high-energy system piping under transient conditions are acceptable. See Subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to vibration. These tests are performed to verify that the dynamic effects are within expected values during transients such as pump starts and stops, valve stroking, and significant process flow changes.

Deflection measurements during various plant transients are recorded and compared to acceptance limits.

d) As described in subsection 3.9.3, temperature sensors are installed on the pressurizer surge line and pressurizer spray line for monitoring thermal stratification and thermal cycling during power operation. Testing is performed to verify proper operation of these sensors. **Note that this verification is required only for the first plant.**

The main control room habitability system is classified as a high energy system based on the pressure criteria not temperature. Tests that measure thermal movements are not required. Vibration testing of the high pressure portion of the main control room habitability system is performed during testing of the air delivery rate provided to the control room. See subsection 14.2.9.1.6 for information on the testing of the main control room habitability system.

14.2.9.1.8 Control Rod Drive System

Purpose

The purpose of the control rod drive system testing is to verify the proper operation of the control rod drive mechanisms, motor-generator sets and system components as described in subsection 3.9.4 and Section 4.6, and in appropriate design specifications.

Prerequisites

The construction tests of the control rod system have been completed. Required interfacing systems, as needed, are completed to the extent sufficient to support the specified testing and the appropriate system configuration. Required electrical power supplies are energized and operational.

For the control rod drive mechanism cooling test, the plant is at or near normal operating temperature and pressure, and hot functional testing is in progress. The integrated head and control rod drive mechanism cooling system are in their normal operational alignment.

For the control rod drive mechanism motor-generator sets tests, a three-phase load bank is available for motor generator set testing under loaded conditions.