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

NSD-NRC-97-4982
DCP/NRC0738
Docket No.: STN-52-003

February 12, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: INFORMAL CORRESPONDENCE

Dear Mr. Quay:

Please find attached correspondence that we are sending you formally. We have previously sent you this correspondence informally over the period January 13, 1997 through January 31, 1997.

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

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

/jml

Attachment

cc: N. J. Liparulo, Westinghouse (w/o Attachment)
T. T. Martin, NRC (w/o Attachment)

E004 1/1

180064

3081a

Attachment 1 to Westinghouse Letter DCP/NRC0738

DATE	ADDRESSEE	DESCRIPTION
1/15/97	Quay	Open item #27 status
1/15/97	Quay	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/15/97	Quay	Open item #123 status
1/13/97	Huffman	Westinghouse understanding of status of CAD comments per discussions with technical staff
1/15/97	Jackson/Kenyon	SSAR markup to resolve open item 5.b of NRC letter of 10/17/96. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/15/97	Jackson/Kenyon	SSAR markup to resolve open item 5.c of NRC letter of 10/17/96. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/15/97	Jackson	Response to open item #3122
1/15/97	Jackson/Kenyon	SSAR markup to close item 1 from 1/7/97 mtg. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/16/97	Quay	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/16/97	Quay	Open item closure status
1/16/97	Jackson/Kenyon	SSAR markup to resolve open item 5.e of NRC letter of 10/17/96. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/16/97	Huffman	SSAR markup to close open item #158. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/17/97	Quay	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/17/97	Quay	Information on open item #135
1/17/97	Jackson/Kenyon	SSAR markup to close parts 1,2 and 4 of open item #306. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/17/97	Jackson/Kenyon	SSAR markup to resolve open item 7.a.(3) of NRC letter of 10/17/96. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/20/97	Jackson	Corrections to Appendix A of PIRT WCAP
1/21/97	Bongarra	Markups of WCAP-14401 to address DSER open item 18.11.3.4-1.

1/21/97	Bongarra	Draft revision 2 to AP3.5 "Design Reviews"
1/30/97	Quay	Information on open item #134.
1/22/97	Huffman	Information for 1/22 phone call
1/23/97	Quay	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/23/97	Quay	Information on open item #137
1/23/97	Quay	Open item closure status
1/23/97	Quay	Request for NRC to acknowledge receipt of information previously sent on open item closure
1/23/97	Quay	Information on open item #140
1/24/97	Quay	Information on open item #139
1/24/97	Quay	Information on open item #141
1/24/97	Quay	Information on open item #138
1/24/97	Quay	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/27/97	Quay	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/27/97	Quay	Information to support 1/27 phone call on open items from sections where FSER input should be complete.
1/27/97	Jackson	SSAR markup to resolve open item 5.f of NRC letter of 10/17/96. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/27/97	Jackson	SSAR markup to resolve open item 244 per 11/5/96 phone call. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/28/97	Jackson	Confirmation of "active" open items as discussed in 1/28/97 phone call.
1/28/97	Quay	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/28/97	Jackson	Information on open item #586
1/28/97	Jackson	First page of DCP/NRC0583 - 11/20/96
1/28/97	Jackson	Response to item 7.i.(1) of NRC letter of 10/17/96.
1/28/97	Sebrosky	PRA page markups
1/28/97	Jackson	Request to change status on open item #338.
1/28/97	Jackson	SSAR markup to resolve open item 243 per 11/5/96 phone call. Will be in Revision 11 to the SSAR unless we hear otherwise.

1/29/97	Scaletti	Request for NRC to acknowledge receipt of information previously sent on open item closure.
1/29/97	Scaletti	Information on open item #586
1/29/97	Scaletti/Huffman	Request for confirmation that there are no Westinghouse actions for the open items for SSAR section 6.1
1/29/97	Scaletti	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/30/97	Scaletti	Reminder list of open items where there is a difference in Westinghouse and NRC status
1/30/97	Scaletti/Jackson	Information on open item #586.
1/30/97	Jackson	Example table for Jeff. Recognize this is beyond BTP requirement.
1/30/97	Quay	Open item closure plot.
1/30/97	Jackson	SSAR markup to resolve open item 243 per 1/30/97 phone call. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/30/97	Kenyon	Information to support phone call on aerosol removal
1/30/97	Huffman	Summary writeups for post-72 hours and spent fuel pool cooling proposed changes to be discussed at February 4, 1997 meeting between Westinghouse and the staff
1/30/97	Huffman	SSAR markup for Chapter 8 in partial satisfaction of open item 4615. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/30/97	Huffman	Information on open item 1078.
1/31/97	Jackson	Markups to WGOTHIC applications report.
1/31/97	Jackson	SSAR markup for to resolve open item #309. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/31/97	Jackson	SSAR markup to resolve open item 5.f of NRC letter of 10/17/96. Change to 12/27/96 faxed material. Will be in Revision 11 to the SSAR unless we hear otherwise.
1/31/97	Huffman	ADS roadmap
1/31/97	Jackson	SSAR markup to resolve open item 333 from 11/5/96 phone call. Will be in Revision 11 to the SSAR unless we hear otherwise.

FAX to TED QUAY

January 15, 1997

CC: Diane Jackson
Bill Huffman
Tom Kenyon
John Butler
Don Lindgren
Gene Piplica
Brian McIntyre

OPEN ITEM #27

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. **On January 10, I** attached are copies of relevant documentation related to Open Item #27. We provided a response on May 13, 1996, and believed it was acceptable. In response to a request related to the IST review, we provided additional clarification on December 17, 1996. ~~Our records show no other outstanding Westinghouse action and we request that NRC provide a definitive action for Westinghouse or provide direction to change this item to "Action N." I am sending this to you in Diane's absence. If I should send these to someone else, please let me know.~~ Thank you.

Upon further investigation I discovered that, during a telecon between Huffman and Piplica and NRC reviewers, this item was discussed. The action remained with Westinghouse and we need to describe how the "B1" test data is related to the overall valve qualification process. This entry has been added to OITS and Westinghouse intends to provide the requested information in the near future. Thanks again.

Jim

Jim Winters
412-374-5290

FAX to TED QUAY

January 15, 1997

CC: Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre

As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
4 (RAI 410.262)	9/10/96	1/9/97
21 (RAI 471.24)	5/20/96	1/9/97
30 (RAI 952.99)	5/13/96 12/17/96 - repeat	1/13/97
37 (RAI 260.74)	4/22/96	1/14/97

Thanks for your help.

Jim

Jim Winters

FAX to TED QUAY

January 15, 1997

CC: Diane Jackson
Tom Kenyon
Don Lindgren
Ed Johnson
Brian McIntyre

OPEN ITEM #123 (M3.6.1-2)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of relevant documentation related to Open Item #123 (M3.6.1-2) We provided an SSAR revision on April 30, 1996, and believed it was acceptable. The information of interest is circled on the attached SSAR pages. Our records show no outstanding Westinghouse action on this item (#123) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/15/97

Selection: [item no] between 123 And 123 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
123	NRR/SPLB	3.6.1	MTG-OI	M3.6.1-2 (PIPING FAILURES OUTSIDE CONTAINMENT) Identify all systems with high-energy lines outside containment. Closed - The SSAR (3.6.1.2.1) notes the area of the aux building where pressurization due to high energy lines must be considered. Also, Table 3.6-1 notes breaks in the turbine building that must be considered for effects on main control room.	Johnson,E.	Closed	Action W		

2 of 5

3.6.1.2 Description

Essential systems are evaluated to demonstrate conformance with the design bases and to determine their susceptibility to the failure effects. Table 3.6-1 identifies systems which contain high and moderate-energy lines. The systems listed include all high- and moderate-energy systems inside containment plus the high- and moderate-energy systems in the auxiliary building near containment penetrations (including access hatches), the main control room, the Class 1E dc and UPS system or the portions of the passive containment cooling system located in the auxiliary building. The table does not list systems that operate at or close to atmospheric pressure including air handling and gravity drains. High energy system piping in the turbine building adjacent to the auxiliary building is evaluated for potential effects on the main control room. These systems are included on Table 3.6-1.

The definition of high and moderate-energy systems is provided in paragraph A of subsection 3.6.1.1.

The postulated break, through-wall crack, and leakage crack locations are determined according to subsections 3.6.2 and 3.6.3.

Equipment is considered to be separated from the dynamic effects of pipe rupture when the equipment is located in a different subcompartment. For the case of pipe whip, equipment may be considered separated for dynamic effects based on the distance from the pipe and the length of pipe that is moving. For the case of jet impingement in a line with saturated or subcooled fluid, equipment more than ten pipe diameters from the break location is considered separated for dynamic effects.

Equipment located in the same subcompartment as a break, through-wall crack, or leakage crack is subject to potential environmental and flooding effects. Equipment may also be subject to environmental and flooding effects of steam and water vented into a subcompartment from an adjoining subcompartment.

3.6.1.2.1 Pressurization Response

Pressurization response analyses are performed for subcompartments containing high-energy piping for which break locations are defined by subsections 3.6.2.1.1.1, 3.6.2.1.1.2, and 3.6.2.1.1.3 or postulated leakage flaws are defined based on subsection 3.6.3.3. Table 3.6-2 identifies those terminal end pipe breaks considered for the evaluation of the effects of pressurization loads on subcompartments. The terminal end pipe breaks inside containment that are postulated in piping that is not evaluated to the leak-before-break requirements of subsection 3.6.3 are summarized in Table 3.6-2. The subcompartments are identified using the room numbers and room names given on Figures 1.2-4 through 1.2-10 as supplemented by Table 3.6-2. The subcompartments inside containment are designed to accommodate the pressurization loads from these breaks. In order to account for high stress break locations and the additional pressure boundary leakages from manways and flanges, pressurization loads on compartments inside containment enclosing high-energy piping are designed as described in subsection 3.8.3.4.

There is no high-energy piping that can pressurize the annulus between the containment vessel and the shield building. Guard pipes are provided for the main steam, feedwater, and steam generator blowdown containment penetrations passing through the annulus as shown on Figure 3.8.2-4. The chemical and volume control system makeup piping is classified as high energy due to its design pressure, but does not cause pressurization because it is at ambient temperature.

The pressurization loads for the in-containment refueling water storage tank are based on the pressure and hydrodynamic loads due to the maximum discharge through the first, second, and third stages of the automatic depressurization system valves.

The pressurization loads for the reactor vessel annulus for the evaluation of asymmetric compartment pressurization are based on a 5-gallon per minute leakage crack in the primary loop piping. The internal reactor pressure vessel asymmetric pressurization loads are based on a break in the largest pipe connected to the reactor coolant system that does not qualify for the application of mechanistic pipe break.

There are limited areas in the auxiliary building where the potential for pressurization loads from high-energy lines are considered. The pressurization loads for the steam tunnels are addressed in the discussion of loads due to a break in the break exclusion zone of the main steam and feedwater lines. The pressurization loads for the Elevation 100′ containment penetration room containing the steam generator blowdown break exclusion zone are based on a circumferential rupture of the 4-inch steam generator blowdown piping. The areas through which the chemical and volume control system make-up line run, including the annulus between the containment and the containment shield building, are not subject to pressurization since the temperature of these lines is less than 212°F.

For a discussion of the criteria and analysis methods for subcompartment pressurization analysis, see subsection 6.2.1.2. The analytical methods for transient mass distribution, used for pressure response analysis, are the same as described in WCAP-8077 (Reference 2).

3.6.1.2.2 Main Control Room Habitability

The high-energy lines in closest proximity to the main control room are the main steam line and main feedwater line. The portions of these lines near the main control room are in the main steam line isolation valve compartment and are part of the break exclusion areas.

The main control room is separated from the isolation valve compartment by two structural walls. The areas between the two walls is used for nonessential office and administrative space associated with the control room. The walls separating the main control room from the main steam isolation valve compartment are thick, reinforced-concrete walls.

Consistent with the criteria for evaluation of leaks in the break exclusion area, the subcompartment, including the walls, is evaluated for the effects of flooding, spray wetting and subcompartment pressurization from a 1-square-foot break from either main steam or feedwater line within the respective break exclusion areas. The wall between the main steam

Table 3.6-1
HIGH-ENERGY AND MODERATE-ENERGY FLUID SYSTEMS CONSIDERED FOR PROTECTION OF ESSENTIAL SYSTEMS[(a)]

System	High-Energy	Moderate-Energy
Reactor coolant (RCS)	•	
Steam generator (SGS)[(b)]	•	
Passive core cooling (PXS)	•	
Passive containment cooling (PCS)[(c)]		•
Main control room habitability (VES)	•	
Chemical and volume control (CVS)	•	
Primary sampling (PSS)	•	
Compressed and instrument air (CAS)		•
Normal residual heat removal (RNS)[(a)]		•
Component cooling water (CCS)		•
Spent fuel pit cooling (SFS)		•
Demineralized water (DWS)		•
Liquid radwaste (WLS)		•
Radioactive drain (WRS)		•
Central chilled water (VWS)[(a)]		•
Fire protection (FPS)		•
Steam generator blowdown (BDS)[(d)]	•	
Main and startup feedwater (FWS)[(d)]	•	
Main steam (MSS)[(d)]	•	
Hot water heating (VYS)	•	

Notes:

a. Systems included on this list are high-energy or moderate-energy fluid systems located in the containment or the auxiliary building. Systems that operate at or close to atmospheric pressure such as ventilation and gravity drains are not included. The normal residual heat removal system lines are classified as moderate-energy based on the 1 percent rule. These lines experience high-energy conditions for less than 1 percent of the plant operating time. The portions of the normal residual heat removal system from the connections to the reactor coolant system and passive core cooling system to the first closed valve in each line are high energy. The spent fuel pit cooling system and central chilled water system inside containment and through the containment penetration to the connection with the hot water heating system are classified as moderate energy based on the 2 percent rule. These systems experience high-energy conditions for less than 2 percent of the system operating time. See subsection 3.6.1.1 Item A and subsection 3.6.1.2 for additional information.

b. Main and startup feedwater, main steam, and steam generator blowdown lines located in the containment and auxiliary building are part of the steam generator system.

c. The essential portion of the system is at atmospheric pressure.

d. The portion of these systems in the turbine building adjacent to the auxiliary building are evaluated for the effect of a circumferential or longitudinal break on the main control room.

BAM / file

We're Making Sure . . .

DATE: JANUARY 13, 1997
SENDER: R. KEMPER
TO: BILL HUFFMAN
bay location: E428
PHONE: (301) 415-1141
CHARGE NUMBER: ARPP-31119
COMPANY: USNRC
Your Extension: X4579
LOCATION: ROCKVILLE, MD
city/state
Division NTD Dept. NSA
return originals: yes ☒ no ☐

COVER + PAGES = 1 + 3 = 4

* REMOVE ALL STAPLES
* PENCIL WILL NOT TRANSMIT - USE BLACK PEN
* PLEASE MAKE COPIES OF TWO-SIDED PAGES

Comments: ATTACHED IS MY UNDERSTANDING OF THE STATUS OF THE CAD COMMENT REPLIES PER OUR DISCUSSIONS OF 1/9 AND 1/10 WITH L. LOIS & C. FINEMAN.

PLEASE REVIEW, AND I WILL CALL THIS AFTERNOON FOR FEEDBACK.

Please provide the following information:

Phone Number of Receiving Equipment: (301) 415-2300

-- WESTINGHOUSE ENERGY CENTER EQUIPMENT NUMBER --

-- Located in the East Tower - Bay 257 --

AREA CODE (412)	Pitney Bowes 9200LP	Dex 740	VERIFICATION
WIN 8 - 284	374-5744	374-5738	374-4930

Question #	Write-up OK	Agreement verbally - (W) provide text	Write-up OK with agreed to expansion	Not Discussed
1.a.		X		
1.b.		X		
1.b.1		X		
1.b.2		X		
1.b.3		X		
1.c		X		
1.d			X	
1.e			X.	
1.f		X		
1.g		X		
1.h		X		
1.i		X		
1.i.1	W NEEDS A FURTHER ARGUMENT RE: DURATION OF 2ϕ FLOW			
1.i.2		X		
1.i.3		X		
2.a		X		
2.b		X		
2.c.	RESPONSE TO BE INCLUDED IN #2.c RESPONSE			
2.d		X		
2.e	IF "7a" ARE DISCUSSED IN 3/4 LP. PIRT, W WILL EXPA			
2.f		X		
2.g		X		
3.		X		
4.	X			
5.	X			
6.	X			
7.a			X	
7.b			X	
7.c	X			
7.d	X			

Question #	Write-up OK	Agreement verbally - (W) provide text	Write-up OK with agreed to expansion	Not Discussed
7.e				X
7.f				X
8.a				X
8.b				X
8.c	X			
8.d			X	
8.e	X			
8.f				X
8.g				
8.g.1	CLIFF FINEMAN TO CLARIFY			
8.g.2	REFERENCE TO "ITEM 6d"			
8.g.3				
8.i		X		
9.a				X
9.b		X		
9.c				X
9.d		X		
10.a				
10.b	W TO PROVIDE THE FINAL 3/4 LOOP PLANT			
10.c	BLOWDOWN COOLING HEAT TRANSFER CFD			
10.d	TO C. FINEMAN FOR STATISTICIAN REVIEW			
10.e		X		
10.f				X
11.a				X
12.a				X
12.b			X	
12.c				X
12.d				X
12.e			X	
12.f	X: C. FINEMAN TO REVIEW CQD AND THEN			

POSSIBLY COMMENT FURTHER

Question #	Write-up OK	Agreement verbally - (W) provide text	Write-up OK with agreed to expansion	Not Discussed
12.g				X
12.h				X
12.i				X
12.j				X
12.k				X
12.l				X
12.m				X
13.		X		
14.a				X
14.b				X
15.		X		
16.				X
17.				X
17.a				X
17.b				X
17.c				X
17.d				X
17.e				X

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 15, 1997	NAME:	JIM WINTERS
TO:	DIANE JACKSON/TOM KENYON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 4

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/Tom

Here is a markup to resolve item 5.b. of your 10/17/96 letter. Note that VAS, VBS, and VFS do NOT include the gravity type backdraft dampers used in the other systems. The addition for VTS (p 9.4-57) also partially resolves item 5.c of the 10/17/96 letter. This will be in SSAR Revision 11 unless we hear from you.

Jim Winters

cc: LINDGREN
MCINTYRE
CUMMINS
RONVITSIK
WINTERS
HUTCHINGS
JEANNE EVANS

and Backdraft

Shutoff, Control, ~~and~~ Balancing, and Backdraft Dampers

Multiblade, two position shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through shut down fans. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

9.4.8.2.3 System Operation

Normal Plant Operation

During normal operation, both supply air handling units and both exhaust fans operate continuously to maintain suitable temperatures in the radwaste building. The radwaste building supply air flow is automatically modulated to maintain a negative pressure in the building. Electric interlocks between the truck access doors and the supply fan flow controller permits the supply air to drop to 6000 cfm below the exhaust flow when any truck bay door is open. This creates a flow into the building through the open door.

Differential pressure drop across the supply units filter banks is monitored, and individual alarms are actuated when any pressure drop rises to a predetermined level indicative of the need for filter replacement. To replace the filters on a supply unit, the affected supply fan and exhaust fan are stopped and isolated from the duct system by means of isolation dampers. During filter replacement, the supply and exhaust systems operate at 50 percent capacity. In this mode of operation, radwaste processing operations are adjusted to obtain acceptable temperature in the radwaste building.

The hot water unit heaters in the mobile systems facility are not normally required to operate to maintain the general building temperature. These heaters operate, in response to local thermostat control, to temper air entering the building when a truck access door is opened.

The hot water unit heater in the electrical/mechanical room operates in response to local thermostat control to maintain the required minimum temperature.

→ INSERT 9.4-57

Unit Heaters

Unit heaters are the down-blow type with propeller type fans directly connected to the fan motor. Each unit heater is equipped with a four-way discharge outlet.

Electric Duct Heaters

Electric duct heaters are open grid type. The duct heaters are UL-listed for zero clearance and meet requirements of NFPA 70 (Reference 28).

Humidifiers

A humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifier is designed and rated in accordance with ARI 620 (Reference 13).

Fire Dampers

Fire dampers are provided at HVAC duct penetrations through fire barriers to maintain fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL-555 (Reference 15) as applicable.

9.4.9.3 System Operation

9.4.9.3.1 General Area Heating and Ventilation

The general area ventilation system is manually controlled. Roof exhaust ventilators are manually started and stopped as required to satisfy space temperature conditions. Wall louvers located at the ground floor and the two intermediate levels of the turbine building are normally open during ventilation operation. The wall louvers located at the operating floor are manually opened to increase ventilation air to the area during outage operations. The operating floor louvers normally remain closed during power operation.

Hot water unit heaters are controlled automatically or manually. In the automatic mode, the heater fan motors are thermostatically controlled by their respective space thermostats. The plant hot water heating system (VYS) supplies hot water to the unit heaters.

9.4.9.3.2 Electrical Equipment and Personnel Work Area HVAC

During normal operation, the two air handling units of the electrical equipment HVAC system operate continuously and the two air handling units of the personnel work area HVAC system operate continuously. The chilled water coils are supplied from the plant central chilled water system (VWS) and the hot water coils are supplied from the plant central hot water heating system.

INSERT 9.4-57

Shutoff, Control, Balancing, and Backdraft Dampers

Multiblade, two position pneumatically operated shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through shut down fans. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Heating Coils

The hot water heating coils are counterflow, finned tubular type. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifier

The humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifier is designed and rated in accordance with ARI 620 (Reference 13).

Shutoff, Control ~~and~~, Balancing, and Backdraft Dampers

Multiblade, two position shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through shut down fans. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

9.4.11.2.3 System Operation

Normal Plant Operation

During normal operation, one supply air handling unit and one exhaust fan operate continuously to maintain suitable temperatures in the health physics and hot machine shop areas of the annex building. The supply air flow is automatically modulated to maintain a

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 15, 1997	NAME:	JIM WINTERS
TO:	DIANE JACKSON / TOM KENYON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 7

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 / TOM

Here is a markup that should resolve item 5.c of your 10/17/96 letter. It will be included in SSAR Revision 11 unless we hear from you.

Jim Winters

cc: LINDGREN
MCINTYRE
RON VIJUK
CUMMINS
WINTERS
HUTCHINGS
JEANNE EVANS

Construction Class A, Leakage Class I bubble tight dampers. These dampers have safety-related operators that fail closed on loss of electrical power.

Tornado Protection Dampers

The tornado protection dampers are split-wing type and designed to close automatically. The tornado protection dampers are designed against the effect of 300 mph wind.

Shutoff and Balancing Dampers electro-hydraulically operated

Multiblade, two-position shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements in accordance with ANSI/AMCA 500 (Reference 14). The supplemental air filtration subsystem dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 or ASME N509 (Reference 2), Section 5.9.

Combination Fire/Smoke Dampers

Combination fire/smoke dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The combination fire/smoke dampers meet the design, leakage testing, and installation requirements of UL-555S (Reference 25).

Ductwork and Accessories

Ductwork, duct supports, and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. Ductwork, supports, and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards – Metal and Flexible (Reference 17). The supplemental air filtration and main control room/technical support center HVAC subsystem's ductwork that maintains integrity of the main control room/technical support center pressure boundary during conditions of abnormal airborne radioactivity are designed in accordance with ASME N509 (Reference 2), Section 5.10 to provide low leakage components necessary to maintain main control room/technical support center habitability.

9.4.1.2.3 System Operation

9.4.1.2.3.1 Main Control Room/Technical Support Center HVAC Subsystem

Normal Plant Operation

During normal plant operation, one of the two 100 percent capacity supply air handling units and return/exhaust air fans operates continuously. Outside makeup air supply to the supply air handling units is provided through an outside air intake duct. The outside airflow rate is automatically controlled to maintain the main control room and technical support center areas

Electric Heating Coils

The electric heating coils are multi-stage fin tubular type. The electric heating coils meet the requirements of UL 1096 (Reference 10).

Electric Unit Heaters

The electric unit heaters are single-stage or two-stage fin tubular type. The electric unit heaters are UL-listed and meet the requirements of UL 1025 (Reference 26) and the National Electric Code NFPA 70 (Reference 28).

Shutoff, Control, Balancing, and Backdraft Dampers

pneumatically or motor operated

Multiblade, two position shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through shut down fans. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

9.4.2.2.3 System Operation

9.4.2.2.3.1 General Area HVAC Subsystem

Normal Plant Operation

During normal plant operation, both supply air handling units and the toilet/shower exhaust fan operate continuously to maintain suitable temperatures in the areas served. The temperature of the air supplied by each handling units is controlled by individual temperature controls with their sensors located in the annex building main entrance. The temperature sensor sends a signal to a temperature controller which modulates the chilled water control valve and the face and bypass dampers across the supply air heating coil to maintain the area

Heating Coils

The heating coils are hot water, finned tubular type. The outside supply air heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output. Coils are performance rated in accordance with ANSI/ARI 410 (Reference 12).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifier

The humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the supply duct system. The humidifier is performance rated in accordance with ARI 620 (Reference 13).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance rating of the barriers. The fire dampers meet the design, testing and installation requirements of UL-555 (Reference 15).

Shutoff and Balancing Dampers

pneumatically operated

Multiblade, two-position shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Isolation Dampers

Isolation dampers are bubble tight, single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power or loss of air pressure. The isolation dampers are constructed, qualified and tested in accordance with ANSI/AMCA 500 (Reference 14).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standard - Metal and Flexible (Reference 17).

Pressure Differential Control Dampers

Pressure differential control dampers utilize opposed-blade type construction and meet the performance requirements of ANSI/AMCA 500 (Reference 14) or ASME N509 (Reference 2), Section 5.9.

Supply and Exhaust Fans

The supply and exhaust air fans are centrifugal type, single width single inlet (SWSI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. Fan performance is rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5) and ANSI/AMCA 300 (Reference 6).

Containment Penetrations

The containment penetrations include containment isolation valves, interconnecting piping, and vent and test connections with manual test valves. The containment isolation components that maintain the integrity of the containment pressure boundary after a LOCA are classified as Safety Class B and seismic Category I. Seismic Category I debris screens are mounted on Safety Class C, seismic Category I pipe to prevent entrainment of debris through the supply and exhaust openings that may prevent tight valve shutoff. The screens are designed to withstand post-LOCA pressures.

The containment isolation valves inside and outside the containment have air operators. The valves are designed to fail closed in the event of loss of electrical power or air pressure. The valves are controlled by the protection and plant safety monitoring system as discussed in subsection 7.1.1. The valves shut tight against the containment pressure following a design basis accident.

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. The system air ductwork inside containment meets seismic Category II criteria so that it will not fall and damage any safety-related equipment following a safe shutdown earthquake. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standard - Metal and Flexible (Reference 17). The exhaust air ductwork and supports meet the design and construction requirements of ASME N509 (Reference 2), Section 5.10.

Shutoff and Balancing Dampers

pneumatically operated

Multiblade, two-position shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow and meet the performance requirements of

Shutoff, Control, and Balancing Dampers

pneumatically operated

Multiblade, two position shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

9.4.8.2.3 System Operation

Normal Plant Operation

During normal operation, both supply air handling units and both exhaust fans operate continuously to maintain suitable temperatures in the radwaste building. The radwaste building supply air flow is automatically modulated to maintain a negative pressure in the building. Electric interlocks between the truck access doors and the supply fan flow controller permits the supply air to drop to 6000 cfm below the exhaust flow when any truck bay door is open. This creates a flow into the building through the open door.

Differential pressure drop across the supply units filter banks is monitored, and individual alarms are actuated when any pressure drop rises to a predetermined level indicative of the need for filter replacement. To replace the filters on a supply unit, the affected supply fan and exhaust fan are stopped and isolated from the duct system by means of isolation dampers. During filter replacement, the supply and exhaust systems operate at 50 percent capacity. In this mode of operation, radwaste processing operations are adjusted to obtain acceptable temperature in the radwaste building.

The hot water unit heaters in the mobile systems facility are not normally required to operate to maintain the general building temperature. These heaters operate, in response to local thermostat control, to temper air entering the building when a truck access door is opened.

The hot water unit heater in the electrical/mechanical room operates in response to local thermostat control to maintain the required minimum temperature.

Low Efficiency Filters and High Efficiency Filters

The low efficiency filters and high efficiency filters have a rated dust spot efficiency based on ASHRAE 52 (Reference 7). Filter minimum average dust spot efficiency is shown in Table 9.4.10-1. The filters meet UL 900 (Reference 8) Class I construction criteria.

Electric Heating Coils

The electric heating coils are multi-stage fin tabular type. The electric heating coils meet the requirements of UL 1096 (Reference 10).

Roof Exhaust Fans

The standby exhaust fans are roof mounted, direct drive upblast ventilators. The fans are equipped with gravity dampers that open when the fan operates and close when the fan is shut down. The diesel oil transfer module enclosure exhaust fans are direct driven centrifugal fan roof ventilators. The ventilators are equipped with gravity dampers that open when the fan operates and close when the fan is shut down.

Electric Unit Heaters

The electric unit heaters are single-stage or two-stage fin tubular type. The electric unit heaters are UL-listed and meet the requirements of UL 1025 (Reference 26) and the National Electric Code (Reference 28).

Shutoff, Control, Balancing, and Backdraft Dampers

Multiblade, two-position motor operated shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Backdraft dampers are provided to prevent backflow through shut down fans and to relieve pressure from the service module and diesel generator building. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Heating Coils

The hot water heating coils are counterflow, finned tubular type. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifier

The humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifier is designed and rated in accordance with ARI 620 (Reference 13).

Shutoff, Control and Balancing Dampers

pneumatically operated

Multiblade, two position shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

9.4.11.2.3 System Operation

Normal Plant Operation

During normal operation, one supply air handling unit and one exhaust fan operate continuously to maintain suitable temperatures in the health physics and hot machine shop areas of the annex building. The supply air flow is automatically modulated to maintain a

FAX to DIANE JACKSON

RESPONSE FOR OPEN ITEM 3122 (10.4.9.1.2)

January 15, 1997

Westinghouse has reconsidered identifying startup feedwater as a DID system in the SSAR. Consistent with the editorial convention for other systems with DID functions throughout the SSAR, Westinghouse will not explicitly identify startup feedwater as a DID system. Note that the DID functions of the startup feedwater system are included in SSAR subsection 10.4.9.1.2. They are also included in the Certified Design Material (ITAAC) subsection 2.4.1.

Jim

Jim Winters

cc: Lindgren
McIntyre
Ron Vijuk
Cummins
McDermott
Nydes
Haag

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	January 15, 1997	NAME:	Jim Winters
TO:	Diane Jackson/Tom Kenyon	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

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

COMMENTS:

Diane/Tom

This markup should resolve item 1) from our 1/7/97 meeting on fire protection. It will be in SSAR Revision 11 unless we hear from you.

Jim Winters

cc: Lindgren
McIntyre
Cummins
Ron Vijuk
Winters
Hutchings
Jeanne Evans

of emergency lighting in locations where these actions are performed and along the access and egress routes thereto.

Emergency Communications

The safe shutdown evaluations consider the need for and availability of emergency communications within the plant following a fire.

9A.2.7.2 Safe Shutdown Methodology

The safe shutdown process, the systems used, and the functional requirements for safe shutdown are described in Section 7.4. As noted above, only safety-related equipment is utilized for safe shutdown. A description of this equipment is provided in the applicable sections.

Table 9A-2 lists the safety-related components used for safe shutdown and their associated electrical divisions. Each fire area is reviewed to identify the potential scope of fire damage and to verify that the capability to achieve and maintain safe shutdown is preserved.

The shutdown process uses controls located in the main control room. In the event of a fire in the main control room, controls located at the remote shutdown workstation are used.

9A.3 Fire Protection Analysis Results

which are shown on the site plot plan, Figure 1.2-2

The fire protection analysis is conducted for the following primary plant structures:

- Nuclear island
- Turbine building
- Annex building
- Radwaste building
- Diesel generator building

Table 9A-3 identifies the type and quantity of combustible materials in each fire area of the primary plant structures and indicates the equivalent fire duration. Fire detection and suppression features are also summarized in Table 9A-3.

Openings through fire barriers for pipe, conduit, and cable trays are sealed or closed to provide a fire resistance rating at least equal to that of the fire barrier itself. Penetration designs conform to the guidelines of BTP CMEB 9.5-1. Fire barrier penetration openings for ventilation are protected by fire dampers having a rating equivalent to that of the fire barrier. For 1-hour rated fire barriers, fire dampers are not required since the duct itself is an adequate barrier. The protection of door openings conforms to the guidelines of BTP CMEB 9.5-1.

Structural steel fireproofing is provided as described in subsection 9.5.1.2.1.1.

FAX to TED QUAY

January 16, 1997

CC: Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre

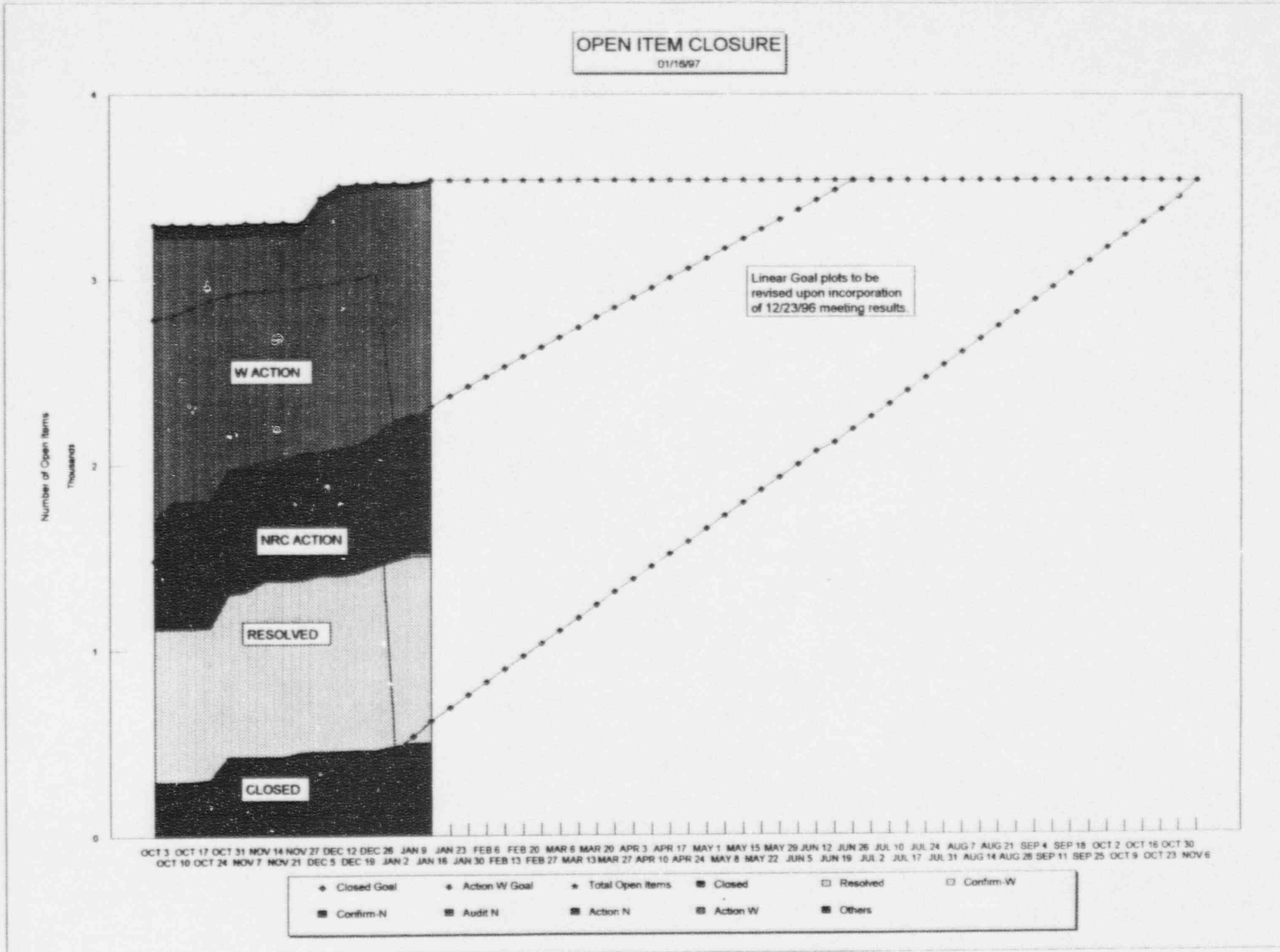
As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
4 (RAI 410.262)	9/10/96	1/9/97
21 (RAI 471.24)	5/20/96	1/9/97
30 (RAI 952.99)	5/13/96 12/17/96 - repeat	1/13/97
37 (RAI 260.74)	4/22/96	1/14/97
123 (M3.6.1-2)	4/30/96	1/15/97

Thanks for your help.

Jim

Jim Winters


OPEN ITEM CLOSURE
01/16/97
Number of Open Items
Thousands
W ACTION
NRC ACTION
RESOLVED
CLOSED
Linear Goal plots to be revised upon incorporation of 12/23/96 meeting results.
Closed Goal
Action W Goal
Total Open Items
Closed
Resolved
Confirm-W
Confirm-N
Audit N
Action N
Action W
Others

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	Diane Jackson / Tom Kenyon	NAME:	Jim Winters
TO:	January 16, 1997	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

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

COMMENTS:

Diane / Tom

This markup should resolve item 5.e from your 10/17/96 letter. It will go into SSAR Revision 11 unless we here from you.

cc: Lindgren
McIntyre
Cummins
Ron Vijuk
Winters
Hutchings
Jeanne Evans

draft dampers are provided at each mobile system connection to prevent blowback through the equipment in the event of exhaust system trip.

9.4.8.2.2 Component Description

The radwaste building HVAC system is comprised of the following major components. These components are located in the non-seismic radwaste building.

Supply Air Handling Units

Each air handling unit consists of a plenum section, a low efficiency filter bank, a high efficiency filter bank, a hot water heating coil, a chilled water cooling coil bank, and a supply fan with automatic inlet vanes.

Supply and Exhaust Air Fans

The supply and exhaust fans are centrifugal type, single width single inlet (SWSI) or double width double inlet (DWDI), with high efficiency wheels and backward inclined blades to produce non-overloading horsepower characteristics. The fans are designed and rated in accordance with ANSI/AMCA 210 (Reference 4), ANSI/AMCA 211 (Reference 5), and ANSI/AMCA 300 (Reference 6).

Low Efficiency Filters and High Efficiency Filters

The low efficiency filters and high efficiency filters have a rated dust spot efficiency based on ASHRAE 52 (Reference 7). The filters meet UL 900 (Reference 8) Class I construction criteria.

Hot Water Unit Heaters

The hot water unit heaters consist of a fan section and hot water heating coil section factory assembled as a complete and integral unit. The unit heaters are either horizontal discharge or vertical downblast type. The coil ratings are in accordance with ANSI/ARI 410 (Reference 12).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Heating Coils

The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating the heat output.

The hot water heating coils are counterflow, finned tubular type. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Heating Coils

The hot water heating coils are counterflow, finned tubular type. The heating coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

The heating coils are provided with integral face and bypass dampers to prevent freeze damage when modulating heat output

Humidifier

The humidifier is a packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifier is designed and rated in accordance with ARI 620 (Reference 13).

Shutoff, Control and Balancing Dampers

Multiblade, two position shutoff dampers are parallel-blade type. Multiblade, control and balancing dampers are opposed-blade type. Air handling unit and fan shutoff dampers are designed for maximum fan static pressure at shutoff flow. Dampers meet the performance requirements of ANSI/AMCA 500 (Reference 14).

Fire Dampers

Fire dampers are provided at duct penetrations through fire barriers to maintain the fire resistance ratings of the barriers. The fire dampers meet the design and installation requirements of UL 555 (Reference 15).

Ductwork and Accessories

Ductwork, duct supports and accessories are constructed of galvanized steel. Ductwork subject to fan shutoff pressure is structurally designed for fan shutoff pressures. Ductwork, supports and accessories meet the design and construction requirements of SMACNA High Pressure Duct Construction Standards (Reference 16) and SMACNA HVAC Duct Construction Standards - Metal and Flexible (Reference 17).

9.4.11.2.3 System Operation

Normal Plant Operation

During normal operation, one supply air handling unit and one exhaust fan operate continuously to maintain suitable temperatures in the health physics and hot machine shop areas of the annex building. The supply air flow is automatically modulated to maintain a

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	January 16, 1996	NAME:	Jim Winters
TO:	Bill Huffman	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

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

COMMENTS:

Bill

Here is a markup to resolve your request on Open Item 158 made during our 12/2 telephone call. I believe this completes our markups for Chapter 5. This will go into SSAR Revision 11 unless we hear from you. I recommend that the NRC Status for item 158 be changed to "Action N."

Jim Winters

cc: Lindgren
McIntyre
Cummins
Ron Vijuk
Winters
Israelson
Jeanne Evans

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/16/97

Selection: [item no] between 158 And 158 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
158	NRR/SPLB	5.2.5	MTG-OI		Hutchings/BPC	Closed	Action W		

M5.2.5-14 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) Discuss how the design satisfies RG 1.45, Positions C.1 through C.9.

Closed - SSAR Revision 3 of 5.2.5 includes commitment for seismic Category I for the sump level monitor and the contaiament atmospheric monitor. Action W - per 12/2 telecon we need to add to SSAR "pointers" showing compliance with RegGuide 1.145.

 Total Records: 1

5.2.4.4 Inspection Intervals

Inspection intervals are established as defined in Subarticles IWA-2400 and IWB-2400 of The ASME Code, Section XI. The interval may be extended by as much as one year so that inspections are concurrent with plant outages. It is intended that in-service examinations be performed during normal plant outages such as refueling shutdowns or maintenance shutdowns occurring during the inspection interval.

5.2.4.5 Examination Categories and Requirements

The examination categories and requirements are established according to Subarticle IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI. The preservice examinations comply with IWB-2200.

5.2.4.6 Evaluation of Examination Results

Examination results are evaluated according to IWA-3000 and IWB-3000, with flaw indications according to IWB-3400 and Table IWB-3410-1. Repair procedures, if required, are according to IWB-4000 of the ASME Code, Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

System pressure tests comply with IWA-5000 and IWB-5000 of the ASME Code, Section XI. These system pressure tests are included in the design transients defined in Subsection 3.9.1. This subsection discusses the transients included in the evaluation of fatigue of Class 1 components due to cyclic loads.

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary (RCPB) leakage detection monitoring provides a means of detecting and to the extent practical, identifying the source and quantifying the reactor coolant leakage. The detection monitors perform the detection and monitoring function in conformance with the requirements of General Design Criteria 2 and 30 and the recommendations of Regulatory Guide 1.45. Leakage detection monitoring is also maintained in support of the use of leak-before-break criteria for high-energy pipe in containment. See subsection 3.6.3 for the application of leak-before-break criteria.

Leakage detection monitoring is accomplished using instrumentation and other components of several systems. Diverse measurement methods including level, flow, and radioactivity measurements are used for leak detection. The equipment classification for each of the systems and components used for leak detection is generally determined by the requirements and functions of the system in which it is located. There is no requirement that leak detection and monitoring components be safety-related. See Figure 5.2-1 for the leak detection approach. The descriptions of the instrumentation and components used for leak detection and monitoring include information on the system.

To satisfy position 1. of Regulatory Guide 1.45

Reactor coolant pressure boundary leakage is classified as either identified or unidentified leakage. Identified leakage includes:

- Leakage from closed systems such as pump gasket or reactor vessel seal leaks that are captured and conducted to a sump or collecting tank
- Leakage into auxiliary systems and secondary systems (intersystem leakage) (This leakage is not considered to be part of the 10 gpm limit identified leakage in the bases of the technical specification 3.4.8. This additional leakage must be considered in the evaluation of the reactor coolant inventory balance.)

Other leakage is unidentified leakage.

5.2.5.1 Collection and Monitoring of Identified Leakage

Identified leakage other than intersystem leakage is collected in the reactor coolant drain tank. The reactor coolant drain tank is a closed tank located in the reactor cavity in the containment. The tank vent is piped to the gaseous radwaste system to prevent release of radioactive gas to the containment atmosphere. The liquid level in the reactor coolant drain tank and total flow pumped out of the reactor coolant drain tank are used to calculate the identified leakage rate. These parameters are available in the main control room. The reactor coolant drain tank, pumps, and sensors are part of the liquid radwaste system. The following sections outline the various sources of identified leakage other than intersystem leakage.

For positions 1 and 7. of Regulatory Guide 1.45

5.2.5.1.1 Valve Stem Leakoff Collection

Valve stem leakoff connections are not provided in the AP600.

5.2.5.1.2 Reactor Head Seal

The reactor vessel flange and head flange are sealed by two concentric seals. Seal leakage is detected by two leak-off connections: one between the inner and outer seal, and one outside the outer seal. These lines are combined in a header before being routed to the reactor coolant drain tank. An isolation valve is installed in the common line. During normal plant operation, the leak-off valves are aligned so that leakage across the inner seal drains to the reactor coolant drain tank.

A surface-mounted resistance temperature detector installed on the bottom of the common reactor vessel seal leak pipe provides an indication and high temperature alarm signal in the main control room indicating the possibility of a reactor pressure vessel head seal leak. The temperature detector and drain line downstream of the isolation valve are part of the liquid radwaste system.

The reactor coolant pump closure flange is sealed with a welded canopy seal and does not require leak-off collection provisions.

Leakage from other flanges is discussed in subsection 5.2.5.3. Collection and Monitoring of Unidentified Leakage.

5.2.5.1.3 Pressurizer Safety Relief Valves

Temperature is sensed downstream of each pressurizer safety relief valve by a resistance temperature detector on the discharge piping upstream of the rupture disc. High temperature indications (alarms in the main control room) identify a reduction of coolant inventory as a result of seat leakage through a pressurizer safety valve. These detectors are part of the reactor coolant system. This leakage is drained to the reactor coolant drain tank during normal plant operation and vented to containment atmosphere during accident conditions that rupture the disc. This identified leakage is measured by the change in level of the reactor coolant drain tank.

5.2.5.1.4 Reactor Coolant Pump Drain

Leakage from the reactor coolant pump drain is directed to the reactor coolant drain tank. This identified leakage is measured by the change in level in the reactor coolant drain tank.

5.2.5.1.5 Other Leakage Sources

In the course of plant operation, various minor leaks of the reactor coolant pressure boundary may be detected by operating personnel. If these leaks can be subsequently observed, quantified, and routed to the containment sump, this leakage will be considered identified leakage.

5.2.5.2 Intersystem Leakage Detection

In accordance with position 4 of Regulatory Guide 1.45,

Substantial intersystem leakage from the reactor coolant pressure boundary to other systems is not expected. However, possible leakage points across passive barriers or valves and their detection methods are considered. Auxiliary systems connected to the reactor coolant pressure boundary incorporate design and administrative provisions that limit leakage. Leakage is detected by increasing auxiliary system level, temperature, flow, or pressure, by lifting the relief valves or increasing the values of monitored radiation in the auxiliary system.

The normal residual heat removal system and the chemical and volume control system, which are connected to the reactor coolant system, have potential for leakage past closed valves. For additional information on the control of reactor coolant leakage into these systems, see subsections 5.4.7 and 9.3.6 and the intersystem LOCA discussion in subsection 1.9.5.1.

5.2.5.2.1 Steam Generator Tubes

An important potential identified leakage path for reactor coolant is through the steam generator tubes into the secondary side of the steam generator. Identified leakage from the steam generator primary side is detected by one, or a combination, of the following:

- High condenser air removal discharge radioactivity, as monitored and alarmed by the turbine island vent discharge radiation monitor
- Steam generator secondary side radioactivity, as monitored and alarmed by the steam generator blowdown radiation monitor
- Secondary side radioactivity, as monitored and alarmed by the main steam line radiation monitors
- Radioactivity, boric acid, or conductivity in condensate as indicated by laboratory analysis

Details on the radiation monitors are provided in Section 11.5, Radiation Monitoring.

5.2.5.2.2 Component Cooling Water System

Leakage from the reactor coolant system to the component cooling water system is detected by the component cooling water system radiation monitor, by increasing surge tank level, by high flow downstream of selected components, or by some combination of the preceding. Refer to Section 11.5, Radiation Monitoring, and subsection 9.2.2, Component Cooling Water System.

5.2.5.3 Collection and Monitoring of Unidentified Leakage

in accordance with position 3 of Regulatory Guide 1.45

To detect unidentified leakage inside containment, the following diverse methods may be utilized to quantify and assist in locating the leakage:

- Containment Sump Level
- Reactor Coolant System Inventory Balance
- Containment Atmosphere Radiation

Other methods that can be employed to supplement the above methods include:

- Containment Atmosphere Pressure, Temperature, and Humidity
- Visual Inspection

The reactor coolant system is an all-welded system, except for the connections on the pressurizer safety valves, reactor vessel head, pressurizer and steam generator manways, and reactor vessel head vent, which are flanged. During normal operation, variations in airborne radioactivity, containment pressure, temperature, or specific humidity above the

normal level signify a possible increase in unidentified leakage rates and alert the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage. The following sections outline the methods used to collect and monitor unidentified leakage.

5.2.5.3.1 Containment Sump Level Monitor

Leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump.

in accordance with position 6 of Regulatory Guide 1.45

A leak in the primary system would result in reactor coolant flowing into the containment sump. Leakage is indicated by an increase in the sump level. The containment sump level is monitored by two seismic Category I level sensors. The level sensors are powered from a safety-related Class 1E electrical source. These sensors remain functional when subjected to a safe shutdown earthquake in conformance with the guidance in Regulatory Guide 1.45. The containment sump level and sump total flow sensors located on the discharge of the sump pump are part of the liquid radwaste system.

Failure of one of the level sensors will still allow the calculation of a 0.5 gpm in-leakage rate within 1 hour. The data display and processing system (DDS) computes the leakage rate and the plant control system (PLS) provides an alarm in the main control room if the average change in leak rate for any given measurement period exceeds 0.5 gpm for unidentified leakage. Unidentified leakage is the total leakage minus the identified leakage. The leakage rate algorithm subtracts the identified leakage directed to the sump.

To satisfy positions 2 and 5 of Regulatory Guide 1.45,

The measurement interval must be long enough to permit the measurement loop to adequately detect the increase in level that would correspond to 0.5 gpm leak rate, and yet short enough to ensure that such a leak rate is detected within an hour. The measurement interval is less than or equal to 1 hour.

When the sump level increases to the high level setpoint, one of the sump pumps automatically starts to pump the accumulated liquid to the waste holdup tanks in the liquid radwaste system. The sump discharge flow is integrated and available for display in the control room, in accordance with position 7 of Regulatory Guide 1.45.

Procedures to identify the leakage source upon a change in the unidentified leakage rate into the sump include the following:

- Check for changes in containment atmosphere radiation monitor indications.
- Check for changes in containment humidity, pressure, and temperature.
- Check makeup rate to the reactor coolant system for abnormal increases.

- Check for changes in water levels and other parameters in systems which could leak water into the containment, and

- Review records for maintenance operations which may have discharged water into the containment.

5.2.5.3.2 Reactor Coolant System Inventory Balance

Reactor coolant system inventory monitoring provides an indication of system leakage. Net level change in the pressurizer is indicative of system leakage. Monitoring net makeup from the chemical and volume control system and net collected leakage provides an important method of obtaining information to establish a water inventory balance. An abnormal increase in makeup water requirements or a significant change in the water inventory balance can indicate increased system leakage.

The reactor coolant system inventory balance is a quantitative inventory or mass balance calculation. This approach allows determination of both the type and magnitude of leakage. Steady-state operation is required to perform a proper inventory balance calculation. Steady-state is defined as stable reactor coolant system pressure, temperature, power level, pressurizer level, and reactor coolant drain tank and in-containment refueling water storage tank levels. The reactor coolant inventory balance is done on a periodic basis and when other indication and detection methods indicate a change in the leak rate.

The mass balance involves isolating the reactor coolant system to the extent possible and observing the change in inventory which occurs over a known time period. This involves isolating the systems connected to the reactor coolant system. System inventory is determined by observing the level in the pressurizer. Compensation is provided for changes in plant conditions which affect water density. The change in the inventory determines the total reactor coolant system leak rate. Identified leakages are monitored (using the reactor coolant drain tank) to calculate a leakage rate and by monitoring the intersystem leakage. The unidentified leakage rate is then calculated by subtracting the identified leakage rate from the total reactor coolant system leakage rate.

Since the pressurizer inventory is controlled during normal plant operation through the level control system, the level in the pressurizer will be reasonably constant even if leakage exists. The mass contained in the pressurizer may fluctuate sufficiently, however, to have a significant effect on the calculated leak rate. The pressurizer mass calculation includes both the steam and water mass contributions.

Changes in the reactor coolant system mass inventory are a result of changes in liquid density. Liquid density is a strong function of temperature and a lesser function of pressure. A range of temperatures exists throughout the reactor coolant system all of which may vary over time. A simplified, but acceptably accurate, model for determining mass changes is to assume all of the reactor coolant system is at $T_{Average}$.

The inventory balance calculation is done by the data display and processing system with additional input from sensors in the protection and safety monitoring system, chemical and volume control system, and liquid radwaste system. The use of components and sensors in systems required for plant operation provides conformance with the regulatory guidance in Regulatory Guide 1.45 that leak detection should be provided following seismic events that do not require plant shutdown.

of position 6

5.2.5.3.3 Containment Atmosphere Radioactivity Monitor

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is continuously monitored for airborne gaseous radioactivity. Air flow through the monitor is provided by the suction created by a vacuum pump. Gaseous and N_{13}/F_{18} concentration monitors indicate radiation concentrations in the containment atmosphere.

The gas channel can respond rapidly to reactor coolant pressure boundary leakage. N_{13} is a neutron activation product which is proportional to power levels. Additionally, N_{13} has a relatively short half life and consequently will reach equilibrium rapidly. An increase in activity inside containment would therefore indicate a leakage from the reactor coolant pressure boundary. Based on the concentration of N_{13}/F_{18} and the power level, reactor coolant pressure boundary leakage can be estimated.

position 6 of Regulatory Guide 1.45

The N_{13}/F_{18} monitoring system has a high sensitivity when the reactor is operating at a power range higher than 20 percent. The N_{13} monitor is seismic Category I. Conformance with the guidance that leak detection should be provided following seismic events that do not require plant shutdown is provided by the seismic Category I classification. Safety-related Class 1E power is not required since loss of power to the radiation monitor is not consistent with continuing operation following an earthquake. Above 20 percent power level, in one hour, a leak less than 0.5 gpm can be detected. Operating experience has indicated the average long-term leakage (from sampling losses, collected leakoffs, and unidentified leakage to the containment) from the reactor coolant system ranges between 0.1 and 0.3 gpm. The N_{13} concentration will increase by at least 25 percent above an existing 0.1 gpm leakage background and almost 10 percent for an existing 0.3 gpm leakage. Both increases are well within the sensitivity of the N_{13}/F_{18} monitor capabilities.

Radioactivity concentration indication and alarms for loss of sample flow, high radiation, and loss of indication are provided. Sample collection connections permit sample collection for laboratory analysis. The radiation monitor can be calibrated during power operation.

5.2.5.3.4 Containment Pressure, Temperature and Humidity Monitors

Reactor coolant pressure boundary leakage increases containment pressure, temperature, and humidity, values available to the operator through the plant control system.

An increase in containment pressure is an indication of increased leakage or a high energy line break. Containment pressure is monitored by redundant Class 1E pressure transmitters. For additional discussion see subsection 6.2.2, Passive Containment Cooling System.

The containment average temperature is monitored using temperature instrumentation at the inlet to the containment fan cooler as an indication of increased leakage or a high energy line break. This instrumentation as well as temperature instruments within specific areas including steam generator areas, pressurizer area, and containment compartments are part of the containment recirculation cooling system.

An increase in the containment average temperature combined with an increase in containment pressure indicate increased leakage or a high energy line break. The individual compartment area temperatures can assist in identifying the location of the leak.

Containment humidity is monitored using temperature-compensated humidity detectors which determine the water-vapor content of the containment atmosphere. An increase in the containment atmosphere humidity indicates release of water vapor within the containment. The containment humidity monitors are part of the containment leak rate test system.

The humidity monitors supplement the containment sump level monitors and are most sensitive under conditions when there is no condensation. A rapid increase of humidity over the ambient value by more than 10 percent is indication of a probable leak.

Containment pressure, temperature and humidity can assist in identifying and locating a leak. They are not relied on to quantify a leak.

5.2.5.4 Safety Evaluation

Leak detection monitoring has no safety-related function. Therefore, the single failure criterion does not apply and there is no requirement for a nuclear safety evaluation. The containment sump level monitors and the containment atmosphere monitor are seismic Category I. The components used to calculate reactor coolant system inventory balance are both safety-related and nonsafety-related components. The containment sump level monitors are powered from the Class 1E dc and UPS system (IDS). Measurement signals are processed by the data display and processing system and the plant control system (PLS).

5.2.5.5 Tests and Inspections

To satisfy position 8 of Regulatory Guide 1.45,

Periodic testing of leakage detection monitors verifies the operability and sensitivity of detector equipment. These tests include installation calibrations and alignments, periodic channel calibrations, functional tests, and channel checks in conformance with regulatory guidance.

satisfy position 7 of Regulatory Guide 1.45 and

5.2.5.6 Instrumentation Applications

The parameters tabulated below are provided in the main control room to allow operating personnel to monitor for indications of reactor coolant pressure boundary leakage. The containment sump level, containment atmosphere radioactivity, reactor coolant system inventory balance, and the flow measurements are provided as gallon per minute leakage equivalent.

Parameter	System(s)	Alarm or Indication
Containment sump level and sump total flow	WLS	Both
Reactor coolant drain tank level and drain tank total flow	WLS	Both
Containment atmosphere radioactivity	PSS	Both
Reactor coolant system inventory balance parameters	PCS, PXS, RCS, VCS, WLS	Both
Containment humidity	VUS	Indication
Containment atmospheric pressure	PCS	Both
Containment atmosphere temperature	VCS	Both
Reactor vessel head seal leak temperature	WLS	Both
Pressurizer safety relief valve leakage temperature	RCS	Both
Reactor coolant pump flange leakoff temperature	RCS	Both
Steam generator blowdown radiation	BDS	Both
Turbine island vent discharge radiation	TDS	Both
Component cooling water radiation	CCS	Both
Main steam line radiation	SGS	Both
Component cooling water surge tank level	CCS	Both

5.2.5.7 Technical Specification

which satisfy position 9 of Regulatory Guide 1.45

Limits for identified and unidentified reactor coolant leakage are identified in the technical specifications, Chapter 16. LCO 3.4.8 addresses leakage limits. LCO 3.4.10 addresses leak detection instrument requirements.

FAX to TED QUAY

January 17, 1997

CC: Sharon, please make copies for: Bill Huffman, Diane Jackson, Tom Kenyon, Joe Sebrosky, John Butler, Cindy Haag, Don Lindgren, Robin Nydes, Brian McIntyre

As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
4 (RAI 410.262)	9/10/96	1/9/97
21 (RAI 471.24)	5/20/96	1/9/97
30 (RAI 952.99)	5/13/96 12/17/96 - repeat	1/13/97
37 (RAI 260.74)	4/22/96	1/14/97
123 (M3.6.1-2)	4/30/96	1/15/97
134 (M3.11-1)	2/29/96 6/19/96 - more	1/16/97

Thanks for your help.

Jim

Jim Winters

FAX to TED QUAY

January 17, 1997

CC: Sharon, please make copies for:

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre
Diane Jackson
Tom Kenyon

OPEN ITEM #135 (M3.11-2)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #135 (M3.11-2). The balance of the relevant information is contained in SSAR Appendix 3D. We provided a revision to Appendix D of the SSAR on February 29, 1996, and believed it was acceptable. We discussed this topic with NRC and made a further revision to SSAR subsection 3.11 on June 19, 1996 (over 6 months ago). Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#135) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/17/97

Selection: [item no] between 135 And 135 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
135	NRR/SPLB	3.11	MTG-OI		Miller	Closed	Action W		

M3.11-2 (EQUIPMENT QUALIFICATION)
The response to Q270.3 proposed a revision to the SSAR to clarify the intent of Section 3.11.2.1. If it is the intent of Westinghouse to comply with the requirements of 10 CFR 50.49 the first sentence of the second paragraph of Subsection 3.11.2.1 should be changed and made to be consistent with the position(s) stated in 10 CFR 50.49 (f).

Resolved - NRC does not accept analysis alone for qualifying equipment under 10 CFR 50.49. Westinghouse will revise to make this clear. Qualification of equipment in mild environment
is normally not part of 3.11 review.
Action W (per NRC) - Revise SSAR.
Closed - SSAR Appendix 3D, Revision 5, addresses the intent of Westinghouse not to qualify by analysis alone. The Revision 8 of SSAR subsection 3.11.2.1 includes appropriate changes.

Nonactive mechanical equipment whose only safety function is structural integrity is designed according to ASME Code guidelines. The accident and post-accident environmental effects are considered in the design of such structural components as pump casings and valve bodies.

The environmental qualification program is restricted to evaluating the design of critical nonmetallic subcomponents of active devices in a harsh environment, where failure results in loss of the active component.

3.11.1.3 Equipment Operability Times

For the AP600 Class 1E electrical and active mechanical equipment, post-accident operability times are shown in Table 3D.4-2 in Appendix 3D.

Specific information for each device qualified as part of the IEEE 323-1974 qualification program is contained in the appropriate equipment qualification data package.

The active mechanical component is qualified for operability as discussed in Section 3.10, using test, analysis, or a combination of tests and analyses. This operability program, combined with the qualification of the electrical appurtenances (for example, valve operators) discussed in the appropriate equipment qualification data packages, demonstrates qualification.

3.11.1.4 Standard Review Plan Evaluation

A discussion of the Standard Review Plan requirements in regard to environmental qualification of mechanical equipment is provided in subsection 1.9.2.

3.11.2 Qualification Tests and Analysis

3.11.2.1 Environmental Qualification of Electrical Equipment

The AP600 approach for environmental qualification of Class 1E equipment is outlined in Appendix 3D. This methodology is developed based on the guidelines provided in IEEE 323-1974 (Reference 1), and 344-1987 (Reference 2).

Qualification for equipment in a harsh environment is based on type testing or testing and analysis. Analysis may be used to determine significant aging mechanisms in mild environment applications. Type testing includes thermal and mechanical aging, radiation, and exposure to extremes of environmental, seismic, and vibration effects. Type testing is done with representative samples of the production line equipment according to the sequence indicated in IEEE 323-1974 to the specified service conditions, including margin. The testing takes into account normal and abnormal plant operation and design basis accident and post-design basis accident operations, as required.

When reliable data and proven analytical methods are available, environmental qualification may be based on analysis supported by partial type test data. This method includes justification of the methods, theories, and assumptions used (that is, mathematical or logical

proof based on actual test data) that the equipment meets or exceeds its specified performance when subjected to normal, abnormal, and design basis accident environmental conditions.

Regulatory guides providing guidance for meeting the requirements of 10CFR50, Appendix A, General Design Criteria 1, 4, 23, and 50; Appendix B, Criterion III to 10CFR50 and 10CFR50.49, include Regulatory Guide 1.89, Regulatory Guide 1.30, Regulatory Guide 1.63, Regulatory Guide 1.73, Regulatory Guide 1.100, and Regulatory Guide 1.131. The maintenance surveillance program follows the guidance of Regulatory Guide 1.33.

Additional information regarding conformance with each of these regulatory guides is given in Section 1.9.

3.11.2.2 Environmental Qualification of Mechanical Equipment

AP600 mechanical components identified in Table 3.11-1 are qualified by design to perform their required functions under the appropriate environmental effects of normal, abnormal, accident, and post-accident conditions as required by General Design Criterion 4 and discussed in Appendix 3D. For mild environments, the area conditions do not change as a result of an accident. There are no degrading environmental effects that lead to common mode failure of equipment in mild environments. Mechanical equipment located in harsh environmental zones is designed to perform under the appropriate environmental conditions.

For mechanical equipment, there are two categories of components:

- Active equipment - equipment that performs a mechanical motion as part of its safety-related function.

The program for environmental qualification of active mechanical components is based on a combination of design, test, and analysis of critical sub-components, which is supported by maintenance and surveillance programs.

- Nonactive equipment - equipment whose only safety-related function is structural integrity. Nonactive components are designed for structural integrity according to ASME Code, Section III, as discussed in Section 3.9.

3.11.3 Loss of Ventilation

The abnormal environmental conditions shown on Tables 3D.5-3 and 3D.5-4 reflect anticipated maximum conditions based on loss of normal ventilation systems.

Normal containment heat removal is provided by the nonsafety-related containment air recirculation cooling system. If this system is out of service for an extended period of time, the passive containment cooling system may be initiated to maintain the temperature and pressure below the limits noted. Environmentally qualified equipment located in containment performs its functions under these conditions until the normal containment cooling system is restored.

January 17, 1997

Diane Jackson / Tom Kenyon

This package provides an SSAR markup to resolve part 1), part 2), part 4) of Open Item 306 on Fire Protection. It will go into SSAR Revision 11 unless we hear from you.

Jim Winters

cc: Lindgren
Winters
Hutchings
Jeanne Evans.

FEDEXED on 1/17/97
JW

Any damage which the fire is capable of causing is assumed to occur immediately. No credit is taken for proper operation of equipment or proper positioning of valves which are not protected from the effects of a postulated fire.

Zone of Influence

A postulated fire does not exceed the boundary of the fire area. For fire areas outside the main control room, remote shutdown workstation, and containment fire areas, all equipment in any one fire area is assumed to be rendered inoperable by the fire and re-entry into the fire area for repairs and operator actions is assumed to be impossible. However, no credit is taken for complete fire damage in cases in which complete damage is beneficial and partial damage is not. Chases for electrical cables, piping or ducts that pass through the fire area but are separated from it by 3-hour fire barriers are outside the zone of influence for that fire area.

Inside the containment fire area, potential fire damage is evaluated by fire zone. All equipment in any one fire zone is assumed to be rendered inoperable by the fire unless the fire protection analysis demonstrates otherwise. Class 1E electrical cables that are located in or pass through the fire zone but are separated from it by a 3-hour fire barrier are outside the zone of influence for that fire zone.

Independence of Affected Fire Areas

Only systems, components, and circuits free of fire damage are credited for achieving safe shutdown for a given fire. Systems, components, and circuits outside the zone of influence are considered free of fire damage if the effects of the fire do not prevent them from performing their required safe shutdown functions.

Event Assumptions

Plant accidents and severe natural phenomena are not assumed to occur concurrently with a postulated fire. Furthermore, a concurrent single active component failure (independent of the fire) is not assumed.

Offsite Power

A loss of offsite power is assumed concurrent with the postulated fire only when the safe shutdown evaluation indicates the fire could initiate the loss of offsite power.

Availability of Nonsafety-Related Systems

Only safety-related components and systems are assumed to be available to perform safe shutdown functions. (This is more stringent than required by BTP CMEB 9.5-1.) Fire protection and smoke control systems are assumed to function as designed to detect and mitigate the effects of the fire.

For each fire area or zone, the safe shutdown evaluation is valid for the worst case fire in the area or zone and initial use of nonsafety-related equipment.

If offsite power is available, nonsafety-related systems are assumed to continue to operate if a more conservative evaluation would result. ~~The~~ Each safe shutdown evaluation ~~also considers~~ is also valid considering the possibility that the operator may initiate safe shutdown using available nonsafety-related systems and that, should the fire later cause those systems to fail, safety-related systems may be automatically or manually actuated to continue the safe shutdown process.

Process Monitoring

Direct process signals are provided to monitor the shutdown process and to assist in determining proper actions for operation of the shutdown methods.

Manual Operation

One of the required manual actions to achieve plant shutdown for a postulated fire event in a fire area is to scram the reactor.

Manual actions by operations personnel include manipulation of equipment located anywhere outside the fire area, if accessibility and staffing levels permit such actions. Entry into the fire area for repairs or operator actions is assumed to be impossible.

Although the typical shutdown sequence does not require manual actions by the operator, fire damage may not be sufficient in many cases to trip the plant. The operator may take appropriate actions to expedite an orderly shutdown. These actions are performed in the main control room. If the fire occurs in the main control room, these actions are performed at the remote shutdown workstation

High-Low Pressure Interfaces

NRC Generic Letter 81-12 (Reference 3) requests the identification and evaluation of the interfaces between the high pressure reactor coolant system and low pressure systems such as the normal residual heat removal system. Typically, these high-low pressure interfaces contain two redundant and independent remotely-operated valves in series. These two valves and their ~~associated~~ control and power cables may be subject to a single fire. This fire may potentially cause the two valves to open, resulting in a fire-initiated loss-of-coolant accident (LOCA) through the high-low pressure system interface. Electrically controlled valves which provide such an interface are identified. These interface valves are considered to be required for safe shutdown.

Spurious Actuation of Equipment

Fire-caused damage is assumed to be capable of resulting in the following types of circuit faults: hot shorts, open circuits, and shorts to ground. Spurious actuation of components caused by these circuit faults are evaluated. Components are assumed to be energized or de-energized by one or more of the above circuit faults. For example, valves are assumed to fail open or closed; pumps are assumed to fail running or not running; electrical distribution breakers could fail open or closed. For three-phase ac circuits, the probability of getting a hot short on all three phases in the proper sequence to cause spurious operation of a motor is

fire protection system in fire areas containing those components. This subject is further discussed in Section 3.4.

Drain systems in the radiological controlled area of the nuclear island Annex Building and Radwaste Building drain to fire zones in the nuclear island where there are no safe shutdown components. Fires in these zones due to potential combustible liquid transport by the drains do not affect safe shutdown.

There is no drain path which could drain combustible liquids to the fire areas in the electrical portion of the nuclear island.

For mechanical equipment fire areas in the nonradioactive auxiliary building, fires caused by potential transport of combustible liquid through the drain system are included in the fire hazards analysis.

9A.3.1.1 Containment/Shield Building

This building comprises one fire area - 1000 AF 01. This fire area includes the areas inside containment as well as the valve room for the passive containment cooling system (PCS), the middle annulus, the upper annulus, and the operating deck staging area outside containment.

The fire protection and the safe shutdown analysis for the containment identifies the location and the separation of the safe shutdown components located inside the containment. The safe shutdown components located inside the containment are primarily ~~associated with~~ the passive core cooling system (PXS), the reactor coolant system (RCS), the steam generator system (SGS), and containment isolation.
components of

For this evaluation, the containment shield building is divided into the following fire zones. These zones are based on the location of the safe shutdown components including termination boxes ~~associated with~~ the containment Class 1E electrical penetrations and the primary cable routing pathway~~s~~ that distribute the Class 1E power and instrumentation and control cabling to the safe shutdown components.
for

Safe Shutdown Evaluation

Table 9A-2 identifies the safe shutdown components located in this fire zone. This compartment is physically separated from other fire zones by structural barriers such that a fire does not propagate to or from this fire zone.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the instrumentation in this zone. Although it is unlikely that all of the components would be damaged, a fire in this fire zone is conservatively assumed to disable all of the above instrumentation. Over-temperature ΔT and over-power ΔT instrumentation located in other fire zones is sufficient to perform the applicable functions to achieve and maintain safe shutdown.

9A.3.1.1.2 Fire Zone 1100 AF 11204

This fire zone is comprised of the following room(s):

Room No.	
11104	Reactor coolant drain tank room
11204	Vertical access area

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components contained in this fire zone. This fire zone is physically separated from other fire zones (except 1100 AF 11300B) by structural barriers or labyrinths such that a fire does not propagate to or from this fire zone.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the instrumentation in this zone. Although it is unlikely that all of the components would be damaged, a fire in this fire zone is conservatively assumed to disable the passive core cooling system containment floodup level and reactor coolant system hot leg instrumentation. The redundant reactor coolant system hot leg instrumentation located in 1100 AF 11206 and passive core cooling system floodup level instrumentation located in 1100 AF 11105 are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

9A.3.1.1.3 Fire Zone 1100 AF 11206

This fire zone is comprised of the following room(s):

Room No.	
11206	Passive core cooling system valve/accumulator room A

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components contained in this fire zone. This compartment is physically separated from other fire zones by structural barriers such that a fire does not propagate to or from this fire zone.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the valves located in this fire zone. A fire in this fire zone is conservatively assumed to disable control of all of the valves and instrumentation in this fire zone. The passive core cooling system safe shutdown components located in fire zones 1100 AF 11207 and 1100 AF 11300B are redundant to those in this fire zone, and are sufficient to perform applicable functions to achieve and maintain safe shutdown. The spent fuel pool cooling system containment isolation valve located outside the containment fire area is redundant to the containment isolation valve inside containment in this fire zone and is sufficient to maintain containment integrity.

Redundant reactor coolant hot leg instruments in fire zone 1100 AF 11204 provide the operator with information required to take corrective action during reduced inventory operation.

9A.3.1.1.4 Fire Zone 1100 AF 11207

This fire zone is comprised of the following room(s):

Room No.	
11207	Passive core cooling system valve/accumulator room B

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire zone. This zone is physically separated from other fire zones by structural barriers such that a fire does not propagate to or from this fire zone. In the case of adjacent fire zone 1100 AF 11208, the accumulator vessel prevents a fire that originates in one zone from propagating to and damaging safe shutdown components located in the other fire zone.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the valves in this fire zone. Although it is unlikely that more than one valve would be damaged, a fire in this fire zone is conservatively assumed to disable control of all of the valves. The passive core cooling system safe shutdown components located in fire zone 1100 AF 11206 and 1100 AF 11300A are redundant to those in this fire zone, and are sufficient to perform applicable functions to achieve and maintain safe shutdown.

9A.3.1.1.5 Fire Zone 1100 AF 11208

This fire zone is comprised of the following room(s):

Room No.	
11208	Normal residual heat removal valve room

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this zone. This fire zone is physically separated from other fire zones by structural barriers such that a fire does not propagate to or from this fire zone. Physical separation from fire zone 1100 AF 11207 is provided by the accumulator vessel as described above.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the valves in this zone. Although it is unlikely that more than one valve would be damaged, a fire in this fire zone is conservatively assumed to disable control of all of the valves. During normal power operation, power to the hot leg suction isolation valves is locked out to protect the high-low pressure interface between the reactor coolant system and the normal residual heat removal such that they will be unaffected by the fire in maintaining the reactor coolant pressure boundary. The normal residual heat removal containment isolation valve, located outside the containment fire area, is redundant to the four containment isolation valves in this zone and is sufficient to maintain containment and reactor coolant pressure boundary integrity.

9A.3.1.1.6 Fire Zone 1100 AF 11209

This fire zone is comprised of the following room(s):

Room No.	
11209	Chemical and volume control system room

Safe Shutdown Evaluation

There are no safe shutdown components in this fire zone. No safe shutdown evaluation is required.

9A.3.1.1.7 Fire Zone 1100 AF 11300A

This fire zone is comprised of the following room(s):

Room No.	
11300	Maintenance floor (southeast quadrant)
11400	Maintenance floor mezzanine

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire zone. The quantity and arrangement of combustible materials in this fire zone are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in adjacent fire zone 1100 AF 11300B.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation related to ~~associated with~~ the above components. Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to disable all of the safe shutdown components in this fire zone.

The redundant passive core cooling system, passive containment cooling system and steam generator system safe shutdown components (listed in Table 9A-2), located in fire zone 1100 AF 11300B, are sufficient to perform applicable functions to achieve and maintain safe shutdown.

The primary sampling system and containment air filtration system containment isolation valves, located outside the containment fire area, are redundant to the containment isolation valves in this fire zone and are sufficient to maintain containment integrity.

The redundant reactor coolant system cold leg flow instrumentation located in fire zones 1100 AF 11300B and 1100 AF 11301 is sufficient to perform applicable functions to achieve and maintain safe shutdown.

9A.3.1.1.8 Fire Zone 1100 AF 11300B

This fire zone is comprised of the following room(s):

Room No.	
11300	Maintenance floor (northern part)
11400	Maintenance floor mezzanine (northern part)

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire zone. This fire zone is physically separated from other fire zones (except 1100 AF 11300A and 1100 AF 11500) by structural barriers or labyrinths such that a fire does not propagate to or from this fire zone. The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation in the termination boxes and cable trays. The quantity and arrangement of combustible materials in this fire zone are such that a fire which damages safe shutdown components in this zone does not propagate to the extent that it damages redundant safe shutdown components in fire zones 1100 AF 11300A and 1100 AF 11500.

The division A and C electrical penetrations listed in Table 9A-2 are conservatively assumed to be disabled as a result of a fire in this fire zone. The B and D electrical penetrations listed

in Table 9A-2, and ~~the associated~~ their cable trays routed from the electrical penetrations to the adjacent fire zone 1100 AF 11500, are protected by a 3-hour fire barrier. These two divisions are sufficient to perform applicable functions to achieve and maintain safe shutdown.

The passive core cooling system passive residual heat removal components and the related reactor coolant system/passive residual heat removal heat exchanger outlet temperature and flow instrumentation (listed in Table 9A-2) are conservatively assumed to be disabled as a result of a fire in this fire zone. The automatic depressurization system, core makeup tank, accumulator, and in-containment refueling water storage tank located outside of this fire zone are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

The passive core cooling system core makeup tank, passive containment cooling system, reactor coolant system pressurizer and steam generator system instrumentation located in this fire zone are conservatively assumed to be disabled as a result of a fire in this fire zone. The redundant passive core cooling system core makeup tank, passive containment cooling system, reactor coolant system pressurizer and steam generator system instrumentation (listed in Table 9A-2) located in fire zone 1100 AF 11300A, 1100 AF 11301 and 1100 AF 11500 are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

The reactor coolant system to chemical and volume control system stop valves located in this fire zone are conservatively assumed to be disabled as a result of a fire in this fire zone. The chemical and volume control system containment isolation valves located outside of this fire zone provide backup isolation capability to maintain the reactor coolant pressure boundary.

The redundant reactor coolant system cold leg flow instrumentation located in fire zones 1100 AF 11300A and 1100 AF 11301 is sufficient to perform applicable functions to achieve and maintain safe shutdown.

The chemical and volume control system and the liquid radwaste system containment isolation valves located outside the containment fire area are redundant to the containment isolation valves inside containment in this fire zone and are sufficient to perform the applicable functions to maintain containment integrity.

The redundant steam line pressure instruments located in area 1201 AF 05 for steam generator 1 and in area 1201 AF 06 for steam generator 2 are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

9A.3.1.1.9 Fire Zone 1100 AF 11300C

This fire zone is comprised of the following room(s):

Room No.	
11300	Maintenance floor (access space between containment shell and west wall of refueling water storage tank)

Safe Shutdown Evaluation

There are no safe shutdown components in this fire zone. No safe shutdown evaluation is required.

9A.3.1.1.10 Fire Zone 1100 AF 11301

This fire zone is comprised of the following room(s):

Room No.	
11201	Steam generator compartment 1
11301	Steam generator 1 lower manway area
11401	Steam generator 1 tubesheet area
11501	Steam generator 1 operating deck
11601	Steam generator 1 feedwater nozzle area
11701	Steam generator 1 upper manway area

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire zone. This fire zone is physically separated from other fire zones (except fire zone 1100 AF 11500) by structural barriers or labyrinths such that a fire does not propagate to or from this fire zone. This fire zone borders fire zone 1100 AF 11500 at the top of the steam generator compartment, which is open to the air space above the operating deck. The quantity and arrangement of combustible materials in this fire zone are such that a fire which damages safe shutdown components in this fire zone does not propagate to the extent that it damages redundant safe shutdown components outside this fire zone.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the components in this fire zone and the reactor coolant pump motors. Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to disable all of the safe shutdown components in this fire zone.

The redundant reactor coolant system fourth stage automatic depressurization system valves and hot leg/cold leg instrumentation located in fire zone 1100 AF 11302, and redundant reactor coolant system pressurizer and steam generator system steam generator level instrumentation located in 1100 AF 11300B are sufficient to perform applicable functions to achieve and maintain safe shutdown.

The four divisions of reactor coolant system/reactor coolant pump bearing water temperature instrumentation are assumed to be disabled and would not be available to detect and provide a trip signal on a loss of component cooling water to the pump. If the fire in this fire zone does not disable the pump, the component cooling water flow to the pump will be unaffected by the fire and will continue to provide cooling water to the pump bearings until the pump is tripped by other means.

The reactor coolant system reactor coolant pump shaft speed instruments are conservatively assumed to be disabled. The redundant reactor coolant system cold leg flow instrumentation located in fire zones 1100 AF 11300A and 1100 AF 11300B is sufficient to perform applicable functions to achieve and maintain safe shutdown.

The four reactor coolant system reactor head vent valves are assumed to be disabled. If power is lost while in the closed position, the head vent valves will maintain reactor coolant pressure boundary integrity. Refer to subsection 9A.3.7.1.1 for a discussion on spurious actuation of reactor coolant system reactor head vent valves.

9A.3.1.1.11 Fire Zone 1100 AF 11302

This fire zone is comprised of the following room(s):

Room No.	
11202	Steam generator compartment 2
11302	Steam generator 2 lower manway area
11402	Steam generator 2 tubesheet area
11502	Steam generator 2 operating deck
11602	Steam generator 2 feedwater nozzle area
11702	Steam generator 2 upper manway area

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire zone. This fire zone is physically separated from other fire zones (except fire zone 1100 AF 11500) by structural barriers or labyrinths such that a fire does not propagate to or from this fire zone. This fire zone borders fire zone 1100 AF 11500 at the top of the steam generator compartment, which is open to the air space above the operating deck. The quantity and arrangement of combustible materials in this fire zone are such that a fire which damages safe shutdown components in this fire zone does not propagate to the extent that it damages redundant safe shutdown components outside this fire zone.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the above components and the reactor coolant pump motors. Although the consequences of a fire are expected to be very limited, a fire in this fire zone is conservatively assumed to disable all of the safe shutdown components in this fire zone.

The redundant reactor coolant system fourth stage automatic depressurization system valves and hot leg/cold leg instrumentation located in fire zone 1100 AF 11301 are sufficient to perform applicable functions to achieve and maintain safe shutdown.

The four divisions of reactor coolant system/reactor coolant pump bearing water temperature instrumentation are assumed to be disabled and would not be available to detect and provide a trip signal on a loss of component cooling water to the pump. If the fire in this fire zone does not disable the pump, the component cooling water flow to the pump will be unaffected

The passive containment cooling system water delivery flow and storage tank level instrumentation are conservatively assumed to be disabled as a result of a fire in this fire zone. The applicable function of verification of passive containment cooling system water delivery can be performed by visual observation via access to the passive containment cooling system air diffuser from the passive containment cooling system valve room.

9A.3.1.1.18 Fire Zone 1270 AF 12701

This fire zone is comprised of the following room(s):

Room No.	
12701	Passive containment cooling system valve room
S06	Stairwell

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire zone. This fire zone is physically separated from other fire zones by structural barriers such that a fire does not propagate to or from this fire zone.

The quantity of combustible materials in this fire zone is very low, consisting primarily of cable insulation ~~associated with~~ related to the valves and instruments in this fire zone. Although it is unlikely that all components would be damaged, a fire in this fire zone is conservatively assumed to disable all of the valves and instruments.

The valves for each passive containment cooling system water delivery path are arranged with a normally open motor-operated valve and normally closed/fail open air-operated valve in series. If the fire causes a loss of power to the valves, the air-operated valves will open and passive containment cooling system flow, which has no adverse impact on achieving and maintaining safe shutdown, will be initiated. Refer to subsection 9A.3.7.1.2 for a discussion of potential spurious actuation of a passive containment cooling system water delivery valve as a result of a fire.

The passive containment cooling system water delivery flow and storage tank level instrumentation are conservatively assumed to be disabled as a result of a fire in this fire zone. The applicable function of verification of passive containment cooling system water delivery can be performed by visual observation via access to the passive containment cooling system air diffuser from the passive containment cooling system valve room or from the upper annulus.

9A.3.1.1.19 Fire Zone 1250 AF 12555

This fire zone is comprised of the following room(s):

Room No.	
12555	Main control room emergency habitability system air storage/operating deck staging area

Safe Shutdown Evaluation

This fire zone is physically separated from other fire zones by structural barriers such that a fire does not propagate to or from this fire zone.

This fire zone contains no components required for safe shutdown after a fire. The pressurized main control room emergency habitability system air storage bottles are not required for safe shutdown after a fire, but are protected from fire-induced overpressure by pressure relief valves.

9A.3.1.2 Auxiliary Building - Nonradiologically Controlled Areas

The safe shutdown systems and components located in the nonradiologically controlled area are ~~associated with~~ portions of the protection and safety monitoring system and the Class 1E dc system, and containment isolation.

The safe shutdown components ~~associated with~~ in the protection and safety monitoring system are the instrumentation and control cabinets located in the nonradiologically controlled area on level 3 (elevation 100'-0"). The safe shutdown components ~~associated with~~ in the Class 1E dc system are the Class 1E batteries on level 1 (elevation 66'-6") and level 2 (elevation 82'-6") and the dc electrical equipment, also on level 2.

The nonradiologically controlled areas of the auxiliary building are designed to provide separation between the mechanical and electrical equipment areas.

The piping compartments in the nonradiologically controlled area are the main steam isolation valve compartments on levels 4 and 5 (elevations 117'-6" and 135'-3", respectively) and the valve/piping penetration compartment on level 3 (elevation 100'-0"). The mechanical equipment rooms in the nonradiologically controlled area are the HVAC compartments on levels 4 and 5.

The nonradiologically controlled areas of the auxiliary building are also designed to provide separation between the Class 1E and the non-Class 1E electrical equipment.

The Class 1E electrical equipment areas have been designed to prevent the migration of smoke, hot gases, and fire suppressant to the extent that they could adversely affect safe shutdown capabilities, including operator actions. These areas are separated from each other and from other plant areas by 3-hour fire barriers. Smoke from a fire in the turbine building

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

Fire dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and combustion products. Smoke and hot gases are subsequently removed from the fire area by reopening the fire dampers after a fire. The nuclear island nonradioactive ventilation system exhausts smoke and hot gases from the battery room to the atmosphere.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by a fire detector which produces an audible alarm locally and both visual and audible alarms in the main control room and the security central alarm station. The fire is extinguished manually using hose streams or portable extinguishers.

The fire resistance of the boundaries of this fire area is greater than the equivalent fire duration, as shown in Table 9A-3. Thus, the fire is contained within the fire area with or without active fire suppression. The battery room is also separated from the other fire zones within this fire area by a 1-hour fire barrier, which limits the spread of fire within the fire area.

The ventilation system does not contribute to the spread of the fire or products of combustion to other fire areas because fire dampers isolate the fire area.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. See Section 3.4 for a discussion of the consequences of a break in a fire protection line in this fire area.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The spare batteries may be connected as a backup power source for any one of the four Class 1E electrical divisions. The terminations of the cables to these divisions from the spare batteries are not normally energized or connected, so a fire in this area has no impact on the unconnected divisions. If the spare batteries are being used as a backup to a Class 1E division, then the consequence of a fire in this area is the same as a fire in the battery room of ~~the associated~~ division to which they are connected.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The electrical equipment in this area is non-Class 1E; however, some division A and C cables are routed through this area. In the event of a fire, the division A and C cabling in this area can be damaged. This damage can result in loss of control of equipment ~~associated with~~ serviced by these cables. Other components ~~associated with the~~ in divisions A and C are not affected.

This postulated fire can disable control of the division A containment isolation valves outside containment. For this event, containment isolation is provided by the redundant containment isolation valves located inside containment outside of this fire area.

Such a fire can also disable control of the division C passive containment cooling system isolation valves. The redundant division B passive containment cooling system isolation valves are not affected. Therefore, the safe shutdown capability of the passive containment cooling system is maintained.

This fire can also disable the division A and C inputs to the reactor trip switchgear. The signals from the remaining two divisions are sufficient to trip the reactor. Furthermore, the reactor can be tripped with the diverse actuation system described in Section 7.7.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.2.7 Mechanical/Piping Areas

9A.3.1.2.7.1 Fire Area 1201 AF 04

This fire area ~~consists of~~ servicing two nuclear island nonradioactive ventilation system equipment rooms ~~associated with~~ divisions B and D. Only division D ~~contains~~ safe shutdown equipment is located within this fire area. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 1241 AF 12405	12405	Lower nuclear island nonradioactive ventilation system divisions B and D equipment room (117'-6")
• 1251 AF 12505	12505	Upper nuclear island nonradioactive ventilation system divisions B and D equipment room (135'-3")

There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

Fire dampers close automatically on high temperature to control the spread of fire and combustion products. Smoke and hot gases are removed from the fire area by reopening the fire dampers after a fire. The radiologically controlled area ventilation system exhausts smoke and hot gases to the atmosphere.

Fire Protection Adequacy Evaluation

A fire in this fire area is detected by fire detectors which produces an audible alarm locally and both visual and audible alarms in the main control room and the security central alarm station. The fire is extinguished manually using hose streams or portable extinguishers.

The fire resistance of the boundaries of this fire area is greater than the equivalent fire duration, as shown in Table 9A-3. Thus, the fire is contained within the fire area with or without active fire suppression.

The ventilation system does not contribute to the spread of the fire or products of combustion to other fire areas because fire dampers isolate the fire area.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system which drains to this fire area are bounded by the consequences of a break in a fire protection line in this fire area. See Section 3.4.

Safe Shutdown Evaluation

Table 9A-2 lists the safe shutdown components located in this fire area. The electrical equipment in this area is non-Class 1E; however, some division A and C cables are routed through this area. In the event of a fire, the division A and C cabling in this area can be damaged. This damage can result in loss of control of equipment ~~associated with~~ served by these cables. Other components ~~associated with the~~ in divisions A and C are not affected.

The spent fuel pool cooling system and normal residual heat removal containment isolation valves are conservatively assumed to be disabled as a result of a fire in this fire area. The redundant spent fuel pool cooling system and normal residual heat removal containment isolation valves located inside containment are outside of this fire area and are sufficient to perform the applicable functions to achieve and maintain safe shutdown.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system is not required because there are no such systems in this fire area. See Section 3.4 for a discussion of the consequences of a break in a fire protection line in this fire area.

Safe Shutdown Evaluation

There are no safe shutdown components in this area, so a fire in this area has no impact on safe shutdown. The electrical equipment in this area is non-Class 1E; however, some division A and C cables are routed through this area. In the event of a fire, the division A and C cabling in this area can be damaged. This damage can result in loss of control of equipment ~~associated with~~ serviced by these cables. Other components ~~associated with the~~ in divisions A and C are not affected. Safe shutdown is possible from equipment in other fire areas.

Neither a fire nor fire suppression activities in this fire area affect the safe shutdown capability of components located in adjacent fire areas.

9A.3.1.3.1.3 Fire Area 1204 AF 01

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 1214 AF 12354	12354	Mid-annulus access room
• 1234 AF 12351	12351	Maintenance floor staging area
• 1234 AF 12352	12352	Personnel hatch
• 1244 AF 12452	12452	Containment air filtration system penetration room
• 1244 AF 12454	12454	Containment air filtration system/spent fuel pool cooling system/primary sampling system penetration room
• 1254 AF 12553	12553	Personnel access area
• 1254 AF 12554	12451	Security room
	12554	Security room
• 1264 AF 12651	12651	Radiologically controlled area ventilation system equipment room

Fire Detection and Suppression Features

- Fire detectors
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

Fire dampers close automatically on high temperature to control the spread of fire and combustion products. If the radiologically controlled area ventilation system is not affected by the fire, smoke and hot gases are removed from the fire area by reopening the fire damper(s)

Fire Protection Adequacy Evaluation

A fire in this fire area is detected through the operation of the dry pipe sprinkler system which produces an audible alarm locally and both visual and audible alarms in the main control room and the security central alarm station. The fire is extinguished by the automatic dry pipe sprinkler system. Water from the sprinklers rapidly fills and cools the small diked area under the tank. If necessary, the fire can also be extinguished manually.

The equivalent fire duration for this fire area exceeds the fire resistance of the fire area boundaries, as shown in Table 9A-3. However, the 3-hour fire resistance of the fire area boundaries provides sufficient time in which to extinguish the fire.

The ventilation system does not contribute to the spread of the fire or products of combustion to other fire areas because fire dampers isolate the area.

9A.3.7 Special Topics

9A.3.7.1 Evaluation of Spurious Actuation

The potential for spurious actuation of equipment as a result of fire damage to electrical circuits is considered for each fire area containing safety-related equipment. As discussed in subsection 9A.2.7.1, one spurious actuation or signal is postulated at a time (except for high-low pressure interfaces). Principal spurious actuation are discussed below. In no case does the spurious actuation of equipment prevent safe shutdown.

9A.3.7.1.1 High-Low Pressure Interfaces

NRC Generic Letter 81-12 requests the identification and evaluation of high-low pressure interfaces between the reactor coolant system and interfacing systems such as the normal residual heat removal system. Per the Generic Letter, these interfaces typically contain two redundant and independent motor-operated valves in series. On a typical pressurized water reactor plant, these two valves and their ~~associated~~ cables may be subject to a single fire. Potential high-low pressure system interfaces of particular interest are discussed below.

control and power

Reactor Coolant System Valve Actuation

NRC Generic Letter 81-12 specifically addresses the reactor coolant/residual heat removal system interface on pressurized water reactors. For AP600, the reactor coolant system to normal residual heat removal system interface is similar to the typical pressurized water reactor configuration. However, the normal residual heat removal system is not a safety-related system and is not required for safe shutdown. To preclude the spurious opening of the interface valves as a result of a fire, the power to the valves is locked out during power operations. Thus, spurious actuation of the reactor coolant system to normal residual heat removal system interface valves does not occur and the safe shutdown capability is not affected.

Passive Core Cooling System Passive Residual Heat Removal Heat Exchanger Inlet Valve Actuation

One normally open valve is provided to isolate the inlet line to the passive residual heat removal heat exchanger. Spurious closure of this valve is assumed to occur where a fire affects ~~the associated~~ its electrical circuity. Such a fire can occur in fire areas or fire zones through which the applicable electrical cables are routed. Spurious closure of this valve disables the passive residual heat removal heat exchanger. Safe shutdown proceeds using the automatic depressurization system as described in subsection 7.4.1.

Passive Containment Cooling System Valve Actuation

Two valves in series isolate each of the two discharge flow paths from the passive containment cooling system storage tank. For purposes of system reliability, one valve in each flow path is normally open and the other is normally closed. Electrical division assignments are shown in Table 9A-2.

Spurious actuation of one of these valves is assumed to occur where a fire affects ~~the associated~~ its electrical circuity. Such a fire can occur in an electrical equipment fire area, in the passive containment cooling system valve room, or in fire areas or fire zones through which the applicable electrical cables are routed.

Spurious actuation of one of these valves causes a passive containment cooling system flow path to be disabled or inadvertently opened, depending on which valve is affected. If a normally closed valve spuriously opens, passive containment cooling system water delivery from that flow path will be initiated which does not adversely affect the capability to achieve and maintain safe shutdown. If one of the normally open valves were spuriously closed to prevent passive containment cooling system water delivery through that flow path when called upon during the safe shutdown process, the redundant passive containment cooling system water delivery flow path would be sufficient to achieve and maintain safe shutdown.

Containment Isolation Valve Actuation

Spurious actuation of a containment isolation valve is assumed to occur where a fire affects ~~the associated~~ its electrical circuitry. Each containment penetration has redundant means of containment isolation.

Reactor Trip Switchgear

The reactor trip switchgear receives signals from each of the four Class 1E electrical divisions. The signals are de-energized to trip. Also, two out of four signals are required to trip. There are two redundant sets of trip switchgear in separate fire areas. There is no single spurious signal which could prevent the reactor from being tripped.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	January 17, 1997	NAME:	Jim Winters
TO:	Diane Jackson/Tom Kenyon	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

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COMMENTS:

Diane/Tom

This markup should resolve Item 7.a.(3) (OITS # 262) of your 10/17/96 letter

It will be in SSAR Revision 11 unless we hear from you.

Please inform us of its current "NRC Status". We recommend "Action N."

cc: Lindgren
McIntyre
Cummins
Ravindra
Hutchins
Jeanne Evans
Winters

Jim Winters

HEPA Filters

HEPA filters are constructed, qualified, and tested in accordance with UL-586 (Reference 9) and ASME N509 (Reference 2), Section 5.1. Each HEPA filter cell is individually shop tested to verify an efficiency of at least 99.97 percent using a monodisperse 0.3-μm aerosol.

Charcoal Adsorbers

Charcoal adsorbers and adsorbent media are constructed, qualified and tested in accordance with ASME N509 (Reference 2), Section 5.2. Each charcoal adsorber is a single assembly with welded construction and 4-inch deep Type III rechargeable adsorber cell, conforming with IE Bulletin 80-03 (Reference 27).

Electric Heating Coils

The electric heating coils are multi-stage fin tubular type. The electric heating coils meet the requirements of UL-1096 (Reference 10). The coils for the supplemental air filtration subsystem are constructed, qualified, and tested in accordance with ASME N509 (Reference 2), Section 5.5.

Electric Unit Heaters

The electric unit heaters are single-stage or two-stage fin tubular type. The electric unit heaters are UL-listed and meet the requirements of UL-1025 (Reference 26) and the National Electrical Code NFPA 70 (Reference 28).

Cooling Coils

The chilled water cooling coils are counterflow, finned tubular type. The cooling coils are designed and rated in accordance with ASHRAE 33 (Reference 11) and ANSI/ARI 410 (Reference 12).

Humidifiers

The humidifiers are packaged electric steam generator type which converts water to steam and distributes it through the air handling system. The humidifiers are designed and rated in accordance with ARI 620 (Reference 13).

Isolation Dampers

Nonsafety-related isolation dampers are bubble tight, single- or parallel-blade type. The isolation dampers have spring return actuators which fail closed on loss of electrical power. The isolation dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500 (Reference 14) or ASME N509 (Reference 2), Section 5.9.

The main control room envelope isolation dampers are seismically analyzed ANSI B31.1 butterfly valves that meet the performance and design requirements of ASME N509 for

13. "Self-Contained Humidifiers," ARI 620-80.

14. "Testing Methods for Louvers, Dampers, and Shutters," ANSI/AMCA 500-83.

15. "Fire Dampers," UL-555, 1990.

16. "High-Pressure Duct Construction Standards," SMACNA, 1975.

17. "HVAC Duct Construction Standards - Metal and Flexible," SMACNA, 1985.

18. "HVAC Duct Leakage Test Manual," SMACNA, 1985.

19. "HVAC Systems - Testing, Adjusting, and Balancing," SMACNA, 1983.

20. Code of Federal Regulations, Title 10, Part 50, Appendix I.

21. Code of Federal Regulations, Title 10, Part 20.

22. "Heat-Stress Management Program for Nuclear Power Plants," EPRI NP-4453 by Westinghouse Electric Corporation, dated February 1986.

23. Branch Technical Position CSB 6-4 to "Containment Isolation System," Standard Review Plan 6.2.4 of NUREG-0800 Rev. 2, July 1981.

24. "Military Specification Filter, Particulate, High-Efficiency, Fire Resistant," MIL-F-51068D.

25. "Leakage Rated Dampers for Use in Smoke Control System," UL-555S, 1993.

26. "Electric Air Heaters," UL-1025, 1991.

27. "Installation of Air Conditioning and Ventilation Systems," NFPA 90A, 1993.

28. "National Electrical Code," NFPA 70, 1990.

29. "Loss of Charcoal from Absorber Cells" IE Bulletin 80-03, 1980

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FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	1/20/97	NAME:	John C. Butler
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COMPANY:	NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

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COMMENTS:

Diane,

Following are corrections to Appendix A references of PIRT WCAP. These corrections were discussed in a phone call on 17th (See Bill H.). I am preparing a letter transmitting these & of course report will be changed with these corrections. JCB

16. WCAP-13566, *AP600 1/8th Large-Scale Passive Containment Cooling System Heat Transfer Test Baseline Data Report*, October 1992.

17. NTD-NRC-95-4489 (WCAP-14382), *WGOTHIC Code Description and Validation*, June 20, 1995.

18. NTD-NRC-95-4561, Scaling Role in AP600 PCS DBA Analysis, September 19, 1995.

**19. NSD-NRC-96-4790, *Scaling Analysis for AP600 Containment Pressure During Design Basis Accidents*, August 8, 1996.

**20. WCAP-14783, *Scaling of the PCS DBA*, (to be issued).

*21. SSAR 6.2.1.1.3, NTD-NRC-95-4504, "Containment Structure Design Evaluation, Proposed Draft/Markups of SSAR, Sections 6.2 and 6.4", July 10, 1995.

*22. NTD-NRC-94-4100, *Enclosure 1, Radiation Heat Transfer Through Fog in the PCCS Air Gap*, April 18, 1994.

23. WCAP-12265, *Tests of Heat Transfer and Water Film Evaporation on a Heated Plate Simulating Cooling of the AP600 Reactor Containment*, Rev. 1, April 30, 1992.

24. WCAP-14048, *Passive Containment Cooling System Bench Scale Wind Tunnel Test*, April 29, 1994.

25. WCAP-13294, *Phase 1 Wind Tunnel Testing for the Westinghouse AP600 Reactor*, April 30, 1992.

26. WCAP-13323, *Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor*, October 2, 1992.

27. WCAP-14068, *Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor*, June 6, 1994.

28. WCAP-14091, *Phase IVB Wind Tunnel Testing for the Westinghouse AP600 Reactor*, July 19, 1994.

29. WCAP-13307, *Condensation in the Presence of a Noncondensible Gas - Experimental Investigation*, April 30, 1992.

**WCAP-14783 will supercede Reference 14

30. I. K. Huhtiniemi, *Condensation in the Presence of Noncondensible Gas: The Effect of Surface Orientation*, Preliminary Thesis (1990), August 16, 1993.

31. A. P. Pernsteiner, *Condensation in the Presence of Noncondensible Gas: Effect of Helium Concentration*, 1993, University of Wisconsin Thesis, November 12, 1993.

32. WCAP-13353, *Passive Containment Cooling System Water Distribution Phase I Test Data Report*, Rev. 0, April 30, 1992.

33. WCAP-13296, *PCS Water Distribution Test Phase II Test Date Report*, April 30, 1992.

34. WCAP-13960, *PCS Water Distribution Phase 3 Test Data Report*, Rev. 0, February 2, 1994.

35. WCAP-12667, *Tests of Heat Transfer and Water Film Evaporation from a Simulated Containment to Demonstrate the AP600 Passive Containment Cooling System*, Rev. 1, April 30, 1992.

36. WCAP-13566, *AP600 1/8th Large Scale Passive Containment Cooling System Heat Transfer Test Baseline Data Report*, Rev. 0, January 1, 1993.

*37. PCS-T2R-050, *Large-Scale Test Data Evaluation*, May 1995.

38. R. Siegel and R.H. Norris, "Tests of Free Convection in a Partially Enclosed Space Between Two Heated Vertical Plates," *Transactions of the ASME*, April 1957, pp. 663-673.

*39. NTD-NRC-94-4100, "Enclosure 2, Liquid Film Model Validation," April 18, 1994.

*40. NTD-NRC-95-4397, "Supporting Information for the Use of Forced Convection in the AP600 PCS Annulus," February 16, 1995.

41. NTD-NRC-94-4166, "AP600 Containment Plume Investigation," June 10, 1994.

42. NTD-NRC-96-4467 (PCS-T2C-059), "Analysis of PCS Wind Tunnel Testing for PCS Heat Removal, June 2, 1995.

*One or more sections of report will be revised as a result of outstanding NRC open items.

43. NTD-NRC-94-4174, "AP600 PCS Design Basis Analysis (DBA) and Margin Assessment," June 30, 1994.

APPENDIX A
SUMMARY TABLE SHOWING
SUMMARY OF SOURCES SUPPORTING PIRT RANKING

APPENDIX A - SUMMARY OF BASES FOR PIRT CLOSURE

Component or Volume		Phenomena/Parameter	Scaling Analyses (Ref. ~~28~~ 20)	Testing Results	Sensitivity Studies Reference No.	First Principles Calc	Engineering Judgment	Test Analysis Report Reference No.	Phenomena Evaluation Report Reference No
					Ranking Basis for Phenomena				
Inside Containment:									
1) Break Source	A	Mass and Energy	yes	yes	5 ~~X~~ Section 10			17 ~~18~~	5 ~~X~~ Section 9
	B	Direction/Elevation		yes	5 ~~X~~ Section 9			17, ~~18~~	5 ~~X~~ Section 9
	C	Momentum		yes				17, ~~18~~, ~~24~~ Section 3.9 9	5 ~~X~~ Section 9
	D	Density							
	E	Droplet/liquid flashing	yes				yes		
2) Containment Volume	A	Mixing/Stratification		yes	17 65 ~~18,~~ ~~X~~ Section 9	yes		17 ~~18~~	5 ~~X~~ Section 9
	B	Intercompartment Flow		yes	~~18,~~ ~~X~~ Section 9 5			17 ~~18~~	5 ~~X~~ Section 9
	C	Gas Compliance	yes						22 ~~20~~
	D	Fog	yes			yes			
	E	Hydrogen Release		yes		yes	yes		
3) Containment Solid Heat Sinks (Steel & Concrete)	A	Liquid Film Energy Transport	yes	yes			yes	29 ~~8~~	39 ~~24~~
	B	Vertical Film Conduction	yes	yes		yes	yes		5 ~~X~~ Section 9
	C	Horizontal Film Conduction	yes			yes	yes		
	D	Internal Heat Sink Conduction	yes		5 ~~X~~ Section 5				
	E	Heat Capacity			5 ~~X~~ Section 5		yes	9 ~~24~~	
	F	Condensation	yes	yes	5 ~~X~~ Section 10			9 ~~24~~	
	G	Convection from cnmt volume	yes						
	H	Radiation from cnmt volume	yes				yes		
4) Initial Conditions	A	Initial Temperature			5 ~~X~~ Section 5		yes		
	B	Initial Humidity			5 ~~X~~ Section 5		yes		
	C	Initial Pressure			5 ~~X~~ Section 5		yes		
5) Break Pool	A	Mixing/Stratification					yes		
	B	Condensation/Evaporation	yes				yes		
	C	Convection from cnmt volume	yes				yes		
	D	Radiation from cnmt volume	yes				yes		
	E	Pool Conduction	yes				yes		
	F	Flooding					yes		

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APPENDIX A - SUMMARY OF BASES FOR PIRT CLOSURE

Component or Volume		Phenomena/Parameter	Scaling Analyses (Ref. ~~28~~ 20)	Testing Results	Sensitivity Studies Reference No	First Principles Calc	Engineering Judgment	Test Analysis Report Reference No	Phenomena Evaluation Report Reference No
			Ranking Basis for Phenomena						
6) IRWST	A	Mixing/Stratification (gas & water)	yes				yes		
	B	Condensation	yes				yes		
	C	Convection	yes				yes		
	D	Radiation	yes				yes		
	E	Conduction in liquid	yes				yes		
	F	Liquid Level	yes				yes		
Containment Shell:									
7) Steel Shell	A	Convection from cnmt volume	yes	yes				9 ~~24~~	
	B	Radiation from cnmt volume	yes					9 ~~24~~	
	C	Condensation	yes						39 ~~24~~
	D	Inside Film Conduction	yes						
	E	Inside Film Energy Transport	yes						
	F	Conduction through shell	yes	yes	5 ~~Section 10~~				
	G	Heat Capacity	yes	yes	5 ~~Section 10~~			9 ~~24~~	40 ~~23~~
	H	Convection to riser annulus	yes	yes					22 ~~28~~
	I	Radiation to baffle	yes	yes		yes			
	J	Radiation to chimney	yes	yes		yes	yes		22
	K	Radiation to fog/air	yes	yes		yes	yes	9 ~~24~~	39 ~~24~~
	L	Outside Film Conduction	yes	yes				17 ~~18~~	
	M	Outside Film Energy Transport	yes	yes	43 ~~28~~			9 ~~24~~	40 ~~23~~
	N	Evaporation to riser annulus	yes	yes	5 ~~Section 10~~				
8) PCS Cooling Water	A	PCCWST Water Flowrate	yes	yes				17 ~~18~~	5 ~~Section 7~~
	B	PCCWST Water Temperature	yes	yes	43 ~~28~~			17 ~~19~~	5 ~~Section 7~~
	C	Film stability and coverage		yes	5 ~~Section 7~~	yes	yes	5 ~~Section 7~~	~~Section 7, 24,~~ 39 45
	D	Film Stripping		yes		yes	yes		
	E	Film Drag				yes			
Outside Containment:									
9) Riser Annulus & Chimney Volume	A	PCS Natural Circulation	yes	yes	5 Section 5 ~~yes~~	yes	yes	17, 9, 11 ~~18,23,~~	40 ~~23~~
	B	Vapor Acceleration				yes	yes		
	C	Fog				yes	yes		
	D	Flow Stability in Chimney	yes						

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APPENDIX A - SUMMARY OF BASES FOR PIRT CLOSURE

			Ranking Basis for Phenomena						
Component or Volume		Phenomena/Parameter	Scaling Analyses (Ref. ~~26~~ 20)	Testing Results	Sensitivity Studies Reference No.	First Principles Calc	Engineering Judgment	Test Analysis Report Reference No	Phenomena Evaluation Report Reference No.
10) Baffle	A	Convection to riser annulus	yes	yes			yes	17 ~~18~~	
	B	Convection to downcomer	yes			yes	yes		
	C	Radiation to shield building	yes			yes	yes		
	D	Conduction through baffle	yes			yes	yes		
	E	Condensation	yes	yes			yes	17 ~~19~~	
	F	Heat Capacity	yes				yes		
	G	Leaks		yes		yes	yes		
11) Baffle Supports	A	Convection to riser air				yes	yes		
	B	Radiation from shell				yes	yes		
	C	Conduction from shell to baffle				yes	yes		
	D	Heat Capacity				yes	yes		
12) Chimney Structure	A	Conduction into chimney	yes			yes	yes		
	B	Convection from chimney air	yes			yes	yes		
	C	Heat capacity of structure				yes	yes		
	D	Condensation on chimney	yes			yes	yes		
13) Downcomer Annulus	A	PCS Natural Circulation	yes					9, 11	
	B	Air Flow Stability	yes	yes		yes			
14) Shield Building	A	Convection to downcomer	yes			yes	yes		
	B	Conduction through shield bldg	yes			yes	yes		
	C	Convection to environment				yes	yes		
	D	Radiation to environment				yes	yes		
15) External Atmosphere	A	Temperature		yes	5 Section 5			14 ~~24~~	41 5
	B	Humidity		yes	5 Section 5			14 ~~24~~	22 7 Section 6
	C	Recirculation		yes				26 Section 6	26 7 Section 6
	D	Pressure Fluctuations		yes				42 5	42

TS.XLS

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RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	January 21, 1997	NAME:	Steve Kerch
TO:	Jim Bongarra / John O'Hara	LOCATION:	Monroeville, PA.
PHONE:	301-415-1046 / 516-344-3638	PHONE:	412-374-5184
COMPANY:	NRC / BNL		
LOCATION:		FAX:	(412) 374-5099

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Comments: Jim and John,

Based on the discussions of this morning's conference call, the attached markups of WCAP-14401 are provided. These markups address DSER open item 18.11.3.4-1, subcriterion 2 (Reference: NRC letter of Jan. 10, 1997, "Comments on the AP600 Human Factors Verification and Validation Plan").

Steve Kerch

Phone Number of Receiving Equipment: 301-415-2222 Jim Bongarra

516-344-4900 John O'Hara

4.0 INTEGRATED SYSTEM VALIDATION

An implementation plan will be developed specifying a methodology for integrated system validation. The objective of integrated system validation is to ensure that the functions and tasks allocated to the plant personnel can be accomplished with the M-MIS design implementation. Explicitly included in the integrated system validation is validation of the AP600 EOPs.

4.1 Methodology

The integrated system validation implementation plan will include a methodology section that addresses:

- Objectives
- Personnel performance issues
- Test methodology and procedures
- Test participants
- Test conditions (including plant conditions, operating sequences, accident scenarios)
- M-MIS description
- Performance measures
- Data analysis
- Acceptance criteria
- Process by which results will be used to determine whether changes to the M-MIS are required, and the process by which change requirements are tracked and verified

4.2 Tools Used for Evaluating Dynamic Task Performance

Integrated system validation will be performed using an AP600-specific, near full-scope, high-fidelity, ~~simulator of the AP600 control room that is similar to a~~ training simulator that satisfies the general requirements of sections 3 and 4 of ANSI/ANS - 3.5 - 1993. The near full-scope, high-fidelity simulator of the AP600 control room will display high physical fidelity (the testbed will physically resemble the actual hardware to be implemented in the AP600 control room), as well as high-fidelity with respect to information content (containing AP600-specific displays and controls), and underlying process dynamics (it shall be driven by an AP600-specific plant simulation). *Near* is used to indicate that features of the simulation ~~are~~ not relevant to the test being made may not be full-fidelity.

Operator actions at non-control room facilities, such as remote shutdown panels, and the TSC, may be evaluated using static mock-ups, or prototypes.

7.0 REFERENCES

ANSI HFS-100-1988, *American Standard for Human Factors Engineering of Visual Display Terminal Workstations*. American National Standards Institute, Santa Monica, California, 1988.

CEI/IEC 964 *Design for Control Rooms of Nuclear Power Plants*. International Electrotechnical Commission, Geneva, Switzerland, 1989.

DOD-HDBK-761A *Human Engineering Guidelines for Management Information Systems*. US Department of Defense, Office of Management and Budget, Washington, D.C., 1990.

IEEE Std. 845-1988 *IEEE Guide to Evaluation of Man-Machine Performance in Nuclear Power Generating Station Control Rooms and Other Peripheries*. Institute of Electrical and Electronics Engineers, 1988.

OCS-T5-001 Roth, E. & Mumaw, R. J. *Man-in-the-Loop Test Plan Description*, Rev. B. March, 1994.

NUREG-0899 *Guidelines for the Preparation of Emergency Operating Procedures*. US Nuclear Regulatory Commission, Washington, D. C., August 1982.

NUREG-1358 *Lessons Learned from the Special Inspection Program for Emergency*. US Nuclear Regulatory Commission, Washington, D. C., April, 1989.

NUREG-0711 *Human Factors Engineering Program Review Model*. US Nuclear Regulatory Commission, Washington, D.C., July, 1994.

NUREG-0700 *Human-System Interface Design Review Guideline*, Rev. 1, Draft Report. US Nuclear Regulatory Commission, Washington, D.C., February, 1995.

NUREG/CR-5908 *Advanced Human-System Interface Design Guidelines*. US Nuclear Regulatory Commission, Washington, D. C., July, 1994.

NUREG/CR-6501 *Human Factors Engineering Guidelines for the Review of Advanced Alarm Systems*. US Nuclear Regulatory Commission, Washington, DC., September, 1994.

Regulatory Guide 1.33, *Quality Assurance Program Requirements*. Revision 2, US Nuclear Regulatory Commission Washington, D. C.

ANSI/ANS-3.5-1993, Nuclear Power Plant Simulator: For Use in Operator Training and Examination

** TX CONFIRMATION REPORT **   AS OF   JAN 21 '97 11:22   PAGE.01

WETSO/RM 468 EC EAST

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10	01/21	11:20	516 344 4900	G3--S	02'00"	003		OK

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09	01/21	11:13	301 504 2222	G3--S	01'55"	003		OK

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03	01/21	10:04	301 504 2222	G3--S	06'37"	012	030	OK
04	01/21	10:11	516 344 4900	G3--S	07'12"	012	030	OK

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Comments: Jim and John,

Attached is a draft of revision 2 to AP3.5, "Design Reviews" and Rev. 20 to the "AP600 Program Procedure MATRIX". Both of these were referenced in the letter and associated package that provided the advance draft of the new WCAP addressing the need to document selected WCAP-12601 procedures. These help address the element 1 open items. Please refer to the Westinghouse letter of January 13, 1997.

Thank You,

Steve Kerch

Phone Number of Receiving Equipment:

301-415-2222 Jim Bongarra
516-344-4900 John O'HARA

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Helen, 1/21/97

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Comments: Jim and John,

Attached is a draft ...5, "Design Reviews" and Rev. 2 to the ... S Procedure MATRIX". Both of these were re... letter and associated package that provided the advance draft of the new WCAP addressing the need to document selected WCAP-12601 procedures. These help address the element 1 open items. Please refer to the Westinghouse letter of January 13, 1997.

Thank You,

Steve Kerch

Phone Number of Receiving Equipment:

301-415-2022 Jim Bongarra
516-344-4900 John O'HARA

Westinghouse Electric Corporation
Nuclear Projects Division

AP600

Program Operating Procedure

Subject: DESIGN REVIEWS

Approved: H. J. Bruschi, General Manager
Nuclear Projects Division

Effective Date:

AUTHOR/COGNIZANT FUNCTION — Contact Manager, AP600 Quality Assurance, on questions concerning this procedure.

PURPOSE — This procedure describes the method for preparing, conducting, and documenting formal Design Reviews (DR) performed for the purpose of Design Verification. This procedure may also be used as a guide for non-verification Design Reviews.

SCOPE — This procedure applies to all Design Reviews conducted for the AP600 project.

DEFINITIONS — See Procedure ESBU 4.12

<u>General</u> — Design Reviews for the AP600 project shall be performed in accordance with procedure ESBU 4.12 of the ESBU Quality Policy/Procedure Manual with the following modifications:

1. In addition to the responsibilities established in ESBU 4.12, the Cognizant Design Manager is responsible for:

   a. obtaining an AP600 document number for the design review report, and

   b. ensuring that design review action items are entered into the AP600 open item tracking system.

2. The Cognizant Design Manager, rather than the Design Review Chairman, is also responsible for following design review action items and ensuring that they are completed.

3. The general design review checklist per ESBU 4.12 is provided for guidance. Alternate checklists may be used as deemed appropriate by the Design Review Chairman. In any case, in addition to the responsibilities established in ESBU 4.12, the Design Review Chairman is responsible for determining the applicability of the Human Factors Checklist per Appendix A of this procedure and incorporating it into the review as applicable.

4. Intermediate and Final Design Reviews shall include a review of the Preliminary and Intermediate Design Reviews (respectively) to assure closure of outstanding actions.

5. The Design Review report format is given in Appendix B of this procedure.

REFERENCES

A. ESBU Quality Policy/Procedures Manual

FORMS/EXHIBITS

AP600 Document Cover Sheet, Form 58202, Exhibit 10

APPENDICES

A. Human Factors Engineering Checklist

B. Design Review Report Format

APPENDIX A

HUMAN FACTORS CHECKLIST

A. Product/User Identification:

1. Are the objectives of the product-user system appropriately defined?

2. Are the functions required to achieve the product-user system objectives appropriately defined?

3. Are the functions shared between the user and the product allocated in a way that most effectively utilizes the capabilities of each (automation or manual or combination)?

4. Are the users' tasks appropriately defined for anticipated modes of operation?

5. Has an operating experience review been conducted to identify human factors issues encountered in previous designs so that they can be avoided in the development of the current system, or in the case of positive features, to ensure their retention?

B. Information Requirements for the Human-System Interface:

1. Are the user's information requirements clearly defined for each of the tasks defined above?

2. Do the displays, reference materials, and navigation links appear to satisfy these information requirements by providing the required amount of data with the necessary accuracy and response time?

3. Are data presented in a concise, directly usable form? If not, can the user interpret the provided data quickly and accurately enough to complete the identified tasks successfully?

4. Have the data provided to the user been limited to that which is necessary to satisfy the identified information requirements?

C. Data Presentation and Controls for the Human-System Interface (HSI):

1. Do control and display hardware and organization appear to match operational requirements as defined by utility requirements?

2. Are numeric data presented in units which the user expects and understands? Does the range of numeric displays encompass minimum and maximum operational values?

APPENDIX A (Continued)

3. Are the schemes for labeling and coding controls, displays, and data legible, meaningful, and consistent? Does the HSI design follow a set of HSI design guidelines so that there is consistency across displays and controls?

4. Does the HSI resource include features to minimize errors and facilitate users in detecting, and recovering from, potential errors they may make?

5. Are display mechanisms fault-tolerant? For example, are there provisions for loss of color in a CRT display, are there provisions for loss of an indicator light, etc.?

6. Do the displays include data quality coding to clearly indicate when sensors have failed or values are out-of-range?

D. Work Station (Operation and Control Center System; MCR, TSC, RSR, Local):

1. Do the physical dimensions of the HSI resource take into account reach, strength, and sensory limitations throughout the range of anticipated users?

2. Does the layout of the HSI resource provide an optimal arrangement for interactions between users and between the user and the equipment?

3. Do the illumination, sound, temperature, and ventilation levels permit the user to perform required tasks satisfactorily?

4. Are these provisions for the user's safety and comfort?

E. Maintenance and Repair:

1. Have the maintenance requirements of the HSI resource been evaluated and documented?

2. Do maintenance and repair tasks for the HSI resource place reasonable technical and physical demands on service personnel?

F. Design Verification:

1. Is the HSI resource evaluated through walk-through studies, simulation studies, or some analysis to verify that the product-user system objectives (see A.4 above) and functions have been achieved?

APPENDIX B

DESIGN REVIEW REPORT FORMAT

COVER PAGE — AP600 Document Cover Sheet, Form 58202

AP600 DOCUMENT NUMBER — A document number should be assigned to the Design review report in accordance with GW GMP 005, "Document Numbering Procedure."

SECTION TITLE

1 Introduction

Give data and place of design review; identify design review Chairperson, members, and secretary.

2 Scope

Define scope of the design review (e.g., "Scope was to evaluate the design impacts involved in changing from Design "A" to Design "B").

3. Summary

Sate the number of action items and provide an overview of the action item concerns.

4. Conclusion

State DR committee's conclusion(s) based on material presented in the DR meeting(s).

5. **Attachments**

a. List of all presenters and observers in attendance at the Design Review meeting(s)

b. Design Review Information Sheet(s)

c. Design Review agenda which identifies the items presented in the Design Review meeting(s)

d. Action Item Chits issued

e. List and copy of the Design Review presentations

NUCLEAR PROJECTS DIVISION

PROCEDURE MATRIX	Rev. 20 ~~19~~

Westinghouse Electric Corporation
Advanced Technology Business Area

AP600

Program Procedure Matrix

Subject:	AP600 PROGRAM PROCEDURE MATRIX
Approved: H. J. Bruschi, ~~General Manager Advanced Technology Business Area~~	Effective Date: ~~07-01-96~~

ESBU QUALITY MANAGEMENT SYSTEM (QMS) DOCUMENT, WHICH HAS BEEN ACCEPTED BY THE US NUCLEAR REGULATORY COMMISSION.

The Westinghouse commitments to the quality assurance requirements of NQA-1-1989 Edition through NQA-1b-1991 Addenda for the AP600 program are established in the ~~US Nuclear Regulatory Commission accepted topical report, WCAP-8370, "Energy Systems Business Unit/Power Generation Business Unit Quality Assurance Plan."~~ The application of THE QMS ~~WCAP-8370~~ to the AP600 program is described in WCAP-12600, "AP600 Advanced Light Water Reactor Design Quality Assurance Program Plan," which has been accepted by the Department of Energy (DOE) for the Design Certification Project and by Advanced Reactor Corporation (ARC) for the First-of-a-kind Engineering (FOAKE) Project.

For AP600 quality-related activities performed by the Westinghouse NUCLEAR PROJECTS DIVISION (NPD) ~~Advanced Technology Business Area (ATBA)~~ and the Nuclear SERVICES ~~Technology~~ Division (NSD) ~~(NTD)~~, these commitments are satisfied by implementing the applicable Level 2 Energy Systems Business Unit (ESBU) QUALITY procedures of the ESBU Policy/Procedures Manual (QPPM) ~~(PPM)~~, the applicable Level 2 division procedures from Section 5 Appendix A of the ESBU PPM, and additional project-specific procedures that address unique program requirements and implementation methodology. The AP600 Program Operating Procedures Manual, WCAP-12601, controls these project-specific procedures. In accordance with the QPPM ~~ESBU PPM~~, the procedures contained in WCAP-12601 are designated as Level 3. Existing Level 3 procedures from other manuals are also implemented where appropriate.

The attached AP600 Program Procedure Matrix is provided to show the relationship between the above described procedures and identify the procedures to be implemented on this program. The Matrix also shows the applicability of AP600 procedures to design organizations external to ~~ATBA/NTD.~~ NPD/NSD This Matrix is updated as required to reflect changes in the body of implementing procedures.

Quality Assurance Program Elements (Ref NQA-1)	Level 2 Implementing Procedures (1) ESBU Policy / Procedures Manual	Level 3 Implementing Procedures (2) (3) (4)
I ORGANIZATION	The NQA-1 requirements for organization are addressed through organization charts AP600 management responsibilities are described in WCAP-12600, AP600 QA Program Plan.	
II QUALITY ASSURANCE PROGRAM	P/P 1.1 Management Review P/P 2.2 Project Quality Plan P/P 4.16 Design Planning and Project Development P/P 18.1 Training	AP-2.1 Indoctrination and Training
III DESIGN CONTROL	P/P 4.4 Reactor Coolant System Configuration P/P 4.10 Design Specifications WP-4.5 Design Analysis WP-4.17 Design Verification by Independent Review or Alternate Calculations WP-4.18 Design Verification by Testing WP-4.19 Computer Software Development Process WP-4.19.1 Verification and Validation of Computer Software WP-4.19.2 Configuration Control of Computer Programs WP-4.19.3 Software Error Reporting & Resolution WP-4.19.4 Dedication and Installation of External Computer Software WP-4.19.5 Single Application and Small Internal Use Computer Programs WP-4.19.6 Maintenance of Configured Computer Programs	* AP-3.1 AP600 Systems Specification Documents # AP-3.2 Design Configuration Change Control for the AP600 Program * AP-3.4 Functional Specification + AP-3.5 Design Reviews * AP-3.6 AP600 Design Criteria Documents * AP-3.7 Interface Control Document * AP-3.8 Design Specification (Component/Software) * AP-3.9 Preparation and Control of Drawings AP-3.10 Fluid Systems Design AP-3.12 Engineering Database Access # AP-3.13 Safety/Seismic Classification AP-3.14 Plant & Instrument Control System AP-3.15 AP-3.15 System Piping & Instrument Control System # AP-3.16 Calculation Numbering & Filing # AP-3.17 AP600 Component Numbering AP-3.18 System Process Flow Diagram (PFD) Preparation AP-3.21 ASME Piping Design Specification
IV. PROCUREMENT DOCUMENT CONTROL	P/P 6.1 Control of Purchased Items and Services	

Quality Assurance Program Elements (Ref. NQA-1)	Level 2 Implementing Procedures (1) ESBU Policy / Procedures Manual	Level 3 Implementing Procedures (2) (3) (4)
V INSTRUCTIONS, PROCEDURES AND DRAWINGS	P/P 2.1 Policies and Procedures WP-5.3 Preparation and Control of Drawings and Engineering Sketches	AP-5.1 SSAR Preparation Procedure AP-5.2 PRA Preparation Procedure # AP-5.3 AP600 Tier 1 Document Development # AP-5.4 SSAR/PRA/ITAAC Procedure for Responding To A Request For Additional Information (RAI) # AP-5.6 Release of Documentation to NRC in Support of AP600 Design Certification
VI DOCUMENT CONTROL	P/P 5.2 Document Control WP-5.3 Preparation and Control of Drawings and Engineering Sketches	AP-0.0 Preparation and Control of Procedures # AP-6.1 Document Numbering # AP-6.2 Technical Document Release & Control # AP-6.3 Preparation, Review & Approval of AP600 Documents
VII CONTROL OF PURCHASED ITEMS AND SERVICES	P/P 6.1 Control of Purchased Items and Services	AP-7.1 Supplier Evaluation, Audit, and Approval # AP-7.2 Control of Subcontractor Submittals AP-7.3 Control of AP600 Contributed Labor AP-7.4 Auxiliary Equipment Design and Costing Process ESBU QA Procedures Manual ESBU-QA-7.1 Evaluation and Qualification of ESBU Suppliers ESBU-QA-7.2 Supplier Audits
XI TEST CONTROL	WP-18.0 Test Control	AP-3.11 AP600 Testing
XII CONTROL MEASURING AND TEST EQUIPMENT	WP-11.1 Control of Inspection, Measuring and Test Equipment	
XV CONTROL OF NONCONFORMING ITEMS	WP-13.3 Deviation Notices	

Quality Assurance Program Elements (Ref . NQA-1)	Level 2 Implementing Procedures (1) ESBU Policy / Procedures Manual	Level 3 Implementing Procedures (2) (3) (4)
XVI CORRECTIVE ACTION	P/P 14.1 ESBU - Significant Quality Issues WP-13.2 Control of Nonconformances WP-14.2 Corrective and Preventive Action	AP-16.1 Customer Feedback
XVII QUALITY ASSURANCE RECORDS	P/P 16.1 Records	WCAP-14530, ESBU Information and Records Management Program Manual: IRM-1.1 Organization and Responsibility IRM-3.2 Protection of Records on Optical Disk
XVIII AUDITS	P/P 17.1 Assessments	AP-18.1 Self-Assessments
(5) SUPPLEMENTAL PROGRAM DOCUMENTS	P/P 21.0 Identification and Reporting of Conditions Adverse to Safety	

NOTES

(1) Level 2 ESBU procedures are identified in this matrix with a "P/P" prefix. Level 2 division procedures from Section 5 Matrix A of the ESBU Policy/Procedures Manual are identified with a "WP" prefix.

(2) Level 3 procedures in WCAP-12601 are identified with an "AP" prefix. Other Level 3 procedures are as specified.

(3) The prefix for Level 2 NPD procedures was changed from "DP" to "WP" with the phase-out of WCAP-9565 effective 8/14/95. Where Level 3 procedures refer to "DP" procedures, the reference shall be considered as shown in the "Procedure Cross References" per pages 5 and 6 of this matrix. Revisions to procedures are not required for the sole purpose of changing Level 2 references, however, the references shall be corrected when the Level 3 procedures are revised.

(4) Procedures that apply to design organizations external to (W) ~~ATBA~~ NSD and ~~NTD~~ NSD are identified with a "*" or "#" as follows:

* These procedures apply only with respect to document format and content requirements.

\# These procedures apply only with respect to definition of interface responsibilities.

\+ This procedure applies only with respect to Human Factors requirements.

Procedures not marked with a "*", "#" or "+" do not apply to design organizations external to (W) NPD.

(5) Not an NQA-1 criterion.

AP600 PROGRAM PROCEDURE MATRIX REV. 20 [mm/dd/yy]
PROCEDURE CROSS-REFERENCES

The following table identifies the ESBU Quality Policy/Procedures Manual references to be used for the "DP" procedure references in WCAP-12601.

PROCEDRURE	REFERS TO	CHANGE TO
AP-0.0	WCAP-9948	WCAP-14530 IRM1.1
	WCAP-9565	ESBU Quality Policy/Procedures
AP-2.1	WCAP-9565	ESBU Quality Policy/Procedures
AP-2.2	DP-2.4	ESBU 3.1
	DP-4.0	ESBU 6.1
	WCAP-13740	ESBU Quality Management System
AP-3.1	DP-3.2.2	ESBU 4.10
AP-3.7	DP-3.2.6	WP-5.3
AP-3.8	DP-3.2.2	ESBU 4.10
	DP-3.7.1	WP-4.19
	DP-3.7.2	WP-4.19.1
	DP-3.7.3	WP-4.19.2
	DP-3.7.4	WP-4.19.3
	DP-3.7.5	WP-4.19.5
AP-3.9	DP-3.2.6	WP-5.3
AP-3.10	DP-3.2.8	WP-4.5
	DP-3.3.2	WP-4.17

AP600 PROGRAM PROCEDURE MATRIX REV. 20 [mm/dd/yy]
PROCEDURE CROSS-REFERENCES

PROCEDRURE	REFERS TO	CHANGE TO
AP-3.11	WCAP-9565	ESBU Quality Policy/Procedures
	DP-4.0	ESBU 6.1
	DP-3.3.3	WP-4.18
	DP-11.0	WP-4.18
	DP-12.0	WP-11.1
AP-3.14	DP-3.2.2	WP-4.17
	DP-3.3.3	WP-4.18
	DP-7.0	ESBU 6.1
AP-3.15	DP-3.2.8	WP-4.5
AP-3.16	DP-3.2.8	WP-4.5
AP-3.21	WP-3.2.2	WP-4.17
AP-6.2	DP-6.0	ESBU 5.2
AP-7.1	ESBU-QA-2.2	ESBU-QA-18.2
	ESBU-QA-7.1	ESBU-QA-6.5
	ESBU-QA-7.2	ESBU-QA-6.6
	WCAP-9565	ESBU Quality Policy/Procedures
AP-7.2	DP-4.0	ESBU 6.1
	DP-7.0	ESBU 6.1
AP-18.1	DP18.1	Delete
	WCAP-8370 Part C	Delete

FAX to TED QUAY

January ~~16~~, 1997
30

CC: Diane Jackson
Tom Kenyon
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #134 (M3.11-1)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #134 (M3.11-1). The balance of the relevant information is contained in SSAR subsection 3.11 and Appendix 3D. We provided a revision to Appendix D of the SSAR on February 29, 1996, and believed it was acceptable. We discussed this topic with NRC and made a further revision to SSAR subsection 3.11 on June 19, 1996 (over 6 months ago). Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#134) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

Summary

Date: 1/16/97

Item # / Description / Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. / Date
M3.11-1 (EQUIPMENT QUALIFICATION) The Nuclear Regulatory Commission (NRC) staff does not agree with the assertion in the response to Q270.2 that qualification to the 1983 revision is equivalent to qualification to the 1974 revision. There are significant differences between the two versions, for example, see the staff's comment on the response to Q270.14 below (M3.11-11). Therefore, this response in unacceptable. Closed - The 1974 version for equipment covered by 10CFR50.49 was referenced in SSAR Appendix 3D, in both the introduction and subsection 3D.4.1, Revision 5.	Miller,D.	Closed	Action W	

M1G-OI

Records: 1

COVER SHEET

FAX

To: W. Huffman (NRC), C. Fineman (INEL), L. Hochreiter (PSU)
cc; J. Butler (Informal NRC Correspondence File), R. Kemper
Subject: Today's telecon on WC/T CAD
Date: January 22, 1997
Pages: Two, including this cover sheet.

COMMENTS:

Folks,

Attached is additional information for our discussions today at 2pm. Speak to you then.

Earl

From the desk of...

Earl H. Novendstern
Manager, Advanced and VVER Plant Safety Analysis
Westinghouse
PO Box 355
Pittsburgh, PA 15235

(412) 374 -4790
Fax: (412) 374-5744

12.a. In the event that fuel grid deformation becomes a concern for AP600, Westinghouse will address its impact on the large break LOCA analysis.

If Westinghouse fuel of a different design or another vendor's fuel is placed into AP600 in the future, an evaluation will be performed of the mixed core. The evaluation will consider any differences in dimensions, hydraulic resistances and burnup effects between the fuel types to be loaded.

12.1. The Table 4.5-1 values indicate the condensation multipliers which are modeled in the WCOBRA/TRAC global model matrix to be run. The wider range of values (0.4 to 1.05) cited on p.4-17 is applied in the uncertainty methodology determination of the 95th percentile PCT.

FAX to TED QUAY

January 23, 1997

CC: Sharon, please make copies for:

Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre

As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
4 (RAI 410.262)	9/10/96	1/9/97
21 (RAI 471.24)	5/20/96	1/9/97
30 (RAI 952.99)	5/13/96 12/17/96 - repeat	1/13/97
37 (RAI 260.74)	4/22/96	1/14/97
123 (M3.6.1-2)	4/30/96	1/15/97
134 (M3.11-1)	2/29/96 6/19/96 - more	1/16/97
135 (M3.11-2)	2/29/96 6/19/96 - more	1/17/97

Thanks for your help.

Jim

Jim Winters

FAX to TED QUAY

January 23, 1997

CC: Sharon, please make copies for: Diane Jackson
Tom Kenyon

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #137 (M3.11-4)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #137 (M3.11-4). We provided a revision to SSAR subsection 3.11 on June 19, 1996 (over 6 months ago). Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#137) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Project Management Summary

Date: 1/23/97

Selection: [item no] between 137 And 137 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	Draft	Review	Transmit
37	NRR/SPLB	3.11	MTG-OI	Fanto / Miller,D.	1	Closed	Action W	12/13/94 A			

M3.11-4 (EQUIPMENT QUALIFICATION)
The expression "Demonstration of qualified life by test and analysis (or both)...." provided in the response to Q270.5 is not clear. If the intent is to state that "demonstration of qualified life by test or test and analysis..." then the intent is acceptable and the SSAR should be corrected to say this. In addition the COL Applicant should not be obligated to suppliers as implicated in the above SSAR revision, they should be able to conduct qualification test themselves if they choose to do so.

See items 134 and 135.
DISCUSSED AT 12/13/94 MEETING BETWEEN WESTINGHOUSE AND NRC PLANT SYSTEMS BRANCH. SEE NRC MEETING SUMMARY FOR SPECIFICS SUPPORTING RESULTING STATUS.

Closed - Westinghouse has revised the SSAR to resolve this issue.

4 of 2

Abnormal environmental conditions are those plant conditions for which the equipment is designed to operate for a period of time without accelerating normal periodic tests, inspections, and maintenance schedules for that equipment. The maximum and minimum conditions identified as the abnormal condition are based on the design limits for the affected areas.

Design basis accident (DBA) and post-design basis accident conditions are those plant conditions resulting from various postulated equipment and piping failures during which the identified equipment must operate without impairment of the function. The design basis accident and post-design basis accident conditions are discussed in Appendix 3D. Qualification is based on component specific calculations when necessary.

Compatibility of equipment with the specified environmental conditions is achieved by the following.

Systems and components required to mitigate the consequences of a design basis accident or to perform safe shutdown operation are qualified to remain functional after exposure to the environmental conditions in Table 3D.5-5.

Environmentally qualified equipment exposed to a harsh environment has a qualified life goal of 60 years. Demonstration of qualified life by test or test and analysis is provided by equipment suppliers to the Combined License applicant, to address applicable aging effects. For critical components susceptible to aging, a qualified life is established that includes the effects of the total integrated radiation dose experienced at their respective locations within the plant. When a 60-year qualified life is not achievable, a shorter qualified life is established, and a replacement program is implemented.

For equipment located in a mild environment, a design life goal is established by using known significant aging mechanisms and reliability data.

Equipment qualification takes into account the most severe environmental conditions resulting from the design basis high-energy line break. Included in these conditions are the short-term peak transient temperature following a main steamline break (MSLB) and a radiation exposure and temperature due to a loss of coolant accident (LOCA) within the reactor containment.

Postulated high-energy line failures as defined in subsection 3.6.2.1.2 are assumed in areas where high-energy lines greater than 1 inch are routed. Essential equipment is protected against the effects of jet impingement (subsection 3.6.2.4.1) and evaluated for spray effects if required (subsection 3.6.2.7).

Active mechanical equipment is qualified for operability as discussed in subsection 3.9.3 and Section 3.10. This operability program, combined with the qualification of the electrical appurtenances (valve operators, solenoids, limit switches), demonstrates qualification under required environmental conditions. Active mechanical equipment is defined as equipment that performs a mechanical motion as part of its safety-related function.

Equipment areas outside containment and outside the main control room are maintained at normal environmental conditions by nonsafety-related HVAC systems. If these systems are disabled, the heat generated by this equipment is absorbed by the surrounding concrete with an ambient temperature rise that does not exceed the abnormal condition. Normal HVAC is restored within 72 hours or temporary ventilation is provided as discussed in Section 6.4.

If the normal nonsafety-related main control room HVAC is lost, the heat generated by equipment and people is absorbed by the surrounding concrete. Normal heating, ventilation, and air-conditioning is restored within 72 hours or temporary ventilation is provided as discussed in Section 6.4.

3.11.4 Estimated Radiation and Chemical Environment

The plant-specific estimates of the radiation dose incurred by equipment during normal operation is shown in Table 3D.5-2 and the estimated doses following a loss-of-coolant accident are defined in Table 3D.5-5.

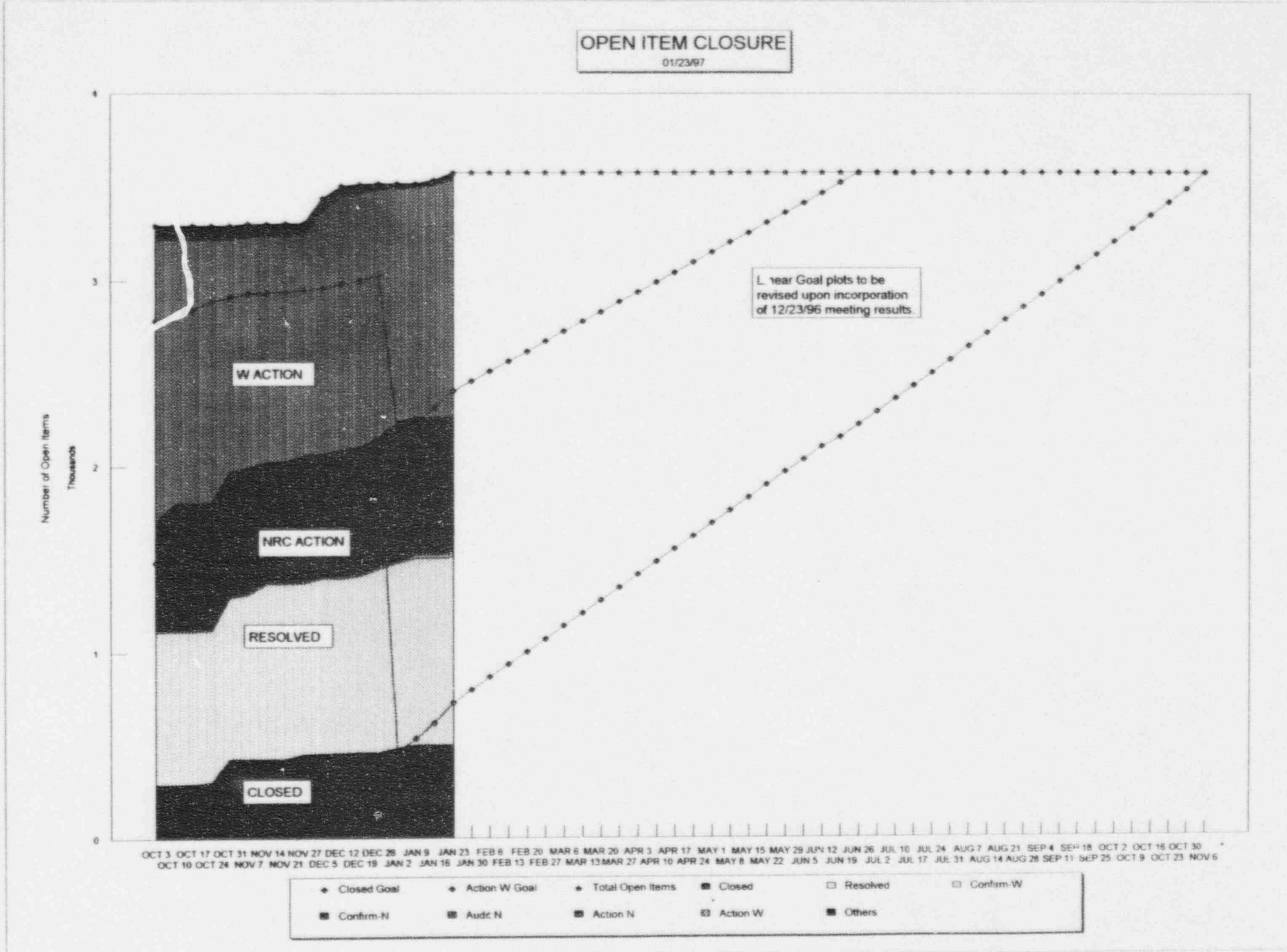
The identified equipment is qualified to perform functions in the radiation environments present during normal and design basis accident conditions. The normal operational exposure is based upon design source terms presented in Chapter 11 and subsection 12.2.1. The equipment and shielding configurations are presented in Section 12.3. Post-accident monitoring, reactor trip and engineered safety features system and component radiation exposures are dependent on the location of the equipment in the plant. Source terms and other accident parameters are presented in subsection 12.2.1 and Chapter 15.

The maximum combined integrated radiation dose inside containment is based on the effects of the normally expected radiation environment (gamma) over the equipment's installed life plus that associated with the most severe design basis event (gamma and beta) during or following which the equipment is required to remain functional.

The chemical environment following a loss of coolant accident is primarily based on the chemistry of the reactor coolant system fluid since there is no caustic containment spray. Sump pH adjustments are considered for certain qualification tests. This is discussed further in Appendix 3D.

3.11.5 Combined License Information Item for Equipment Qualification File

The Combined License applicant is responsible for the maintenance of the equipment qualification file during the equipment selection and procurement phase.


OPEN ITEM CLOSURE
01/23/97
W ACTION
NRC ACTION
RESOLVED
CLOSED
L. near Goal plots to be revised upon incorporation of 12/23/96 meeting results.
Number of Open Items
Thousands
0
1
2
3
Closed Goal
Action W Goal
Total Open Items
Closed
Resolved
Confirm-W
Confirm-N
Action N
Action W
Others

FAX to TED QUAY
January 23, 1997

CC: Sharon, please make copies for: Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

Robin Nydes
Cindy Haag
Don Lindgren
John Butler
Bob Tupper
Bruce Rarig
Brian McIntyre

NRC is requested to please acknowledge receipt of information related to each of the following Open Items. Recognizing that reviewing for completeness of the response in each case constitutes an NRC action, we recommend that receipt acknowledgement be accompanied by direction to change their "NRC Status" to "Action N". This is the second weekly request of this type.

4, 21, 30, 37, 123, 134, 135, 137, 140, 157, 158, 164, 182, 184, 262, 293, 300, 305, 308, 319, 457, 458, 801, 802, 805, 807, 809, 972, 973, 1009, 1037, 1038, 1039, 1040, 1041, 1043, 1045, 1052, 1053, 1055, 1101, 1102, 1195, 1197, 1317, 1458, 1461, 1697, 1698, 1699, 1700, 1701, 1702, 1703, 1704, 1707, 1716, 1727, 1730, 1731, 1742, 1745, 1747, 1749, 1753, 1760, 1996, 1999, 2018, 2019, 2023, 2024, 2025, 2040, 2045, 2051, 2199, 2200, 2201, 2202, 2272, 2273, 2442, 2457, 2962, 2963, 2964, 2965, 2966, 2967, 2968, 2969, 2970, 2971, 2972, 2973, 2974, 2975, 2976, 2977, 2978, 2979, 3122, 3126, 3127, 3128, 3197, 3372, 3505, 3517, 3895, 3944, 3945, 3946, 3947, 3948, 3949, 3950, 3951, 3952, 3953, 3954, 3955, 3956, 3957, 3958, 4123, 4124, 4125, 4126, 4127, 4128, 4129, 4130, 4131, 4132, 4133, 4134, 4135, 4136, 4137, 4138, 4139, 4140, 4141, 4142, 4143, and 4144.

Thanks
Jim Winters
412-374-5290

Jim

FAX to TED QUAY

January 23, 1997

CC: Sharon, please make copies for:

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre
Diane Jackson
Tom Kenyon

OPEN ITEM #140 (M3.11-7)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #140 (M3.11-7). We provided a revision to SSAR subsection 3D.4.5.4 on February 29, 1996 (over 10 months ago). Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#140) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/23/97

Selection: [item no] between 140 And 140 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
140	NRR/SPLB	3.11	MTG-OI	M3.11-7 (EQUIPMENT QUALIFICATION) It is the NRC staff's position that review of Section 3D.4.5.4 of the SSAR requires the staff to develop it's position on the extension of the life of nuclear power plants beyond 40 years before it can address this topic in the AP600 design in a meaningful way. The staff believes that the development of it's position will conflict with the AP600 review schedule. Therefore, the staff recommends removal of this section from the AP600 design certification review. Closed - The title of section 3D.4.5.4 was changed to "Qualified Life Reevaluation" in Revision 5.	Miller,D.	Closed	Action W		

Total Records: 1

Common practice for the evaluation of irradiation-induced degradation is to consider the sum of estimated life and the accident radiation doses before design basis event testing. When testing, the total dose is applied during the radiation aging simulation portion of the qualification test sequences. This is considered conservative because the equipment has accumulated an exposure, or total integrated dose, before the initiation of the seismic and accident environment testing. Further bases for test dose determination are provided in Subsection 3D.5.1.2. Sufficient margin must be included in test parameters (see Subsection 3D.4.8). The same margins are applied in an analysis of radiation life or design basis event radiation dosage.

The simulation of age also includes the effects of operational cycling, both electrical and mechanical. Generally, these considerations are applied specifically to electromechanical equipment such as valve operators, limit switches, motors, relays, switches, and circuit breakers. Furthermore, the simulation of these effects is waived where existing data demonstrates equipment durability greatly in excess of estimated number of operating cycles for Class 1E service. Analysis or justification is provided for any case where operational cycling is omitted in the test sequence.

It is not practicable to simultaneously simulate the aspects of aging. Development of each test plan considers known synergies and sequences the simulation of the various applicable aging mechanisms with regard for conservatism of the overall effect on the test specimens.

3D.4.5.4 Qualified Life Reevaluation

It may be possible to extend the qualified life of a particular piece of equipment at some future date by comparing the actual in-plant environments and conditions during the equipment's installed life to the values assumed for the AP600 in establishing the qualified life. For example, the thermal qualified life might be extended by performing an analysis of actual internal or external temperatures (or both) experienced. Continuous temperature monitoring or use of sample devices for testing and trending materials aging may be used. These efforts reveal the conservatism of the original thermal life calculation, which assumes that the maximum value specified for the normal plant operating environment endured at all times.

Although a strict Arrhenius calculation may yield an extended qualified life, care is taken in using this extrapolation because of uncertainties in the methodology. The Arrhenius time-temperature relationship relies on empirically determined activation energies of materials. This parameter has been determined for a number of materials to at least a good approximation for small temperature extrapolations. Extrapolation of the Arrhenius model to time periods of temperature beyond the range of materials test data is questionable and may result in large errors.

Calculated qualified lives based on this methodology should be limited to 20 years unless sound technical bases can be cited. This position is consistent with industry guidelines such as IEEE 98-1984, NUREG/CR-3156 (Reference 4), and EPRI NP-1558 (Reference 5).

FAX to TED QUAY

January 24, 1997

CC: Sharon, please make copies for: Diane Jackson
Tom Kenyon

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #139 (M3.11-6)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #139 (M3.11-6). We provided a revision to SSAR Appendix 3D on February 29, 1996 (over 10 months ago) and subsection 3.11.2.1 on June 19, 1996 (over 7 months ago). This particular item, the use of IEEE 323-1974, was accepted by Westinghouse over a year ago and it seems appropriate that NRC should at least acknowledge receipt of our resolution. Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#139) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/24/97

Selection: [item no] between 139 And 139 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
139	NRR/SPLB	3.11	MTG-OI	M3.11-6 (EQUIPMENT QUALIFICATION) With respect to the response to Q270.7, as indicated above, the NRC staff does not agree that IEEE Standard 323-1974 is essentially identical to IEEE Standard 323-1983 and has not approve the use of IEEE Standard 323-1983. Therefore Westinghouse's position on this issue is unacceptable. Resolved - The introduction to SSAR Appendix 3D and SSAR subsections 3D.1 and 3D.4.1, Revision 5, includes the proper references and descriptions of IEEE 323-1974. The Revision 8 of SSAR Subsection 3.11.2.1 will include appropriate changes.	Miller	Closed	Action W		

2 of 6

Total Records: 1

Nonactive mechanical equipment whose only safety function is structural integrity is designed according to ASME Code guidelines. The accident and post-accident environmental effects are considered in the design of such structural components as pump casings and valve bodies.

The environmental qualification program is restricted to evaluating the design of critical nonmetallic subcomponents of active devices in a harsh environment, where failure results in loss of the active component.

3.11.1.3 Equipment Operability Times

For the AP600 Class 1E electrical and active mechanical equipment, post-accident operability times are shown in Table 3D.4-2 in Appendix 3D.

Specific information for each device qualified as part of the IEEE 323-1974 qualification program is contained in the appropriate equipment qualification data package.

The active mechanical component is qualified for operability as discussed in Section 3.10, using test, analysis, or a combination of tests and analyses. This operability program, combined with the qualification of the electrical appurtenances (for example, valve operators) discussed in the appropriate equipment qualification data packages, demonstrates qualification.

3.11.1.4 Standard Review Plan Evaluation

A discussion of the Standard Review Plan requirements in regard to environmental qualification of mechanical equipment is provided in subsection 1.9.2.

3.11.2 Qualification Tests and Analysis

3.11.2.1 Environmental Qualification of Electrical Equipment

The AP600 approach for environmental qualification of Class 1E equipment is outlined in Appendix 3D. This methodology is developed based on the guidelines provided in IEEE 323-1974 (Reference 1), and 344-1987 (Reference 2).

Qualification for equipment in a harsh environment is based on type testing or testing and analysis. Analysis may be used to determine significant aging mechanisms in mild environment applications. Type testing includes thermal and mechanical aging, radiation, and exposure to extremes of environmental, seismic, and vibration effects. Type testing is done with representative samples of the production line equipment according to the sequence indicated in IEEE 323-1974 to the specified service conditions, including margin. The testing takes into account normal and abnormal plant operation and design basis accident and post-design basis accident operations, as required.

When reliable data and proven analytical methods are available, environmental qualification may be based on analysis supported by partial type test data. This method includes justification of the methods, theories, and assumptions used (that is, mathematical or logical

APPENDIX 3D

Methodology for Qualifying AP600 Safety-Related Electrical and Mechanical Equipment

Safety-related electrical equipment is tested under the environmental conditions expected to occur in the event of a design basis event. This testing provides a high degree of confidence in the safety-related system performance under the limiting environmental conditions. Qualification criteria were revised by IEEE 323-1974 (Reference 1) and by Regulatory Guide 1.89, which endorses this IEEE standard. The concept of aging was highlighted in IEEE 323-1974, and interpretation of the scope of aging and implementation methods were subsequently developed. 10CFR 50.49 provides the NRC requirements for qualification of equipment located in potentially harsh environments. Therefore, the guidance provided by IEEE 323-1974 is the evolutionary root of requirements, recommended methods, and qualification procedures described in this appendix.

Specific treatment of seismic qualification, part of the qualification test sequence recommended in IEEE 323-1974, is addressed in IEEE 344-1987 (Reference 2). This appendix bases technical guidance, recommendations, and requirements for seismic qualification on IEEE 344-1987.

The AP600 Equipment Qualification methodology addresses the expanded scope of IEEE 627-1980 (Reference 3), which encompasses the qualification of Class 1E electrical and safety-related mechanical equipment. IEEE 627 generalizes the principles and technical guidance of IEEE 323 and 344. Compliance with the IEEE 323-1974 and 344-1987 is the specific means of compliance with the intent of IEEE 627-1980 for safety-related electrical and mechanical equipment.

Safety-related electrical and mechanical equipment is typically qualified using analysis, testing, or a combination of these methods. The specific method or methods used depend on the safety-related function of the equipment type to be qualified. Safety-related mechanical equipment, such as tanks and valves, is typically qualified by analysis, with supplementary functional testing when functional operability is demonstrated only through testing, as is the case for active valves. Testing is the preferred method for environmental and seismic qualification of safety-related (Class 1E) electrical equipment.

The technical discussions of this appendix follow the format headings of the equipment qualification data packages (EQDPs) to be issued as specific qualification program documentation. This formatting (see Section 3D.7) permits easy cross-reference between the methodology defined in this report and the detailed plans contained in the equipment qualification data packages. Attachment A of this appendix is the format used for the equipment qualification data package.

Attachment B of this appendix, "Aging Evaluation Program," describes methods for addressing potential age-related, common-mode failure mechanisms used in AP600 equipment qualification programs. The approach conforms with current industry positions and makes

maximum use of available data and experience in the evaluation, test, and analysis of aging mechanisms.

Attachment C, "Effects of Gamma Radiation Doses Below 10^4 rads on the Mechanical Properties of Materials," provides the basis that radiation aging below 10^4 rads is not a significant factor in the ability of the equipment to perform properly during a seismic event. For some devices, electrical properties are degraded below 10^3 rads. Radiation aging for equipment not required to perform a safety-related function in a high-energy line break environment and subject to lifetime doses of less than 10^4 rads is not addressed in AP600 test programs.

Attachment D, "Accelerated Thermal Aging Parameters," describes the methodology employed in calculating the accelerated thermal aging parameters used in this program.

Attachment E, "Seismic Qualification Techniques," discusses available methods for establishing a seismic qualification basis, by either test or analysis, and its application to the qualification of safety-related equipment for the AP600.

3D.1 Purpose

The basic objectives of qualification of safety-related electrical and mechanical equipment follow:

- To reduce the potential for common cause failures due to specified environmental and seismic events
- To demonstrate that safety-related electrical and mechanical equipment is capable of performing its designated safety-related functions.

This appendix describes the methodology that has been adopted to qualify equipment according to IEEE 627-1980, "IEEE for Design Qualification of Safety System Equipment Used in Nuclear Power Generating Stations." The two standards primarily used to demonstrate compliance with this standard are IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE 344-1987, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

3D.2 Scope

The qualification criteria, methods, and environmental conditions described herein constitute the methodology that has been adopted to comply with the forenamed standards for the AP600. This methodology applies to safety-related, seismic Category I electrical and mechanical equipment and is also utilized for certain monitoring equipment. Seismic Category II equipment is not within the scope of this program.

Performance during abnormal environmental conditions, while not specifically designated as an industry or a regulatory qualification requirement, is also addressed by this appendix. Performance during normal service conditions is demonstrated by tests and inspections addressed by the equipment specification. Electromagnetic interference (EMI) testing or analysis is not included in the qualification process and is addressed on an individual equipment basis, as necessary.

3D.3 Introduction

This appendix identifies qualification methods used for the AP600 to demonstrate the performance of safety-related electrical and mechanical equipment when subjected to abnormal and accident environmental conditions including loss of ventilation systems, feedline, steamline and main coolant system breaks, and seismic events. This appendix provides the expected conditions for various locations in the AP600. General requirements for the development of plans/procedures/reports are also provided. Section 3D.4 identifies the various industry and regulatory criteria upon which the program is based. Section 3D.5 defines the design specifications and applicable test environments. Section 3D.6 defines the basis for the qualification method selection. Section 3D.7 outlines the documentation requirements.

3D.4 Qualification Criteria

The environmental requirements considered in the design of safety-related equipment are embodied in GDC 2, "Design Bases for Protection Against Natural Phenomena"; GDC 4, "Environmental and Missile Design Bases"; and GDC 23, "Protection System Failure Modes." GDC 1, "Quality Standards and Records," and Section III, "Design Control," of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to require that the environmental design of safety-related equipment is verified, documented, and controlled.

The qualification methods described in this appendix are used to verify the environmental design basis and capability of the safety-related electrical and mechanical equipment supplied for the AP600. The results of the verification, as well as the design basis for each equipment, is documented in an equipment qualification data package. (See Attachment A for sample format.) Design control is performed through the AP600 Quality Assurance Program. (See Chapter 17.)

3D.4.1 Qualification Guides

IEEE 323-1974 and 344-1987 serve as the basis upon which the AP600 equipment qualification methodology demonstrates compliance with IEEE 627-1980. NRC regulations stated in 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," and NRC guidance provided in Regulatory Guide 1.89, and Regulatory Guide 1.100, endorse IEEE 323-1974 and IEEE 344-1987, respectively. The intent of the more general IEEE 627-1980 is addressed through conformance with IEEE 323 and 344.

FAX to TED QUAY

January 24, 1997

CC: Sharon, please make copies for: Diane Jackson
Tom Kenyon

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #141 (M3.11-8)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #141 (M3.11-8). We provided a revision to SSAR subsection 3D.4.8.2 on February 29, 1996 (over 10 months ago). This was a very straightforward change and implemented into the SSAR as agreed. It seems a reasonable request that NRC acknowledge receipt of the change. Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#141) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

123

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/24/97

Selection: [item no] between 141 And 141 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
141	NRR/SPLB	3.11	MTG-OI	M3.11-8 (EQUIPMENT QUALIFICATION) The staff does not agree that the discussions in Subsections 3D.4.6, 3D.4.7, and 3D.4.8 of the SSAR are consistent with NUREG-0588, RG 1.89, and past NRC positions and approvals as stated in the response to Q270.9. One of the primary reasons is that the staff has not approved the use of IEEE 323-1983 which is being use to demonstrate compliance. Further discussions between the NRC staff and Westinghouse must be conducted in order to resolve these issues. Closed - Section 3D.4.8.2, Revision5, was revised to include: "If margin is not specifically defined and quantified through conservatism in the aging parameters or calculation, then a +10 percent time margin will be included."	Miller,D.	Closed	Action W		

2 of 3

3D.4.8.2 Aging

No specific margin is applied to the time component in deriving appropriate aging parameters, if margin is included in deriving the accelerated aging parameters employed for simulating each applicable aging mechanism.

Margin may be addressed by demonstrating the adequacy of the aging simulated by test through the calculation of time-temperature equivalence (See Attachment B of this appendix) or the comparison of simulated parameters with those applicable to the intended service of the equipment. The installed life of equipment must not exceed the thermal qualified life demonstrated by this calculation. Additionally, the selection and use of the thermal aging parameters both for test and subsequent calculations are subject to criteria, including the following:

- Test temperature must endure for at least 100 hours
- Test temperature must exceed any application temperature (that is, the normal or abnormal environment in which the equipment is to be used, and for which the life is calculated)
- Test temperature must be less than state-change temperature for materials critical to the equipment safety-related function or capability to endure the subsequent design basis event testing
- A conservative activation energy is used. Activation energies for materials critical to the equipment safety-related function or capability to endure the subsequent design basis event testing are considered. Materials may have several activation energies, each for a different ~~material property~~. ~~Relevant material properties~~ are considered.

If margin is not demonstrated through conservatism in the aging parameters or calculation, then a +10 percent time margin is included.

A margin of 10 percent in the other parameters (for example, irradiation, operational cycling) applies to both the aging simulation and the post-accident simulated aging, with few exceptions.

For equipment required by design to perform its safety-related function within a short time period into the design basis event (that is, within seconds or minutes), and having completed its function, subsequent failure is shown not to be detrimental to plant safety, margin by percentage of additional time or equivalent time-temperature is not applied. Margins for trip function requirements are contained in the worst-case high-energy line break envelope. Test parameters are simulated on a real-time basis with the transient condition margins listed in Table 3D.4-3. Trip signals, once generated by the sensors, are locked in by the protection system and do not reset in the event of subsequent sensor failure.

FAX to TED QUAY

January 24, 1997

CC: Sharon, please make copies for: Diane Jackson
Tom Kenyon

Don Lindgren
Robin Nydes
Dick Miller
Brian McIntyre

OPEN ITEM #138 (M3.11-5)

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items from the oldest on. Attached are copies of some of the relevant documentation related to Open Item #138 (M3.11-5). We provided a revision to SSAR subsection 3D.4.3 on February 29, 1996 (over 10 months ago). This straightforward change incorporated the documented request of NRC. Acknowledgement of NRC's receipt of this item (#138) is requested. Although we have other open items on environmental qualification, our records show no outstanding Westinghouse action on this item (#138) and we request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N." Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/24/97

Selection: [item no] between 138 And 138 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
138	NRR/SPLB	3.11	MTG-OI	M3.11-5 (EQUIPMENT QUALIFICATION) With respect to the response to Q270.6, the staff position is that to be in compliance with the requirements of 10 CFR 50.49, qualification must be demonstrated for equipment that has not been demonstrated to be qualified. If it can be demonstrated that any equipment (including electronic) is qualified in accordance with applicable requirements it will be found acceptable. To date, the accepted position for electronic equipment, by both industry and the NRC, is that electronic equipment that experience a total integrated radiation dose in excess of 103R is considered to be in a harsh environment. Westinghouse's position on this issue is unacceptable. Closed - Section 3D.4.3, last paragraph has been revised in Revision 5 to address radiation harsh environments.	Miller,D.	Closed	Action W		

292

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.

FAX to TED QUAY

January 24, 1997

CC: Sharon, please make copies for: Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre

As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
4 (RAI 410.262)	9/10/96	1/9/97
21 (RAI 471.24)	5/20/96	1/9/97
30 (RAI 952.99)	5/13/96 12/17/96 - repeat	1/13/97
37 (RAI 260.74)	4/22/96	1/14/97
123 (M3.6.1-2)	4/30/96	1/15/97
134 (M3.11-1)	2/29/96 6/19/96 - more	1/16/97
135 (M3.11-2)	2/29/96 6/19/96 - more	1/17/97
137 (M3.11-4)	6/19/96	1/23/97
140 (M3.11-7)	2/29/96	1/23/97

Thanks for your help.

Jim Winters

FAX to TED QUAY

January 27, 1997

CC: Sharon, please make copies for: Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre

As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

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37 (RAI 260.74)	4/22/96	1/14/97
123 (M3.6.1-2)	4/30/96	1/15/97
134 (M3.11-1)	2/29/96 6/19/96 - more	1/16/97
135 (M3.11-2)	2/29/96 6/19/96 - more	1/17/97
137 (M3.11-4)	6/19/96	1/23/97
138 (M3.11-5)	2/29/96	1/24/97
139 (M3.11-6)	2/29/96 6/19/96 - more	1/24/97
140 (M3.11-7)	2/29/96	1/23/97
141 (M3.11-8)	2/29/96	1/24/97

Thanks.
Jim Winters

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 27, 1997	NAME:	JIM WINTERS
TO:	TED QUAY	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 5

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:

TED,

BASED ON A CALL WITH DIANE ON FRIDAY, WE DECIDED WE SHOULD FOCUS ON THE SECTIONS OUR JOINT NRC/W SCHEDULE SAYS SHOULD BE DONE FIRST. HERE ARE THE "ACTION W." FOR THE SECTIONS THAT SHOULD HAVE FSER DONE. I'LL CALL AT 9:30.

Jim Winters

AP600 Open Item Tracking System Database: Executive Summary **Date: 1/27/97**

Selection: [nrc st code]='Action W' And [DSER Section] like '3.5.1.3*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
586	NRR/EMCB	3.5.1.3-13	DSER-OI	(TURBINE MAINTENANCE PROGRAM & MISSILE PROBABILITY CALCULATION)Westinghouse should add COL Action Item 3.5.1.3-5 to the SSAR. Provide turbine maintenence program including probability calculations of turbine missile generation. Closed - Addressed by COL item in SSAR section 10.2.6 Action W - NRC wants time table for submittal of turbine mainteance program in SSAR.	Lindgren	Closed	Action W		

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/27/97

Selection: [nrc st code]='Action W' And [DSER Section] like '6.1*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
969	NRR/EMCB	6.1.2-1	DSER-OI		Lindgren/Schulz	Closed	Action W		
				Westinghouse should further justify the non-safety related classification of all coatings within the AP600 containment.					
				Closed - Added requirement for debris transport evaluation of coatings and description of methodology to SSAR Rev. 7, subsection 6.1.2.1					
970	NRR/EMCB	6.1.2-2	DSER-OI		Lindgren/Schulz	Closed	Action W		
				Westinghouse should provide more information indicating that the appropriate coatings will be correctly applied and will provide adequate protection throughout the plant's life. In addition, Westinghouse should supply data and an in-depth analysis to provide justify using new coating types (such as high-top coatings) in containment.					
				Closed - SSAR Rev. 7, subsection 6.1.2.1.6 includes the requirement for a program by the Combined Lisence applicant to control testing, application, and monitoring of nonsafety-related coatings.					
971	NRR/EMCB	6.1.2-3	DSER-OI		Grover	Closed	Action W		
				Westinghouse should indicate in the SSAR whether the changes in the recommended practices result in a greater or lesser amount of predicted hydrogen production.					
				Closed - See status detail for open item 6.2.5.2-3. Westinghouse will recalculate the design basis hydrogen generation using the source term specified in RG 1.7.					

Total Records: 3

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/27/97

Selection: [nrc st code]='Action W' And [DSER Section] like '7*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1044	NRR/HICB	7.2.8-2	DSER-OI		ITAAC/Deutsch, K.	Closed	Action W	NTD-NRC-95-4464	

Westinghouse should provide information concerning environmental qualification of PMS components addressing local temperature rises above the room ambient experienced by the components during operation.
It is desirable to have additional margin built into the design. The components should, therefore, be qualified by testing to higher temperatures than specified in the SSAR for a given room environment. Westinghouse should address this concern in the SSAR. Westinghouse should also provide mild environment equipment qualification in the CDM with the corresponding ITAAC.

Closed - Technical information agreeded to by NRC during meeting on May 15-16. Additional technical information regarding the equipment design margin to loss of HVAC has been incorporated into Revision 3 of the SSAR, Subsection 7.1.4.1.8. rkn 12/2
Westinghouse needs to decide approach to close this item. rkn 12/6

Action N - NRC still has the action to evaluate the Westinghouse proposal on procedural fix of instrument overheating after 24 hour period. (6/21 meeting with W/SPLB/HICB). Based on 11/21 W/NRC telecon, this approach is reasonable, see qualification program in SSAR Section 3.11.

Action W - NRC requested W provide proposed COL item for qualification margin and instrument setpoint data or document in the CDM and corresponding ITAAC (W is considering options; did not commit to either approach). rkn 12/2

Westinghouse does not consider there to be an applicable COL action to identify. Techaical information related to design margin against a loss of HVAC was provided in SSAR 7.1.4.1.6 and is considered technically resolved, as waws previously agreed to by NRC. This item is considered closed since there is no Westinghouse action required at this time to address this item (since the NRC relates this comment to the PMS ITAAC, the responsible engineer is changed to ITAAC). rkn 1/14/97.

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3965	NRR/HICB	7	TEL-OI		ERG	Closed	Action W	NTD-NRC-97-4936	

Respond to NRC Letter from Huffman to Liparulo of 8/19/96 which provided comments on PAM and the related ERGs.

Comments have been incoporated such that this letter (identifying changes to PAMS table in SSAR 7.5 and to the ERGs) should be out by 12/6. rkn 12/3
Letter is pending DCP approval, possibly 12/9; status would then be resolved although W owes SSAR changes and ERGs by end-Dec. rkn 12/6
Closed with completion of ERGs on Jan 13 NSD-NRC-97-4936 and SSAR Rev 10 incorporation of changes to the PAMS table. rkn 1/14/97

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/27/97

Selection: [nrc st code]='Action W' And [DSER Section] like '8*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1061	NRR/EELB	8.2.3.5-2	DSER-OI		TECHSPEC/Winters	Closed	Action W		
				(Offsite ac interface requirement) Westinghouse included in the AP600 standard design an interface identified as the "minimum number of ESF trains to be energized simultaneously." This interface appears to convey some safety significance, and the staff requires further clarification.					
				Closed: SSAR Table 1.8-1 revised to delete interface requirement because there is no requirement for ESF trains to be energized simultaneously on the AP600. Closed - With issuance of the Tech Specs in SSAR Rev. 9.					
1072	NRR/EELB	8.3.2.1-6	DSER-OI		Hayes	Closed	Action W	NTD-NRC-95-4464	
				(Minimun separation) Westinghouse should justify establishing a minimum separation distance of 1 inch between non-Class 1E conduit and Class 1E open top cable trays, since this separation is not in accordance with IEEE 384-1981.					
				Closed - SSAR Section 8.3.2.4.2 (Rev. 6) and Appendix 1A (Rev. 7) revised to address these exceptions. Letter NSD-NRC-96-4669 provided justification including test results industry may have to support partial exemption.					
1077	NRR/EELB	8.4.4-1	DSER-OI		TECHSPEC/Deutsch	Closed	Action W		
				(BTP ICSB 18, Item B4 Power lockout to MOV)Westinghouse should address the aspect of redundant indication be powered from different sources.					
				Closed: SSAR Chapter 7 (Section 7.5.4) revised to identify MOVs with redundant indication power requirements. Closed - With issuance of the Tech Specs in SSAR Rev. 9.					
1078	NRR/EELB	8.5-1	DSER-OI		Hayes	Closed	Action W		
				(Supplemental informations provided in RAI responses)Westinghouse should include in the SSAR supplemental information involving additional description of the design.					
				Closed - Information included in Rev. 6, subsection 8.3.1.1.2.1					
3394	NRR/EELB	8	RAI-OI		Hayes	Closed	Action W	NSD-NRC-96-4801	
				435.84 a. Westinghouse should provide the basis or justification for making emergency lighting non-Class 1E. b. Westinghouse should address the concern that during a seismic event emergency lighting would not be available. c. Westinghouse RAI response to 435.34 stated that the Class 1E dc distribution system design was in compliance with IEEE Standard 384 and NRC Regulatory Guide 1.75. There are no non-safety related loads fed from the Class 1E dc sytem. Changing the emergency lighting to non-Class 1E invalidates this response since the power supply is still Class 1E. The staff requests that this RAI be updated to reflect the current design status and include an evaluation of the potential of this design to degrade Class 1E power supplies.					
				Closed - Response provided via Westinghouse letter NSD-NRC-96-4801, August 14, 1996.					

Total Records: 5

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/27/97

Selection: [nrc st code]='Action W' And [DSER Section] like '17*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
37	nrr/hqmb	17.	RAI-OI		Fanto	Closed	Action W	NSD-NRC-96-4696	
				Question 260.34 (Quality Assurance) Westinghouse states that non-safety-related systems identified during implementation of the process used to determine the regulatory treatment of non-safety-related systems (RTNSS) are classified as AP600 Class D. The staff considers that the QA program applied to RTNSS-identified structures, systems, and componenets (SSCs) should follow guidelines comparable to those of Generic Letter 85-06 regarding anticipated transients without scram, and Regulatory Position 3.5 and Appendix A of Regulatory Guide 1.155, "Station Blackout;" for blackout non-safety-related equipment. Describe how the AP600 quality assurance program that is applied to RTNSS-identified SSCs is comparable to these guidelines.					
				Closed - Response provided by NSD-NRC-96-4696.					
1300	nrr/hqmb	17.1.3-1	DSER-OI		RTNSS/Kloes	Closed	Action W		
				The QA applied to RTNSS should be comparable to that described in Generic Letter 85-06 for ATWS, as well as Regulatory Position 3.5 and Appendix A of RG 1.155 for non-safety-related station blackout equipment.					
				Closed - Quality Assurance requirements are graded based on the safety classification of the item or service, as described in WCAP-8370. Appropriate industry quality assurance requirements will be applied to non-safety-related items and services. See item no. 37.					
1301	nrr/hqmb	17.1.3-2	DSER-OI		Kloes	Closed	Action W	NTD-NRC-95-4464	
				Westinghouse should add COL Action Item 17.1.3-1 to the SSAR. (When completing the detailed design during the COL design phase, a COL applicant will be responsible to submit its design phase QA program for staff review. This will be in addition to the staff review of the COL applicant's QA program for construction of the facility.)					
				Closed - Included in SSAR Revision 3.					

 Total Records: 3

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	January 27, 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:			

Cover + Pages 1 + 0

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 is to cover a complete response to item 5.f of your 10/17/96 letter. Item 5.f is a subset of open item 293 (Item 7.f(1) of your 10/17/96 letter). Open Item 293 was answered by NSD-NRC-97-4932 of January 7, 1997.

Jim Winters

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 27, 1997	NAME:	JIM WINTERS
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 1

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,

HERE IS THE MARKUP TO RESOLVE ITEM 244 PER OUR 11/5/96 TELECON.

IT WILL BE IN SSAR REVISION 11 UNLESS WE HEAR FROM YOU.

I CHANGED THE OITS FOR 244 IN ACCORDANCE WITH 11/5/96 INSTRUCTIONS.

cc: LINDGREN
McINTYRE
CUMMINS
RON VIJUK
WINTERS
HUTCHINGS
PIPLICA
JEANNE EVANS.

Jim Winters

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. Periodic checks on the high-pressure air compressor are made on a regular basis to verify that the breathing air meets the Quality Verification Level E as indicated in the ANSI/CGA G-7.1 standard.

sudden

During the initial plant testing prior to reactor startup, safety systems utilizing instrument air are tested as part of the safety system test to verify fail-safe operation of air-operated valves upon loss of instrument air or reduction of air pressure as described in Regulatory Guide 1.68.3. Section 1.9 summarizes conformance with Regulatory Guide 1.68.

gradual

9.3.1.5 Instrumentation Applications

An instrumentation package is included with each of the instrument and service air compressors. Each package consists of temperature and pressure transducers, indicators, and automatic protection devices. The temperature and pressure transducers support the automatic control modes of compressor operation. A manual mode of operation is also provided for each control system. Compressed air system indication and control are available in the main control room.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 28, 1997	NAME:	J. W. 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:			

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

COMMENTS:

DIANE

HERE ARE THE "ACTIVE" OPEN ITEMS WE DISCUSSED THIS MORNING. FOR NOW, AS AGREED, WE WILL CHANGE THEM ALL TO "ACTION W."

Jim Winters

cc: MANIKOWSKI
MCINTYRE
WINTERS

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/24/97

Selection: [nrc st code]='Active' And [type] like 'DSER-OI' Sorted by NRC Status

Item No	Branch	DSER Section/ Question	Type	Title/Description / Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
972	NRR/SCSB	6.2.1-1	DSER-OI		Narula/Grover	Closed	Active		
				The new method for calculating the source term for the AP600 is currently under staff review.					
				Closed: The source term originally proposed in the SSAR is being abandoned in favor of the source term defined in NUREG-1465 (the historical source term as defined in TID-14844 and RG 1.4 has been superseded by the source term described in NUREG-1465).					
973	NRR/SCSB	6.2.1-2	DSER-OI		Gresham/Grover	Action N	Active		
				Because the AP600 design does not have containment sprays, natural deposition on surfaces in containment is far more important than in past designs. The elimination of containment sprays from the design requires further staff review.					
				Active: Communication ongoing between Westinghouse and NRC relative to the concept of aerosol removal capability that could be used for severe accident mitigation.					
1009	NRR/SCSB	6.2.5.2-4	DSER-OI		Narula/McDermott	Closed	Active		
				The staff is currently evaluating the HRS to determine if the design conforms to the regulations and standards in Section 6.2.5 of the SRP.					
				Closed - Adoption of PARs closes this issue. SSAR revision to section 6.2.4 provides documentation of switch to PARs.					
1101	NRR/EMCB	9.3.6-2	DSER-OI		Winters	Closed	Active		
				Westinghouse must satisfactorily address the issue of the regulatory treatment of non-safety-related systems for the chemical and volume control system, as well as other non- safety-related systems.					
				Closed. The AP600 chemical and voleme control system has no RTNSS importance functions.					
1102	NRR/EMCB	9.3.6-3	DSER-OI		Winters	Closed	Active		
				Westinghouse should address criteria identified in Section 9.3.4 of the SRP concerning the CVS.					
				Closed. The AP600 chemical and volume control system is not a safety related system. Therefore, the criteria listed in Section 9.3.4 of the SRP is not applicable since they address only safety related systems.					
1458	NRR/SCSB	19.2.2.1-4	DSER-OI		Winters	Closed	Active		
				Westinghouse should address the resolution of fire protection concerns.					
				Closed - SSAR subsection 9.5.1 and Appendix 9A, Revision 8, provide detail of the fire protection system and resolve known NRC concerns. There are no outstanding NRC requests for additional information on fire protection.					
1461	NRR/SCSB	19.2.3.3-2	DSER-OI		PRA-2/Scobel/FAI	Action W	Active		
				Westinghouse should address the resolution of hydrogen generation and control concerns.					
				Active - Meeting with NRC on April 10 & 11, 1995 to discuss hydrogen issues. Also discussed follow-on questions 480.116 through 480.136.					

NRC STATUS = "ACTIVE"

TYPE = DSER-OI

AP600 Open Item Tracking System Database: Executive Summary **Date: 1/24/97**

Selection: [nrc st code]='Active' And [type] like 'DSER-OI50' Sorted by NRC Status

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1999	NRR/PDST	1.1	DSER-OI50		Butler	Closed	Active		
				3. Proprietary Information The AP600 Design Control Document cannot contain any proprietary information since it becomes part of 10CFR52. The present AP600 SSAR contains 3 proprietary volumes. Revision 0 of the PRA contained one proprietary volume out of 4.					
				Closed - Proprietary material in SSAR & PRA have been significantly reduced.					
2018	NRR/PDST	6	DSER-OI50		Fanto	Closed	Active		
				22. Post 72-hour support actions (See item 1, RTNSS)					
				Closed - This issue is being followed under DSER open item 6.3.4-1					
2019	NRR/SCSB	6.3	DSER-OI50		Schulz	Closed	Active		
				23. Debris in IRWST and Containment Sumps The staff is concerned that the strainers in the IRWST and containment sumps could be clogged by debris. Important factors are the use of non-safety-related coatings in the AP600, and possible sensitivity in this design to screen clogging because of dependence on gravity-driven flows. (See DSER Open Item 6.2.1.8-1)					
				Closed - The IRWST sump and the containment recirculation sump of the AP600 are designed and located to have small potential for plugging. The two types of sumps are in different areas and have separate flooding conditions. IRWST Sump The IRWST sump is at the bottom of the IRWST tank and is separated from the remainder of containment. The IRWST tank is fully enclosed (except for vents and condensate collection pipes) and is lined with stainless steel. The water has a high cleanliness as it is filtered and demineralized (by the spent fuel pit cooling system) during and after each refueling. Sludge will be minimal and the COL cleanliness program will prevent foreign debris from being introduced into the tank. During a LOCA vented RCS steam will condense on the containment shell and be directed by gutters to 4 inch pipes which drain into the IRWST. Containment paint or other lose debris will have to be smaller than 4 inches to be drained into the tank. Since the tank is normally full, floating debris will stay on the surface. Containment paint has a high specific gravity and will quickly sink to the bottom of the tank. Curbs in the gutters and inside the IRWST will trap the heavier debris preventing migration to the screens. With the low injection flows and the long tank drain down times (>6 hours), no significant transport of heavy debris is expected. When the tank reaches its minimum level during recirculation the water leve is above the top of the screens and floating debris can not be trapped on the screens. Containment recirculation sump The intakes for containment recirculation are located on the walls above the floor elevation at 83 feet. This is 11.5 feet above the waste sump below the reactor vessel (at elevation 71.6 feet). The bottom of the inlet screen is one foot off the floor, this provides a curb function. During a LOCA, water will flood the vessel cavity and adjacent floors up to the 107 foot elevation. The containment recirculation line is not opened until the water level in the IRWST reaches a low level setpoint. Water level in the flooded containment when IRWST reaches the setup is above the top of the recirculation inlet screens. Thus during the long floodup time (>6 hours) there is no transport of floating debris to the screens and heavy materials will have settled to the waste sump level or the 83 foot level. During recirculation the water level in containment will not change significantly nor will it drop below the screens. Thus the recirculation screens will be not be clogged by floating debris or by heavy debris. The loss of non safety coating will not have an effect on sump operation. In addition the COL will implement a cleanliness program to eliminate other potential clogging debris.					

NRC
STATUS
DSER-OI50
"ACTIVE" STATUS

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2024	NRR/HICB	16	DSER-OI50		Deutsch	Closed	Active		
				28. Design of the Diverse Actuation System The DAS has been identified by Westinghouse to be a RTNSS-important system for ATWS considerations. The staff needs additional information regarding the design and reliability (See DSER Open Item 7.7.2-1)					
				Closed - SSAR Chapter 7 revised to address. Per 11/21 telecon, NRC has action to discuss lack of DAS/Tech Spec relationship within the staff. rkn 12/2.					
2040	NRR/TSB	16.	DSER-OI50		TECHSPEC/RTNSS	Closed	Active		
				44. Technical Specifications for RTNSS-Identified Items (See item 1, RTNSS)					
				same as 2454 rkn 3/26 Closed - With issuance of the Tech Specs in SSAR Rev. 9.					
2045	NRR/SCSB	19.	DSER-OI50		Corletti	Closed	Active	NTD-NRC-95-4506	
				49. Intersystem Loss-of-Coolant Accident (ISLOCA) SECY-90-016 specified recommendations that plants should consider protection against the possibility of an ISLOCA. SECY-93-087 requires that passive plants adhere to the recommendations of SECY-90-016. The NRC has stated in the DSER that the staff is still evaluating the adequacy of the AP600 relative to these criteria.					
				Discussed at 2/9/95 SMM) Will be discussed during severe accident meetings. 5/2/95 Status: Not discussed yet. Closed - Response provided NTD-NRC-95-4506.					

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2051	NRR/SRXB	19.	DSER-OI50		Haag/Ohkawa	Closed	Active		

55. Passive System Reliability
Per DSER OI 19.1.3.1-3, the Staff requests detailed documentation on the success criteria assumed for various systems and operator actions modeled in the event tree top events.

At 7/27/95 meeting between NRC and W, Westinghouse presented how we were proceeding on the T/H uncertainty evaluation. NRC's interpretation of the scope of what T/H uncertainty evaluation should be was expanded. NRC Action to provide to Westinghouse again what the mission of this evaluation is to be.
At 6/29/95 SMM, W/NRC appear to be diverging on the purpose of the evaluation. Actions were provided for both W and NRC.
At 4/20/95 meeting between W/NRC, agreed to the purpose of the T/H uncertainty evaluation and the steps involved.

Discussed at 2/9/95 SMM)
Staff will develop focussed issues that will define path to resolution.

5/2/95 Status: Discussed at April 1995 SMM. Issue discussed during March 30 & April 20, 1995 mtg. W & staff agree on a resolution path for this issue (margins approach).

(Discussed at 4/4/95 SMM)
55. Passive System Reliability
Westinghouse will be benchmarking MAAP4 against NOTRUMP to demonstrate the adequacy of using it for addressing this issue. The staff will also benchmark MAAP4 against RELAP to identify any problems with using this code.
The staff recommended that Westinghouse identify preferred analysis cases of prolonged, maximum core uncovery events, and evaluate these cases (with sensitivity runs) first to provide some confidence and feedback in the areas of concern in the short term.

Action W Westinghouse will provide the NRC with the list of preferred analysis cases.
(5/2/95 Status) Action W - Analysis cases to be provided as part of MAAP4 T/H uncertainty benchmarking process.

Action N The staff will provide details of a calculation using the RELAP code that indicates that MAAP4 is nonconservative.
(5/2/95 Status) Action N - Still under staff review.

CLOSED - This item number is being closed because it is a duplicate of another entry in the database and is covered by DSER OI 19.1.3.1-3.

AP600 Open Item Tracking System Database: Executive Summary   Date: 1/24/97

Selection: [nrc st code]='Active' And [type] like 'MTG-OI' Sorted by NRC Status

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
457	NRR/ECGB	3.8	MTG-OI		Orr/Lindgren	Closed	Active	NTD-NRC-96-4617	1/4/96
				CONTAINMENT MEETING (3/28&29/94) ACTION ITEM 1 - Review Proprietary classification of containment vessel drawings and provide results to staff					
				Closed - The revision draft of SSAR section 3.8.2 transmitted by NTD-NRC-96-4617 provided nonproprietary figures for the containment.					
458	NRR/ECGB	6.2	MTG-OI		McDermott, D.	Action W	Active		
				CONTAINMENT MEETING (3/28&29/94) ACTION ITEM 2 - Provide additional details on containment design transients associated with passive containment function.					
				Action W - Include transients in SSAR section 6.2.1					
2442	NRR/HICB	16.1	MTG-OI		TECHSPEC/Birsa	Closed	Active		
				Evaluate the potential for failures that could defeat the capability for placing functions into bypass. This needs to be considered in the development of the actions (operator could be required to take an action to put the channel in bypass, and be unable to perform the action). At very minimum, the bases should clearly explain what is meant by placing the channel in bypass. Is taking the action (switch operation) without the system succeeding ok? This action is there to go from 1/3 logic to 2/3 logic (which affords operating fault tolerances). Staying in the 1/3 condition is not unacceptable. This concern is valid for both the RPT and ESF.					
				Closed - With issuance of the Tech Specs in SSAR Rev. 9.					
2457	NRR/HICB	16.1	MTG-OI		TECHSPEC/Birsa	Closed	Active		
				For PAMS, standard specs were determined using all type A, and category 1 variables. Consider use of action J (no LCO 3.0.3) producing a special report. Need to consider how this applies to AP600 PAMS and AP600 RG 1.97 categorization.					
				Closed - With issuance of the Tech Specs in SSAR Rev. 9.					

NRC STATUS= "ACTIVE"

TYPE= MTG-OI

FAX to TED QUAY

January 28, 1997

CC: Sharon, please make copies for: Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre

As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
4 (RAI 410.262)	9/10/96	1/9/97
21 (RAI 471.24)	5/20/96	1/9/97
30 (RAI 952.99)	5/13/96 12/17/96 - repeat	1/13/97
37 (RAI 260.74)	4/22/96	1/14/97
123 (M3.6.1-2)	4/30/96	1/15/97
134 (M3.11-1)	2/29/96 6/19/96 - more	1/16/97
135 (M3.11-2)	2/29/96 6/19/96 - more	1/17/97
137 (M3.11-4)	6/19/96	1/23/97
138 (M3.11-5)	2/29/96	1/24/97
139 (M3.11-6)	2/29/96 6/19/96 - more	1/24/97
140 (M3.11-7)	2/29/96	1/23/97
141 (M3.11-8)	2/29/96	1/24/97
586	6/19/96	1/28/97

Thanks

Jim

FAX to DIANE JACKSON

January 28, 1997

CC: Sharon, please make copies for: Ted Quay
Tom Kenyon

Don Lindgren
Don Hutchings
Brian McIntyre

OPEN ITEM #586

In my quest to make sure we have provided NRC with everything you need to prepare an FSER, I am researching open items for sections scheduled to complete FSER drafts. Attached are copies of some of the relevant documentation related to Open Item #586. This is the only open item associated with section 3.5.1. with an "NRC Status" of "Action W". We (NRC and W) promised each other in late 1996 that section 3.5.1.3 would be ready for a final FSER draft in December of 1996. See the draft Activity Plan sheet attached which includes all NRC comments. We provided a revision to SSAR subsection 10.2.6 before the current revision of June 19, 1996 (over 6 months ago). This was a very straightforward change and implemented NRC requests into the SSAR as we believed it was agreed. It is outside the scope of Design Certification for the SSAR to explicitly impose a time table for submittal of a turbine maintenance program. This will be part of the COL application process as required by subsection 10.2.6. It seems a reasonable request that NRC acknowledge receipt of the change. Our records show no outstanding Westinghouse action on this item (#586) or any other item associated with subsection 3.5.1.3. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Resolved". I will be calling about 1:30 today to see if you have additional information on this item. Thank you.

Jim

Jim Winters
412-374-5290

1 of 4

Selection: [item no] between 586 And 586 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
586	NRR/EMCB	3.5.1.3-13	DSER-OI	(TURBINE MAINTENANCE PROGRAM & MISSILE PROBABILITY CALCULATION) Westinghouse should add COL Action Item 3.5.1.3-5 to the SSAR. Provide turbine maintenence program including probability calculations of turbine missile generation. Closed - Addressed by COL item in SSAR section 10.2.6 Action W - NRC wants time table for submittal of turbine mainteance program in SSAR.	Lindgren	Closed	Action W		

AP600 DESIGN CERTIFICATION
(DRAFT ACTIVITY PLAN)

ID	Unique ID	Start	Finish	Task
276	1577	4/30/96	3/31/97	**(Ch. 3.12) Piping Criteria**
277	3433	4/30/96	4/30/96	SSAR Revision (Section 3.7.3)
278	3465	12/5/96	12/5/96	NRC/W Meeting
279	3437	1/31/97	1/31/97	W/NRC Piping/LBB Meeting
280	3340	2/28/97	2/28/97	SSAR Revision to formalize agreed-to changes (if necessary)
281	3042	4/30/96	6/24/96	Staff Review
284	3046	6/25/96	1/31/97	Comment Resolution
285	3047	3/31/97	3/31/97	
286	3048	3/31/97	3/31/97	FSER Input Complete (EMEB)
287				
288				
289	1584	2/29/96	12/31/96	**Materials & Chemical Eng.**
290				
291	1586	2/29/96	12/31/96	**(Ch. 3.5.1.3) Turbine Missiles**
292	2435	2/29/96	2/29/96	SSAR Revision (Section 3.5)
293	3038	2/29/96	4/24/96	Staff Review
296	3053	4/25/96	6/18/96	Comment Resolution
297	3054	12/31/96	12/31/96	
298	3055	12/31/96	12/31/96	FSER Input Complete (EMCB)
299				

Timeline: 1994 – 1998 (J F M A M J J A S O N D)

Lead: Lindgren/Winters
NRC: EMEB

Lead: Lindgren/Winters
NRC: EMCB

3/4

Project: AP600 Design Certification
Date: 12/30/96

W Task — W Milestone — Completed Milestone
NRC Task — NRC Milestone — Summary

- Hydrogen gas pressure
- Hydrogen gas purity
- Generator winding overtemperature
- Generator ampere, voltage, and power

Additional generator protective devices are listed in Table 10.2-3.

10.2.6 Combined License Information on Turbine Maintenance and Inspection

The Combined License holder will implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis.

10.2.7 References

1. WSTG-4-P, Proprietary and WSTG-4-NP, Nonproprietary, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines," October 1984.

2. WCAP-11525, Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency, 1987.

3. NUREG-1275, Vol. 11, Operating Experience Feedback Report - Turbine-Generator Overspeed Protection Systems, Commercial Power Reactors, H. L. Ornstein, Nuclear Regulatory Commission, April 1995.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	January 28, 1997	NAME:	Jim Winters
TO:	Diane Jackson	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 1

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,

Here is first page of our response for SECY-96-128. It says we concur with the staff position on "cold-vs-safe shutdown" since it was contained in Item V. Technical Specifications". This answers first bullet for 308 on our notes from 11/19 meeting, third bullet for 308 in your 1/3/97 letter.

Jim Winters

File - SECY - 96 - 128

Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

DCP/NRC0583
Docket No.: STN-52-003

August 20, 1996

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-001

ATTENTION: MR. JOHN HOYLE

SUBJECT: WESTINGHOUSE COMMENTS ON "POLICY AND KEY TECHNICAL ISSUES PERTAINING TO THE WESTINGHOUSE AP600 STANDARD PRESSURIZED REACTOR DESIGN", SECY-96-128

Dear Mr. Hoyle:

Westinghouse appreciates the opportunity, as the designer of the AP600 advanced passive nuclear power plant, to provide our perspective on SECY-96-128, Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standard Pressurized Reactor Design. Detailed comments on each of the 10 issues are presented in Attachment 1. These detailed comments include specific recommendations for the following items where the Westinghouse position differs from that proposed by the staff:

- I. Design Basis Radiological Consequences
- II. Prevention and Mitigation of Severe Accidents
- IV. Post-72 Hour Actions
- VIII. Spent Fuel Pool Cooling System

These comments and recommendations were discussed with the Advisory Committee on Reactor Safeguards on July 19, 1996 and August 8, 1996.

Please contact me if you have any questions concerning these comments.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

2896A

FAX to DIANE JACKSON

January 28, 1997

This is in response to item 7.i.(1) of your 10/17/96 letter, item 7.i.(1) of our 12/17/96 telecon and OITS item 304. HEPA filtration is not required on the VHS exhaust to the plant vent because the high radiation alarm in the exhaust line would lead the operators to terminate discharge on high radiation. This approach is consistent with the VRS and the two VAS exhaust paths to the plant vent shown on Figure 9.4.7-1 (Sheet 1 of 2). We recommend that the "NRC Status" for item 304 be changed to "Action W".

Thanks
Jim Winters
412-374-5290

cc: McIntyre
Cummins
Hutchings

FAX to JOE SEBROSKY

January 28, 1997

To: Joe Sebrosky (NRC)

SUBJECT: AP600 PRA PAGE MARKUPS

Attachment 1 contains markup page changes to the AP600 PRA. Specifically, there are changes to PRA page 6-24 and page 45-7. These changes will be included in the next revision to the PRA. The staff reviewers should include these markup pages with their PRA report.

On AP600 PRA page 45-7 (see Attachment 1), reference 45-3 has been updated. Attachment 2 contains a copy of the referenced paper.

Please contact me if you have any questions concerning this fax material.

Cynthia L. Haag
Advanced Plant Safety and Licensing

ATTACHMENT 1

CICA (CIC) CONTAINMENT ISOLATION FOR LARGE LOCA

Large LOCA: LLOCA

The success criterion for this case is the same as specified for case CIC (containment isolation following core damage). However, since containment isolation must occur in this case in time to prevent core damage, there is less time available for operator action to manually actuate containment isolation if automatic actuation fails.

(Note that CIC is used in the determination of XCICPO for event tree top event CIS. XCICPO = CIC + PO, where ~~CIC and~~ PO ~~are as~~ is defined in Chapter ~~[TBD]~~ 31.)

Basis for Success Criteria:

The requirements for CIC for large LOCA prior to core damage are generally as specified for CIC (Chapter [TBD]) following core damage. However, for large LOCA prior to core damage, there is less time for operator action to actuate containment isolation if automatic isolation fails than there would be for the post-core-damage action. Although the amount of time available for this backup operator action is dependent on break size and other parameters, this simple action is specified within the first several steps of emergency response guideline AE-0 (Reactor Trip or Safety Injection Response). Hence, it is expected to succeed, especially for this smaller category of large breaks.

6.4.9 Core Makeup Tank System

This top event represents injection of borated water into the reactor coolant system by the core makeup tank system. For events with loss of RCS water inventory, the core makeup tanks system is automatically actuated on an S-signal following low pressurizer pressure. For large LOCA, the analysis assumes that CMT injection does not occur fast enough to prevent core uncovery, and no credit is taken for CMTs. For medium, intermediate, and small LOCA, the CMTs provide RCS inventory makeup and core cooling until normal RHR or gravity injection can be established.

During transients, the core makeup tanks system may be automatically actuated if the event causes an RCS cooldown (e.g., steamline break), resulting in RCS coolant shrinkage and low level. Event sequences in these cases are the same as for small LOCA, once CMTs inject. For heatup transients, if secondary side cooling and passive RHR are unavailable, primary side relief may occur through the pressurizer safety valves; in this case, CMT injection is needed in order to allow short term RCS cooling (feed and bleed cooling) and keep the core covered until depressurization to normal RHR operating pressure occurs.

Due to pressure distribution in the reactor coolant system, core makeup tanks system injection is assumed to be possible for events other than large or medium LOCA only if all four reactor coolant pumps are tripped. The protection and safety monitoring system (PMS)

category, the releases are very similar over the first 24 hours after core uncovery. This is because the containment remains intact. The release at 24 hours after core damage for each of these sequences is based upon design-basis containment leakage area. The differences in the releases among release categories occur over the long term as the containment pressurizes.

The BP, CI, CFE, and CFI release categories represent large releases that occur prior to 24 hours after the onset of core damage. Release category CFL represents large releases that occur after 24 hours.

A fission-product release source term is also developed to address the sensitivity assuming that the IC source term from the containment is released directly to the environment with no holdup or decontamination in the auxiliary building. The sensitivity release fractions also represent source terms for consequence analysis, as discussed in Chapter 49.

Two sensitivity analyses were completed to determine the impact on MAAP model parameters PRAT and FAERDC. These variables control the fission-product release correlation and the airborne aerosol mass, respectively. The results of the sensitivity analyses demonstrate that the default value yields conservative results.

Finally, a comparison of the AP600 source term results with NUREG-1150 shows reasonable agreement between the AP600 source terms for release categories CFE and CFI and the closest NUREG-1150 equivalent.

45.7 References

45-1 EPRI MAAP 4.0 Users Manual.

45-2 EPRI Letter to James Wilson, USNRC, dated April 30, 1993, attachment titled, "Passive ALWR Secondary Building Mixing and Leak Rate Monitoring."

45-3 ~~March 28, 1996 letter from David E. W. Leaver (Polestar Applied Technology) to Brian McIntyre (Westinghouse), attachment of draft paper titled, "Aerosol Retention During a Steam Generator Tube Rupture Severe Accident Event".~~

45-4 NUREG-0772, Technical Bases for Estimating Fission Product Behavior During LWR Accidents, 1981.

45-5 NUREG/CR-4551, Volume 2, Revision 1, Part 4, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Experts' Determination of Source Term Issues," 1992.

Jun Li, David Leaver, James Metcalf, "Aerosol Retention During An Unisolated Steam Generator Tube Rupture Severe Accident Event," to be presented at the Fifth International Topical meeting on Nuclear Thermal Hydraulics Operations & Safety, Beijing, China, April 13-16, 1997.

ATTACHMENT 2

Aerosol Retention During An Unisolated Steam Generator Tube Rupture Severe Accident Event

To be presented at

the Fifth International Topical meeting on Nuclear Ternal Hydraulics Operations & Safety

Beijing, China

April 13-16, 1997

by

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AEROSOL RETENTION DURING AN UNISOLATED STEAM GENERATOR TUBE RUPTURE SEVERE ACCIDENT EVENT

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INTRODUCTION

A steam generator tube rupture (SGTR) initiated core damage sequence can potentially lead to containment failure. In such a sequence, fission products released from the damaged core to the primary system may be carried by the flow through the broken tube(s) into the secondary side of the steam generator and finally out to the reactor building via, for example, a stuck open safety relief valve.

The purpose of this work is to describe a model for estimating aerosol retention during a severe accident involving an unisolated SGTR. The AP600 plant is used as an example in developing the model. This work, under sponsorship of the U.S. Department of Energy Advanced Reactor Severe Accident Program (ARSAP), has been performed to support the Advanced Light Water Reactor (ALWR) Program position on emergency planning. Most PRAs and severe accident codes assume that a significant fraction of fission products flowing through an unisolated break in a steam generator tube escape to the environment. For example, in NUREG 1150 [1], the median estimate of the fraction of the core inventory of iodine released to the environment was 27%, and the 95th percentile estimate was 80%. This estimate was based on an expert elicitation panel. NUREG 1150 states that based on the work of the panel, "there is a very good chance that there would be little retention of radionuclides in the steam generator and the piping." [1] In the AP600 PRA [2], the iodine release from an unisolated SGTR is 22% based on MAAP results.

These results indicate that little credit has traditionally been given to aerosol removal mechanisms for unisolated SGTR sequences. The work reported here considers, and develops models for, applicable aerosol removal mechanisms for these sequences. Inertial impaction, turbulent deposition, and thermophoresis are important aerosol removal mechanisms for unisolated SGTR sequences. Consideration of these removal mechanisms will provide more realistic source term releases and more accurate risk estimates.

As the fission product aerosols travel through the steam generator, mechanisms exist to remove them from the flow, e.g., turbulent deposition, inertial impaction, thermophoresis, etc. The importance of each removal mechanism will vary depending on the location of the break. When the break is located high along the tube, a large fraction of the aerosols carried in the flow in the broken tube(s) will be removed via turbulent deposition on the inner surface of the tube(s) before they reach the break. When the break location is low (for example, at the joint between the tube and the tube sheet, see Figure 1), a significant amount of aerosol will escape turbulent deposition inside the tube and will pass through to the secondary side of the steam generator where it will be subject to removal processes (e.g., inertial impaction, turbulent deposition and thermophoresis) as it impinges on the surrounding tubes, support plates and wrappers, and as it travels in the tube bundle.

The analysis of the retention of aerosols in the event of an unisolated SGTR can be divided into two problems. One is the aerosol deposition on the primary side of the steam generator (i.e., inside the tube) and the other is the aerosol deposition on the secondary side of the steam generator. We shall analyze the two problems separately.

The geometry and arrangement of an AP600 steam generator are shown in Figure 1. The steam generator plenum is directly connected to the primary system and is divided into two sides separated by the partition plate. During normal operation, water from the core comes via the hot leg into one side of the plenum flowing through thousands of inverted U-tubes to reach the other side of the plenum and then back to the core via the cold leg. As the water travels through the tubes, it passes heat to the water on the secondary side of the steam generator and vaporizes it.

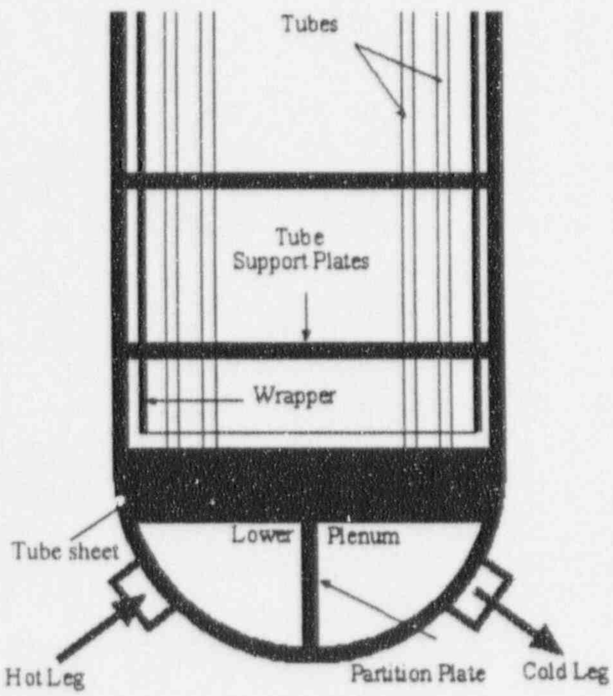


Figure 1. Schematic of the Steam Generator

FLOW VELOCITY IN BROKEN TUBES

To predict aerosol behavior, it is first necessary to estimate the gas flow velocity in the broken tubes for the core damage sequence, which requires the application of the theory of compressible fluid dynamics. According to the theory [3], the tube flow velocity under the ideal condition is determined by the pressure ratio of the primary side to the secondary side across the break, and the adiabatic exponent (k) that is the ratio of the heat capacity at constant pressure (c_p) to that at constant volume (c_v) (for steam, k is about 1.3). If subscripts 1 and 2 are denoted for the primary side and the secondary side of the steam generator, the expressions for the flow velocity and mass flow rate inside the broken tube are given by Equations (1) and (2):

$$v = \sqrt{\left(\frac{2P_1}{\rho_1}\right)\left(\frac{k}{k-1}\right)\left[1-\left(\frac{P_2}{P_1}\right)^{\frac{k-1}{k}}\right]} \tag{1}$$

$$\frac{\dot{m}}{A} = \sqrt{(2P_1\rho_1)\left(\frac{k}{k-1}\right)\left[\left(\frac{P_2}{P_1}\right)^{\frac{2}{k}}-\left(\frac{P_2}{P_1}\right)^{\frac{k+1}{k}}\right]} \tag{2}$$

Also according to the theory [3], the flow will be choked when the pressure ratio of the secondary side to the primary side is less than the critical pressure ratio. The critical pressure ratio is determined solely by the adiabatic exponent as follows:

$$\frac{P_2}{P_1} = \left(\frac{2}{k+1}\right)^{\frac{k}{k-1}} = 0.546 \qquad \text{for } k = 1.3$$

The expressions for the flow velocity and mass flow rate at the exit of the nozzle when the pressure ratio is less than or equal to the critical value are given by Equations (3) and (4).

$$v_n = \sqrt{\left(\frac{k}{k+1}\right)\left(\frac{2P_1}{\rho_1}\right)} \tag{3}$$

$$\frac{\dot{m}}{A} = \sqrt{P_1\rho_1 k\left(\frac{2}{k+1}\right)^{\frac{k+1}{k-1}}} \tag{4}$$

In these equations, ρ_1 is the gas density in the lower plenum of the steam generator, which can be derived based on the ideal gas law, i.e., $P/\rho = \Re T/\hat{M}$, where $\Re$ is the universal gas constant and $\hat{M}$ is the molecular weight of the gas in question.

AEROSOL AGGLOMERATION IN LOWER PLENUM

The average residence time for the particles in the lower plenum is expected to be significant as is the average residence time for the particles trav-

eling from the primary system to the lower plenum, due to limited gas flow (as controlled by the small cross-section area of a tube). As a result, it is expected that substantial agglomeration will occur among fission product aerosols. The fact that the plenum volume is small so that the aerosol concentration will be high will further enhance agglomeration.

Studies have been done on aerosol size distribution in the area of nuclear safety analysis. Powers and Burson [4], after examining the results from extensive Source Term Code Package calculations and from the QUEST study [5], concluded that the aerosol size distributions present in containment will be lognormal with the mass mean aerosol diameter between 1.5 to 5.5 μm and geometric standard deviation between 1.6 to 3.7. It is expected that fission product aerosols would have longer residence time for agglomeration in traveling from the primary system to steam generator tubes than in traveling from the primary system into the containment. Thus, the size of the aerosols entering the steam generator tube is expected to be larger than that entering the containment. For conservatism, a mass mean diameter of 3 μm and a geometric standard deviation of 1.81 are used here.

AEROSOL RETENTION INSIDE TUBES

In a broken tube, the flow velocity inside the tube is expected to be of the order of hundreds of meters per second, so the flow is expected to be turbulent. As a result, the turbulent deposition model [6] should be used for the aerosol in-tube retention calculation, i.e.,

$$E_D = 1 - \exp\left[-4\left(\frac{L}{D}\right)\sqrt{\frac{f}{2}}V_d^+\right] \tag{7}$$

where E_D is the retention efficiency of aerosol particle due to turbulent deposition (defined as the ratio of the particle mass deposited in the tube to the particle mass entering the tube), D is the diameter of the tube, L is the distance between the bottom of the tube sheet and the break, f is the Fanning friction factor ($=0.046/\mathrm{Re}^{0.2}$ [7] for turbulent flow in a circular tube with the Re in the range of 3×10^4 to 10^6). V_d^+ is the dimensionless deposition velocity of particles determined by the following equations:

$$V_d^+ = \begin{cases} 0.065Sc^{-2/3} & \tau_+ < 0.2 \\ 3.5\times10^{-4}\tau_+^2 & 0.2 < \tau_+ < 20 \\ 0.13 & \tau_+ > 20 \end{cases} \tag{8}$$

where Sc is the Schmidt number and τ_+ is the so-called dimensionless particle relaxation time. They are defined, respectively, as:

$$Sc = \frac{\mu}{\rho \wp} \tag{9}$$

$$\tau_+ = \frac{\rho\tau u_*^2}{\mu},\ \tau = \frac{\rho_p d_p^2 Cu}{18\mu},\ u_* = \sqrt{\frac{f}{2}}U \tag{10}$$

In Equations 9 and 10, ρ is the gas density, $\wp$ is the diffusion coefficient of the particle in gases and U is the flow velocity in the tube. In the expression for τ, ρ_p and d_p are, respectively, the density and diameter of the particle, μ is the viscosity of the gas and Cu is the Cunningham slip factor

Very often, the decontamination factor (DF) is used to represent the removal efficiency of aerosols in a given system. DF is defined by the ratio of the rate of aerosols leaving the system under consideration to the rate of aerosols entering the system. The relationship between DF and E_D is given by the following equation, according to their respective definition:

$$DF = \frac{1}{1-E_D} = \exp\left[4\left(\frac{L}{D}\right)\sqrt{\frac{f}{2}}V_d^+\right] \tag{11}$$

Evidently, the DF of aerosols inside the SG tubes increases exponentially with the flow distance along the tube.

AEROSOL RETENTION ON SECONDARY SIDE

The gas and aerosols coming out of the break in the tube will be in the form of a jet with an initial velocity in the range between hundreds of meters per second. The jet is expected to impinge on an array of tubes, or surfaces of plates and structures

in the near field of the break, causing significant removal of aerosols due to aerosol impaction and turbulent deposition. The jet will eventually slow down at some distance from the break due to both the entrainment of initially quiescent secondary side gas and drag from structures. By then, turbulent deposition and inertial impaction become unimportant as compared to thermophoresis. It is therefore logical to consider the aerosol retention on the secondary side as two sub-problems: the near field problem and the far field problem.

Near Field Aerosol Removal

The jet coming out of the break is likely to be horizontal since the tubes are vertical. A vertical jet is only possible when it is a guillotine break and the both sections of the broken tube are out of alignment.

When the jet is horizontal, aerosols in the jet will experience inertial impaction and cross flow turbulent deposition on the surrounding tube surfaces. The inertial impaction combined with turbulent deposition of aerosol on tubes have been studied both experimentally and theoretically by Douglas and Ilias [8]. If we define the collection efficiency of aerosols in the cross flow by a single tube as the ratio of the aerosol mass deposited on the tube per unit time to the aerosol mass that would flow across the projected area of the tube per unit time if the tube were not present, the efficiency is governed by the Stokes numbers as shown in Figure 2 (reproduced from Figure 9 on page 460 of Reference [8]).

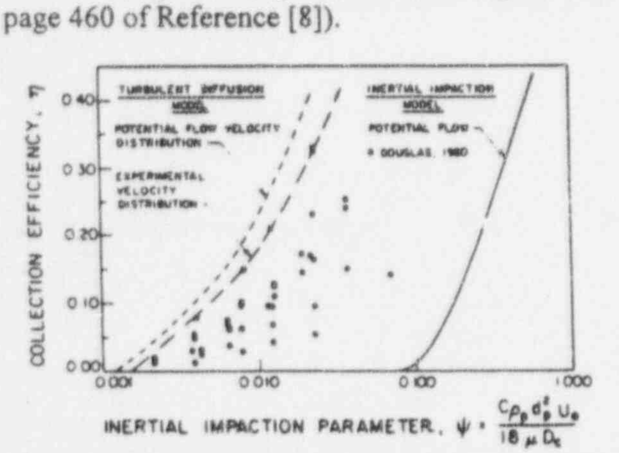


Figure 2. Collection efficiency of aerosol by cylindrical collectors in turbulent cross flow

If the jet shoots vertically, parallel to the tubes, the jet would be expected to travel in between confined channels which tend to be bounded by the tubes. The expansion of a circular jet into an infinite volume of quiescent gas is about 12° measured from the centerline of the jet. With the presence of tubes and the likelihood of circulation established near the break as a result of the blowdown, the jet expansion is expected to be limited. The jet velocity reduction in the near field is estimated according to momentum conservation to be about a factor of 2 to 3, which means that the gas velocity in flow channels is still in the range of hundreds of meters per second. Since the hydraulic diameter of each flow channel is almost the same as that of the tube based on the fact that the total flow area inside tubes is about the same as that outside the tubes for the AP600 steam generator (which is true in general for most steam generator designs), the aerosol removal in the channels will not be too different from that inside the tube. The channel flow will not travel very far before hitting a support plate, where it will then change direction and be perpendicular to the tubes, thus becoming similar to the cross flow situation described in the previous paragraph.

Thus, the methodology we need to quantify the aerosol removal in the near field is for horizontal jets. Aerosol deposition on an array of cylinders in turbulent cross flow is not a simple problem. This is true even if only one cylinder is involved. The correlation for calculation of aerosol inertial impaction on a single cylinder in cross flow has been proposed by many researchers. To consider turbulent deposition on a single cylinder, on the other hand, requires a numerical scheme and computer programming, as demonstrated in Reference [8], which is beyond the scope of this work at this time. Therefore, in this analysis, we have used the results from Reference [8] to estimate turbulent deposition removal of aerosols on the tubes in the near field.

The key parameter in determining the collection efficiency of aerosols by a tube in the near field cross flow situation is the jet velocity. The initial velocity of the jet is very high, but the jet is ex-

pected to be decelerated due to the transfer of its momentum to the initially quiescent secondary side gas and the form drag as the jet flows across the nearby tubes. The slowing down of the jet due to the entrainment of quiescent gas is expected to be unimportant in the near field because most of the space nearby is occupied by the tubes and the quiescent gas near the break is expected to be mobilized before the release of fission product aerosols. The drag force, however, is expected to slow down the jet significantly, and the jet will expand to satisfy the mass conservation. This expansion due to the slowing down of the jet from the drag force needs to be considered.

We now estimate the deceleration of the jet due to the form drag of a tube. If we draw a control volume around a tube, the jet flow rate coming into the control volume from the control surface (A_1) in front of the tube is $\rho A_1 U_1$ and the momentum flux is $\rho A_1 U_1^2$ where A_1 equals the projected area of the tube, ρ is the density of the gas and U_1 is the velocity of the jet before hitting the tube. The flow rate leaving the control surface (A_2) is $\rho A_2 U_2$ and the momentum flux is $\rho A_2 U_2^2$, where U_2 is the velocity of the jet after passing the tube and before hitting the next tube. The drag force in the control volume (from the tube) is represented by $\frac{1}{2} C_D \rho A_1 U_1^2$, where C_D is the drag coefficient. According to Reference [9], the drag coefficient in the flow region where the cylinder diameter-based Reynolds number is greater than 100 is about unity, and it is possibly less than unity when the Reynolds number is greater than 10^5. The Reynolds number for the cases of interest here is usually greater than 10^5. For conservatism, a drag coefficient of unity will be used. Now, we write the mass and the momentum balance equations:

$$\rho A_1 U_1 = \rho A_2 U_2 \tag{12}$$

$$\rho A_1 U_1^2 - \frac{1}{2} \rho A_1 U_1^2 = \rho A_2 U_2^2 \tag{13}$$

so

$$U_2 = \frac{1}{2} U_1 ,\ A_2 = 2 A_1 \tag{14}$$

Equation 14 indicates that a jet will lose about half of its velocity and will double its flow area as it flows across a tube (due to the form drag of the tube). If we assume that the jet will lose half of its remaining velocity as it flow across another tube, we obtain the following expression:

$$U_3 = \frac{1}{2} U_2 = \frac{1}{2^2} U_1 \tag{15}$$

After passing n tubes, the velocity of the jet in terms of its initial velocity is then expressed by

$$U_{n+1} = \frac{1}{(2)^n} U_1 \tag{16}$$

$n = 1,2,3,......$

Now we turn to the Stokes number for the aerosols entrained in the jet, and examine how the Stokes number will affect the rate of the aerosol deposition on tubes. The Stokes number is defined as follows:

$$Stk = \frac{\rho_p d_p^2 U Cu}{18 \mu D} \tag{17}$$

where ρ_p and d_p are, respectively the density and diameter of the particle, μ is the viscosity of the gas and Cu is the Cunningham slip factor, U is the jet velocity, and D is the outer diameter of the tube.

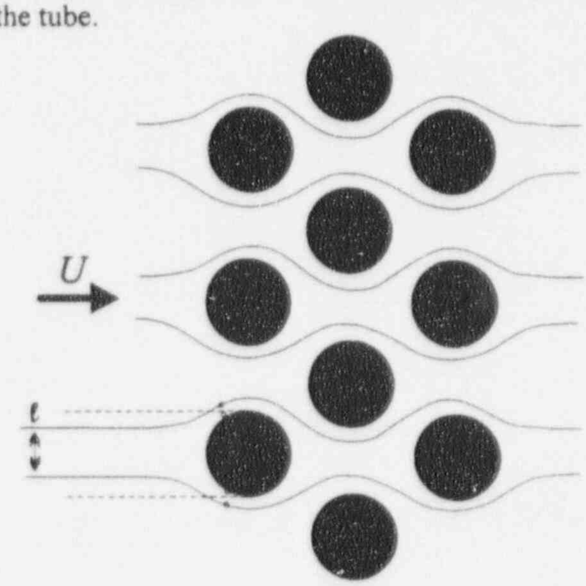


Figure 3. Schematic of the Flow Across a Bundle of Tubes

As pointed out previously, after the jet enters the secondary side of the steam generator, aerosol retention mechanisms exist due to the presence of

a large number of closely packed and staggered U-tubes as illustrated in Figure 3 (~6300 tubes per SG for AP600). Typically, the classical modeling of aerosol deposition on cylinders considers inertial impaction and interception only. This leads to an under-prediction of the aerosol collection efficiency for the high Reynolds number cross flow. This was discovered experimentally by Peter Douglas in his Ph.D. research, according to Douglas and Ilias [8] as they were developing a theoretical model for the deposition of aerosol particles on cylinders in turbulent cross flow in which eddy diffusion of aerosols was considered as the primary deposition mechanism. The model was found to agree very well with the experimental results. The collection efficiency of aerosol particles on a single cylinder was found to be 40% for Stokes number of 0.04. Although it was not shown in Reference [8] how the collection efficiency will vary as the Stokes number increases from 0.04, it is expected that the collection efficiency will increase. This is because the higher the Stokes number, the more important the inertial impaction (which will add to the overall aerosol deposition) and the larger the turbulent eddy diffusivity (measurements in Reference [8] showed that eddy diffusivity increased from 6 to 25 cm^2/s as the Reynolds number varied from 20,000 to 140,000), which will only increase the turbulent deposition. For conservatism, it is assumed that the collection efficiency of a single tube remains at 40% as the Stokes number increases from 0.04.

For example, if the initial Stokes number based on the mass mean diameter (at the break) is 4, it is then 100 times greater than 0.04 (the minimum Stokes number at which the tube retention efficiency is 40%). Thus, the jet velocity can be reduced by a factor of 100 (i.e., pass roughly 7 tubes according to Equation 16) before the Stokes number drops to approximately 0.04. In other words, the Stokes number of the jet flow can be assumed to be greater than or equal to 0.04 before crossing the first eight tubes in its path. Assuming that the collection efficiency on each tube of the eight is 40%, the overall efficiency of total eight tubes is then $1-0.6^8=0.983$ and, in terms of DF, is about 60.

Far Field Aerosol Removal

When the jet is slowed down to the point that inertial impaction and turbulent deposition are no longer effective for aerosol removal, aerosols are expected to travel in the steam generator at very low velocity. Due to self heating from the fission products, the gas is likely to be at higher temperature than the tube surface far from the break. Thus, heat transfer from the gas to the tube is expected. As a result, thermophoretic deposition of aerosols on the tube surface is likely to occur. The rate of thermophoretic deposition of aerosols in the far field is estimated below.

The far field flow condition can be modeled by a uniform gas flow through a large number of flow channels in between an array of tubes. Since the total flow area inside the tubes is about the same as that outside the tubes, the hydraulic diameter of the flow channels is about the same as the diameter of the tubes and total heat transfer area (A) is πDLN, where D is the hydraulic diameter or tube diameter, L is the channel length and N is twice the total number of U-tubes. Removal efficiency of aerosols in each flow channel is given by the following equation, i.e.,

$$E_T = 1 - \exp\left(-\frac{4v_{th}L}{Du}\right) \tag{18}$$

where v_{th} is the thermophoretic velocity given below and u is the uniform velocity that equals $2U/N$ for the case in which 2 tubes are ruptured. U is the initial jet velocity.

The widely used model for evaluating the thermophoretic velocity of particles driven by the temperature gradient was developed by Talbot et. al. [10], in which the thermophoretic velocity v_{th} is expressed as follows:

$$v_{th} = f(\alpha, Kn, T)\frac{dT}{dy} \tag{19.a}$$

$$f = \frac{2c_S Cu(\mu_g/\rho_g)[\alpha + c_T Kn]}{[1+2(\alpha + c_T Kn)][1+3c_M Kn]}\frac{1}{T} \tag{19.b}$$

where α is the ratio of the thermal conductivities of the gas and particle. Kn is the Knudsen number

(defined by the ratio of two times the gas molecular mean free path to the particle diameter) and Cu is the Cunningham slip correction factor, which is a function of the Knudsen number. μ_g is the gas viscosity and ρ_g is its density. c_S, c_T, c_M are three constants, for which Talbot et. al. give their best values as 1.17, 2.18 and 1.14, respectively.

As shown in Equation 19.a, thermophoretic velocity is represented by two parts, the temperature gradient (dT/dy) and the coefficient of dT/dy (f). f can be obtained when the thermal hydraulic conditions are known. The temperature gradient, on the other hand, can be estimated based on the heat transfer information.

If the total heat transfer rate (Q) in the secondary side of steam generator is known, the average temperature gradient can be approximated by the following expression:

$$\frac{dT}{dy} = \frac{Q}{k_g A} = \frac{Q}{k_g \pi DLN} \qquad (20)$$

where k_g is the gas thermal conductivity.

Combining Equations (18), (19) and (20), we can obtain the new expression for the far field removal efficiency of aerosols in the flow channel, i.e.,

$$E_T = 1 - \exp\left(-\frac{2fQ}{\pi D^2 U k_g}\right) \qquad (21)$$

SAMPLE CALCULATION

Thermal hydraulic conditions

The sample case is similar to the one from the Westinghouse AP600 PRA [2] in which breaks were assumed to occur in two tubes. Both the primary system and the secondary side of the steam generator (SG) are de-pressurized. The secondary side is assumed to be dry.

The thermal hydraulic conditions in both the primary and secondary sides of the SG for this case are given in Figures 4 and 5.

Aerosol retention inside tubes

Substituting these thermal hydraulic conditions into Equations 7, 8, 9 and 10, the retention efficiency of aerosol inside a tube can be evaluated for any given break location. The results in terms of DF are shown in Figure 6. The maximum and minimum are due to the variation of the thermal hydraulic conditions with time.

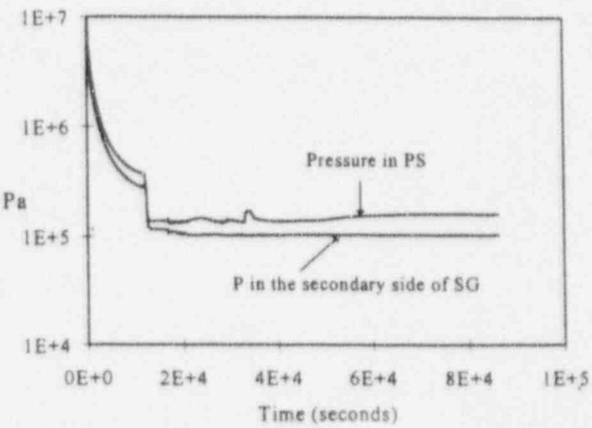


Figure 4. Pressures in the primary and secondary sides of the steam generator.

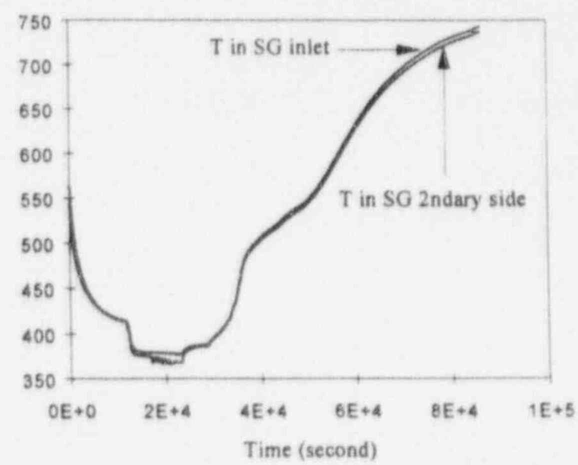


Figure 5. Temperatures in the primary and secondary sides of the steam generator.

Aerosol retention in near field

Assuming 3 μm mass mean diameter for the aerosols entering the secondary side of the steam gen-

erator, the initial Stokes number based on the mass mean diameter (at the break) is 4. Thus, the Stokes number will remain high enough before hitting the eighth tube in the flow path (based on Equation 16) such that the retention efficiency of aerosols by a single tube is 40% according the Figure 2. As the result, the overall efficiency of total eight tubes is $1\text{-}0.6^8$=0.983 and, in terms of DF, is about 60. Even for the 1.5 μm mass mean diameter, the minimum value from Reference [5], the same analysis shows that the overall aerosol retention efficiency in the near field is about 95% or a DF of 20. For the 5.5 μm mass mean diameter, on the other hand, the analysis shows that the overall aerosol retention efficiency in the near field is as high as 98.99% or a DF of 99.

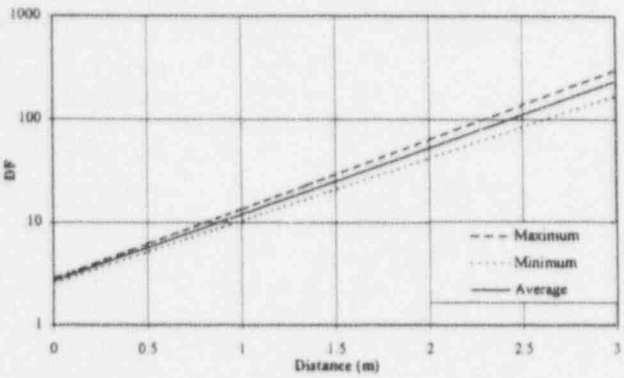


Figure 6. Aerosol decontamination factor in broken tube vs. distance from tube sheet

Aerosol retention in far field

For the AP600 case under consideration, the coefficient (f) as a function of time, parametric in the particle size (i.e., the Knudsen number) and α is found to be insensitive to α, but relatively sensitive to particle size, as shown in Figure 7. The particle size for evaluating the coefficient is assumed to be in the range of 0.2 to 20 μm . This range constitutes more than 99.9% of aerosol mass for the aerosol distribution of with mass mean diameter = 3 μm and geometric standard deviation = 1.81. It should be noted that the actual particle size distribution in this far field is expected to be smaller than in the near field since most of the large particles are likely to have been removed by the near field removal mechanisms. Also, for 20 μm particles, the sedimentation velocity is in the range of centimeters to tens of centimeters per second. Thus it is the 0.2 to 2 μm

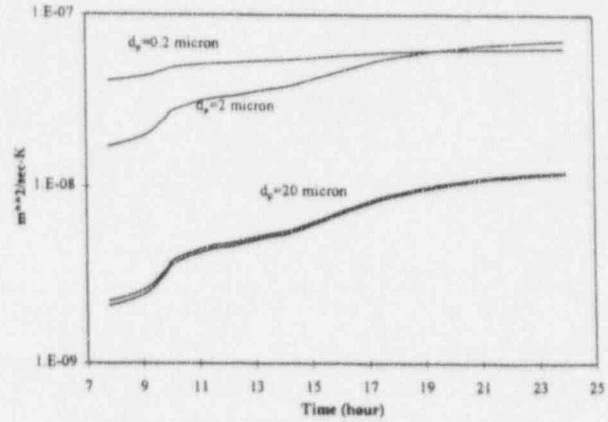


Figure 7. The coefficient of thermophoretic velocity as a function of time for AP600

range that is likely to be of most interest here.

Four values of the total heat transfer rate in the steam generator are assumed: 0.2 megawatts, 1 megawatt, 2 megawatts and 4 megawatts. These values represent roughly 1%, 5%, 10% and 20% of the decay power for AP600 (i.e., about 20 megawatts). For three particle sizes, i.e., 0.2 μm , 2 μm and 20 μm, the removal efficiencies are calculated using Equations 19 and 21. It is found that a minimum DF of 2 can easily be achieved for particles smaller than 2 μm even when the heat transfer rate in the steam generator is only about 1% of the decay power. As mentioned earlier, particles larger than 2 μm are likely to be captured during the impaction phase in the near field.

CONCLUSIONS AND RECOMMENDATIONS

A model for fission product aerosol retention for unisolated SGTR sequences has been developed using the AP600 design as an example. Several aerosol removal mechanisms have been modeled which have traditionally not been considered, or have had little if any effect, for unisolated SGTR sequences in PRAs and integrated accident analysis codes such as MAAP. These aerosol removal mechanisms include: turbulent deposition in high Reynolds number flow inside the broken tube up to the location of the break; turbulent deposition and inertial impaction in the near field, high Stokes number flow exiting the break and flowing across the tube array; and thermophoretic depo-

sition in the far field, low velocity flow regime where the fission product laden gas temperature will exceed the steam generator secondary side metal temperature.

The following conclusions are drawn regarding these aerosol removal mechanisms:

- For aerosol retention inside the tube, the flow regime is well defined and for any given accident sequence the velocity, and thus the aerosol retention from turbulent deposition, can be calculated quite readily. The main uncertainty is the location of the break. For breaks beyond about 2 meters from the tube sheet, the DF will be in the range of 100 or higher. For breaks at the tube sheet the DF is about 3 (i.e., aerosol removal over the 0.7 meter thick tube sheet only).

- For the secondary side, near field aerosol removal (i.e., jet flow exiting the break), the flow regime is more complex. The aerosol removal has been estimated using an empirical model based on measurements of turbulent deposition aerosol removal from gas flow over an external cylinder (see Figure 2). The gas velocity used in application of the empirical model was estimated by calculating the deceleration of the jet flow due to the form drag of the tube. The main uncertainties in this approach are the aerosol removal efficiency variation with gas velocity, particle size, and tube diameter, and the gas velocity as the jet expands and flows over the tube array. To minimize the effect of these uncertainties, the empirical model was applied in a conservative manner as follows: a mass mean particle diameter (3 μm) was used which is less than that expected for aerosol which has been transported from the core to the steam generator tube break; the aerosol removal efficiency was limited to 40% per tube; and at the location where velocity dropped below the value corresponding to the Stokes number at which removal efficiency was 40% (i.e., after flow over 8 tubes), no further credit for removal was taken. The DF resulting from this turbulent deposition effect is about 60.

- For the secondary side, far field aerosol removal, uniform volumetric flow was assumed across essentially the entire cross section of the steam generator. Thermophoretic deposition has been estimated as a function of heat transfer rate and particle size. Heat transfer rate has not been calculated as part of this work. Rather, several different fractions of decay heat have been assumed. Even at heat transfer rates as low as 1% of decay power (i.e., 0.2 mega-watt), the resulting DF is in the range of 2 to 3 for expected particle sizes. For heat transfer rates of 5% of decay power, the DF from thermophoresis is in the range of 95% to 99%.

- These three aerosol removal mechanisms are robust in that they are quite independent from one another and thus can be combined by multiplying the individual DFs. This results in an overall DF of several hundred, even for the most conservative break location (i.e., the tube sheet). Even without any credit for thermophoresis, the DF is expected to be 100 or higher.

Two additional phenomena which will increase aerosol removal and which have been neglected in this work are accident management actions to add feedwater to the secondary side and aerosol plugging in the broken tube(s). Adding feedwater and covering the break with water will result in aerosol scrubbing. Based on the model in reference [11], for a tube diameter of the order of 0.6 inches plugging would be expected to occur after several hundred grams of aerosol had passed through the tube. This plugging would increase aerosol retention inside the tube.

The potential for revaporization, and resulting escape to the environment, of fission products which have deposited in the steam generator tubes or on the secondary side has not been considered here. Revaporization and escape is considered unlikely for several reasons: (1) plugging will tend to occur in or near the tube sheet, which is a

26 inch thick metal plate with very large heat capacity, which will tend to hold down temperature increases from fission product heating; (2) a significant portion of the aerosols plugging the tube will be metallic material (e.g., Zr and Fe), so the tube sheet should be able to remove heat and thus reduce temperatures, and (3) even if revaporization occurs, the fission product vapors will flow through a very large, relatively cool heat transfer surface (i.e., the steam generator separators and dryers) which will tend to promote condensation on the surface and thus removal of the fission product vapors.

Reentrainment of deposited aerosols as solids has not been modeled here since it is considered that the reentrained solids will no longer be micron-sized particles but rather agglomerates that are large enough to settle easily once they pass to the secondary side of the steam generator.

Consideration should be given to performing some simple experiments for confirmatory purposes to address the uncertainties in the secondary side, near field flow regime (i.e., turbulent deposition aerosol removal efficiency and gas velocity). More detailed modeling of this flow regime would also be useful, as would heat transfer calculations to support credit for aerosol removal by thermophoresis.

The overall conclusion from this paper is that robust aerosol removal processes exist for unisolated SGTR core damage sequences, and that these processes will allow significant reductions in estimates of fission product release to the environment compared to estimates from previous work. For the unisolated SGTR sequence from AP600, a DF of 100 or larger can be justified based on the work reported here. For this DF, the iodine release for the AP600 unisolated SGTR sequence would be reduced to well under 1% of the core inventory.

REFERENCE

[1] U.S. NRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG 1150, Vol. 2, Appendix C, Dec., 1990

[2] Westinghouse, "AP600 Probabilistic Risk Assessment," DE-AC03-90SF18495, Vol. 3, Chapter 45, Rev. 3, Feb. 28, 1995

[3] Ascher H. Shapiro, (1953), "The Dynamics and Thermodynamics of Compressible Fluid Flow", John Wiley & Sons, Inc., New York.

[4] D.A. Powers and S.B. Burson, (1993), "A Simplified Model of Aerosol Removal by Containment Sprays", NUREG/CR-5966, SAND92-2689, Sandia National Laboratories, Albuquerque, NM, page 84.

[5] R.J. Lipinski et al., (1985), "Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions Volume II TMLB' Analyses", SAND84-0410, Vol. 2, Sandia National Laboratories, Albuquerque, NM.

[6] B. Y. H. Liu and J. K. Agarwal, (1974) "Experimental Observation of Aerosol Deposition in Turbulent Flow", J. Aerosol Sci. Vol. 5, pages 145-155.

[7] W.M. Kays and M.E. Crawford, (1980), "Convective Heat and Mass Transfer", McGraw-Hill Book Company, New York.

[8] P.L. Douglas and S. Ilias, (1988), "On the Deposition of Aerosol Particles on Cylinders in Turbulent Cross Flow", J. Aerosol Sci. Vol. 19, No. 4, pages 451-462.

[9] F.M. White, (1974), "Viscous Fluid Flow", McGraw-Hill Inc., New York.

[10] L. Talbot et al., (1980), "Thermophoresis of particles in a heated boundary layer", J. Fluid Mech. Vol. 101, pages 737-758

[11] H.A. Morewitz, "Leakage of Aerosols from Containment Buildings," Health Physics, Vol. 42, No. 2, February, 1982.

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/28/97

Selection: [item no] between 338 And 338 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
338	NRR/SPLB	9.5.4	MTG-OI		Winters/BRC	Closed	Action W		

M9.5.4-13 (DIESEL GENER' .OR AUXILIARY SYSTEMS)
Revise the AP600 SSAR to make Table 8.3.1-1 and Table 9.5.4-2 agree.
a. Add the fuel oil storage tank (Lo) level alarm to Table 8.3.1-1 of the SSAR.
b. Show the day tank level as alarming locally in Table 8.3.1-1 of the SSAR.
c. Table 8.3.1-1 of the SSAR does not show low fuel oil pressure, moisture separator differential pressure, filter differential pressure, pump suction strainer differential pressure, heater in service, heater low temp out, and fuel oil tank fill strainer differential pressure as alarming locally in the DG building and as a combined trouble alarm in the control room.

Closed. DISCUSSED AT 12/13/94 MEETING BETWEEN WESTINGHOUSE AND NRC PLANT SYSTEMS BRANCH.

NRC status changed to Action W based on 11/5/96 telecon. NRC will provide markup of deleted table to indicate items important enough to put back into the SSAR. Then Westinghouse will provide an SSAR markup. See NRC letter of 12/9/96.

We haven't received this yet. Maybe this should be "Action N" until we do.

Jim
01/28/97

Post-it™ brand fax transmittal memo 7671 | # of pages ▸ 1
To DIANE JACKSON | From JIM WINTERS
Co. USNRC | Co. WESTINGHOUSE
Dept. | Phone # 412-374-5290
Fax # | Fax #

 Total Records: 1

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 28, 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:			

Cover + Pages 1 + 2

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 243 PER OUR 11/5/96 TELECON.

IT WILL GO INTO SSAR REVISION 11 UNLESS WE HEAR FROM YOU.

I CHANGED THE "NRC STATUS" PER OUR 11/5/96 TELECON.

Jim Winters

CC: LINDGREN
MCINTYRE
CUMMINS
RON VIJUK
WINTERS
MCDERMOTT
HUTCHINGS
JEANNE EVANS.

The high-pressure air subsystem includes pressure and carbon monoxide instrumentation, automatic protection devices, and temperature indication.

9.3.2 Plant Gas System

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to the plant systems as required.

Other gases, such as oxygen, methane, acetylene, and argon, are supplied in smaller individual containers and are not supplied by the plant gas system.

9.3.2.1 Design Basis

9.3.2.1.1 Safety Design Basis

The plant gas system serves no safety-related function and therefore has no nuclear safety design basis.

9.3.2.1.2 Power Generation Design Basis

The nitrogen portion of the plant gas system supplies nitrogen for pressurizing, blanketing, and purging of various plant components.

The hydrogen gas portion of the plant gas system supplies hydrogen to the main plant electrical generator for cooling as well as to other plant auxiliary systems.

The carbon dioxide portion of the plant gas system supplies carbon dioxide to the generator for purging of hydrogen and air during layup or plant outages.

9.3.2.2 System Description

Classification of equipment and components is given in Section 3.2.

9.3.2.2.1 General Description

The nitrogen portion of the plant gas system is a packaged system consisting of a liquid nitrogen storage tank and vaporizers. Nitrogen gas is supplied in both a high-pressure and a low-pressure subsystem. The high-pressure subsystem uses a pump to pressurize the gas supplying the accumulators in the passive core cooling system. The high-pressure supply is then reduced to supply makeup to the reactor coolant drain tank for purging and blanketing. Low-pressure nitrogen is provided for component purging, layup/blanketing, and pressurization.

stored within the valves and their operators

The main steam isolation valves (MSIVs) and main feedwater isolation valves (MFIVs) use compressed nitrogen as the motive force to close the valves. The main steam isolation valves are described in subsection 10.3.2.2.4 and the main feedwater isolation valves are described

in subsection 10.4.7.2.2. Nitrogen makeup for these valves (if needed) is provided ~~via~~ from temporary connections, ~~to~~ portable high-pressure nitrogen bottles using on the valves.

The packaged nitrogen system is located inside the turbine building.

The hydrogen gas portion of the plant gas system is a packaged system consisting of a liquid hydrogen storage tank and vaporizers to supply hydrogen gas to the main generator for generator cooling and to the demineralized water transfer and storage system to support removal of dissolved oxygen and to other miscellaneous services. The hydrogen supply package system is located outdoors at the hydrogen storage tank area.

The carbon dioxide portion of the plant gas system, which is a packaged system consisting of one liquid storage tank and a vaporizer, produces gaseous carbon dioxide to purge the main generator. This packaged system is located in the turbine building.

Liquid gas storage tanks are built in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, 1989.

9.3.2.2.2 Component Description

Liquid Nitrogen Storage Tank

Liquid nitrogen is stored under its own vapor pressure as a saturated liquid in a dual wall tank. This tank supplies nitrogen for the high- and low-pressure nitrogen gas systems. The annular space between the inner and outer tank walls is filled with insulation and evacuated when the tank is cold.

Liquid Nitrogen Pump

A cryogenic liquid nitrogen pump is utilized to provide a supply of high-pressure nitrogen. It is a single-cylinder, positive displacement pump with the entire "cold" pumping assembly enclosed in a vacuum-jacket, which permits the pump to remain cold in the standby condition.

Nitrogen High-Pressure Ambient Air Vaporizer

Liquid nitrogen is vaporized by a high-pressure natural convection vaporizer, which vaporizes and superheats cryogenic nitrogen using heat from the ambient air.

Nitrogen Low-Pressure Ambient Air Vaporizer

The low-pressure vaporizer unit has two parallel banks. In the event of frost buildup on the active bank, flow is redirected to the opposite bank while the other bank defrosts.

FAX to DINO SCALETTI

January 29, 1997

CC: Sharon, please make copies for:

Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
Robin Nydes
Cindy Haag
Don Lindgren
John Butler
Bob Tupper
Bruce Rarig
Brian McIntyre
Ed Cummins
Bob Vijuk

NRC is requested to please acknowledge receipt of information related to each of the following Open Items. These are a subset of the items with "Action W" in "NRC Status" for which I have personnally checked, since the first of the year, that we have submitted what we believe is the resolving information. Recognizing that reviewing for completeness of the response in each case constitutes an NRC action, we recommend that receipt acknowledgement be accompanied by direction to change their "NRC Status" to "Action N". If these are truly "Action W", please provide a description of the action Westinghouse is expected to take. We know of no action required. This is the third weekly request of this type.

4, 21, 30, 37, 123, 134, 135, 137, 138, 139, 140, 141, 157, 158, 164, 182, 184, 262, 293, 300, 305, 308, 319, 457, 458, 586, 801, 802, 805, 807, 809, 972, 973, 1009, 1037, 1038, 1039, 1040, 1041, 1043, 1045, 1052, 1053, 1055, 1101, 1102, 1195, 1197, 1317, 1458, 1461, 1697, 1698, 1699, 1700, 1701, 1702, 1703, 1704, 1707, 1716, 1727, 1730, 1731, 1742, 1745, 1747, 1749, 1753, 1760, 1996, 1999, 2018, 2019, 2023, 2024, 2025, 2040, 2045, 2051, 2199, 2200, 2201, 2202, 2272, 2273, 2442, 2457, 2939, 2942, 2945, 2958, 2959, 2960, 2961, 2962, 2963, 2964, 2965, 2966, 2967, 2968, 2969, 2970, 2971, 2972, 2973, 2974, 2975, 2976, 2977, 2978, 2979, 2981, 2982, 2983, 2984, 2985, 2986, 3122, 3126. 3127, 3128, 3197, 3372, 3427, 3505, 3517, 3895, 3944, 3945, 3946, 3947, 3948, 3949, 3950, 3951, 3952, 3953, 3954, 3955, 3956, 3957, 3958, 4123, 4124, 4125, 4126, 4127, 4128, 4129, 4130, 4131, 4132, 4133, 4134, 4135, 4136, 4137, 4138, 4139, 4140, 4141, 4142, 4143, and 4144.

Jim

Thanks
Jim Winters
412-374-5290

FAX to DINO SCALETTI

January 29, 1997

CC: Sharon, please make copies for: Ted Quay
Diane Jackson
Don Lindgren
Don Hutchings
Brian McIntyre

OPEN ITEM #586

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items for sections scheduled to have complete FSER drafts by now. Attached are copies of some of the relevant documentation related to Open Item #586. This is the only open item associated with section 3.5.1. with an "NRC Status" of "Action W". We (NRC and W) promised each other in late 1996 that section 3.5.1.3 would be ready for a final FSER draft in December of 1996. See the draft Activity Plan sheet attached which includes all NRC comments on the schedule. We provided a revision to SSAR subsection 10.2.6 with Revision 5, February 29, 1996, (over 10 months ago). This was a very straightforward change and implemented NRC requests into the SSAR as we believed it was agreed. It is outside the scope of Design Certification for the SSAR to explicitly impose a time table for submittal of a turbine maintenance program. This will be part of the COL application process as required by subsection 10.2.6. It seems a reasonable request that NRC acknowledge receipt of the change. Our records show no outstanding Westinghouse action on this item (#586) or any other item associated with subsection 3.5.1.3. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Resolved". I will be calling you or Diane Jackson about 1:30 today to see if you have additional information on this item. Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/28/97

Selection: [item no] between 586 And 586 Sorted by Item #

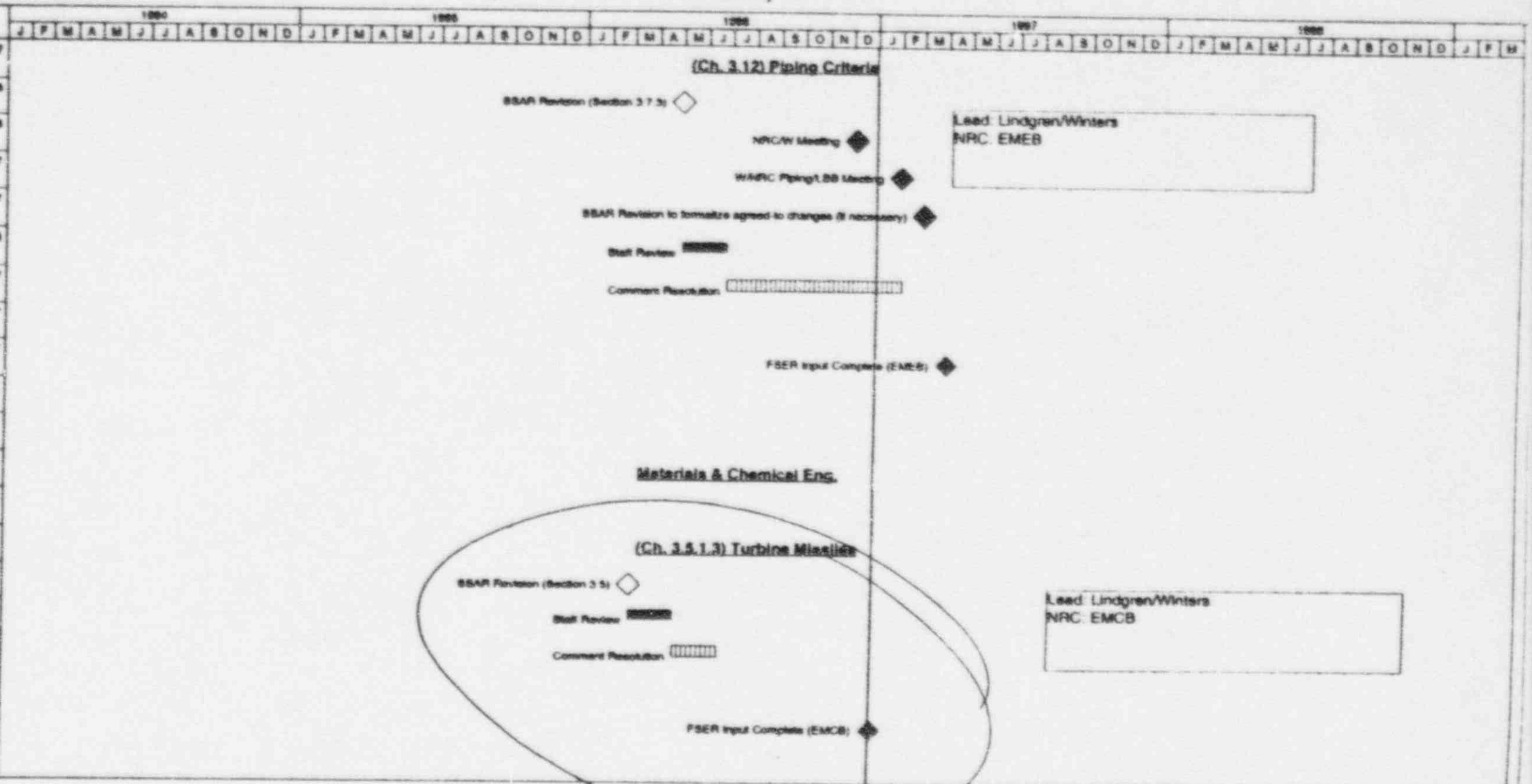
Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No /	Date
586	NRR/EMCB	3.5.1.3-13	DSER-OI	(TURBINE MAINTENANCE PROGRAM & MISSILE PROBABILITY CALCULATION) Westinghouse should add COL Action Item 3.5.1.3-5 to the SSAR. Provide turbine maintenence program including probability calculations of turbine missile generation. Closed - Addressed by COL item in SSAR section 10.2.6 Action W - NRC wants time table for submittal of turbine mainteance program in SSAR.	Lindgren	Closed	Action W		

Total Records: 1

2b4

AP600 DESIGN CERTIFICATION
(DRAFT ACTIVITY PLAN)

ID	Unique ID	Start	Finish
276	1877	4/30/96	3/31/97
277	3480	4/30/96	4/30/96
278	3486	12/5/96	12/5/96
279	3437	1/31/97	1/31/97
280	3240	2/28/97	2/28/97
281	3042	4/30/96	8/24/96
284	3046	8/25/96	1/31/97
285	3047	3/31/97	3/31/97
286	3048	3/31/97	3/31/97
287			
288			
289	1884	2/26/96	12/31/96
290			
291	1886	2/26/96	12/31/96
292	3435	3/26/96	3/26/96
293	3049	3/26/96	4/24/96
296	3053	4/25/96	6/10/96
297	3054	12/31/96	12/31/96
298	3055	12/31/96	12/31/96
3??			



3d4

Project: AP600 Design Certification Date: 12/30/96	W Task NRC Task		W Milestone NRC Milestone		Completed Milestone Summary

- Hydrogen gas pressure
- Hydrogen gas purity
- Generator winding overtemperature
- Generator ampere, voltage, and power

Additional generator protective devices are listed in Table 10.2-3.

10.2.6 Combined License Information on Turbine Maintenance and Inspection

The Combined License holder will implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis.

10.2.7 References

1. WSTG-4-P, Proprietary and WSTG-4-NP, Nonproprietary, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines," October 1984.

2. WCAP-11525, Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency, 1987.

3. NUREG-1275, Vol. 11, Operating Experience Feedback Report - Turbine-Generator Overspeed Protection Systems, Commercial Power Reactors, H. L. Ornstein, Nuclear Regulatory Commission, April 1995.

FAX to DINO SCALETTI
& BILL HUFFMAN

January 29, 1997

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

Don Lindgren
Jim Grover
Brian McIntyre
Ed Cummins

OPEN ITEMS FOR SECTION 6.1

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items for sections scheduled to have complete FSER drafts by now. Attached are copies of some of the relevant documentation related to Section 6.1. There are 3 open items (969, 970, and 971) associated with section 6.1. with an "NRC Status" of "Action W". We (NRC and W) promised each other in late 1996 that section 6.1 would be ready for a final FSER draft in December of 1996. See the draft Activity Plan sheet attached which includes all NRC comments on the schedule. We provided a revision to SSAR subsections 6.1.2, 6.1.3, and 6.2.4 with Revision 7, April 30, 1996, (over 8 months ago). These were very straightforward changes and implemented NRC requests into the SSAR as we believed it was agreed. It seems a reasonable request that NRC acknowledge receipt of the changes. Our records show no outstanding Westinghouse action on these items or any other item associated with subsection 6.1. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Resolved". I will be calling you or Bill Huffman about 2:00 today to see if you have additional information on this item. Thank you.

Jim

Jim Winters
412-374-5290

1 of 10

Selection: [nrc st code]='Action W' And [DSER Section] like '6.1*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
969	NRR/EMCB	6.1.2-1	DSER-OI		Lindgren/Schulz	Closed	Action W		
				Westinghouse should further justify the non-safety related classification of all coatings within the AP600 containment.					
				Closed - Added requirement for debris transport evaluation of coatings and description of methodology to SSAR Rev. 7, subsection 6.1.2.1					
970	NRR/EMCB	6.1.2-2	DSER-OI		Lindgren/Schulz	Closed	Action W		
				Westinghouse should provide more information indicating that the appropriate coatings will be correctly applied and will provide adequate protection throughout the plant's life. In addition, Westinghouse should supply data and an in-depth analysis to provide justify using new coating types (such as high-top coatings) in containment.					
				Closed - SSAR Rev. 7, subsection 6.1.2.1.6 includes the requirement for a program by the Combined Lisence applicant to control testing, application, and monitoring of nonsafety-related coatings.					
9[illegible]	NRR/EMCB	6.1.2-3	DSER-OI		Grover	Closed	Action W		
				Westinghouse should indicate in the SSAR whether the changes in the recommended practices result in a greater or lesser amount of predicted hydrogen production.					
				Closed - See status detail for open item 6.2.5.2-3. Westinghouse ~~will recalculate~~ recalculated the design basis hydrogen generation using the source term specified in RG 1.7.					

2910

AP600 Open Item Tracking System Database: Executive Summary

Date: 1/29/97

Selection: [DSER Section] like '6.2.5.2-3' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
1008	NRR/SCSB	6.2.5.2-3	DSER-OI	The percentage of core fission product inventory in the sump solution proposed by Westinghouse is unacceptable. Closed: The design basis LOCA hydrogen generation is analyzed using the source term as specified in RG 1.7.	Narula/Grover	Closed	Action W		

3810

 Total Records: 1

AP600 DESIGN CERTIFICATION
(DRAFT ACTIVITY PLAN)

ID	Unique ID	Start	Finish	Task (1994 – 1998)
361	2086	1/11/96	7/21/97	Chapter 6 - Engineered Safety Features
362				(Ch. 6.1) ESF Materials
363	2437	4/30/96	12/31/96	
364	2438	4/30/96	4/30/96	SSAR Revision (Section 6.1)
365	2804	4/30/96	6/24/96	Staff Review
368	2808	6/25/96	8/19/96	Comment Resolution
369	2809	12/31/96	12/31/96	
370	2810	12/31/96	12/31/96	FSER Input Complete (EMCB)
371				
372				(Ch 6.2) Containment Systems
373	2730	1/11/96	7/21/97	
374	2738	3/29/96	3/29/96	SSAR Section 6.2 Revision (except 6.2.1)
375	3880	1/11/96	1/11/96	W submit PAR supporting information
376	3881	2/27/96	2/27/96	Mtg on PARS
377	3882	8/30/96	8/30/96	W responds to PAR RAIs & OIs
378	3813	12/6/96	12/6/96	NRC Response to West. Igniter Power Supply Change
379	3864	12/31/96	12/31/96	NRC RAIs on PAR Mixing and Stratification
380	3836	2/17/97	2/17/97	Response to PAR Mixing/Stratification RAIs
381	3865	12/31/96	12/31/96	Response to RAI on IRWST/Sparger Loads
382	3809	2/28/97	2/28/97	SSAR Revision (Section 6.2.1)
383	3810	4/15/97	4/15/97	Resolution of Hydrogen Igniter Power and Placement Issue
384	3263	1/11/96	3/27/97	Staff Review
387	3267	3/28/97	5/22/97	Comment Resolution
388	3288	7/21/97	7/21/97	
389	3289	7/21/97	7/21/97	FSER Input Complete (SCSB)
390				

Lead: Lindgren/Winters
NRC: EMCB

Leads: Butler/Gresham
NRC: SCSB

Resolution of H2 igniter placement issue affected by diffusion flame report review covered under Chapter 19

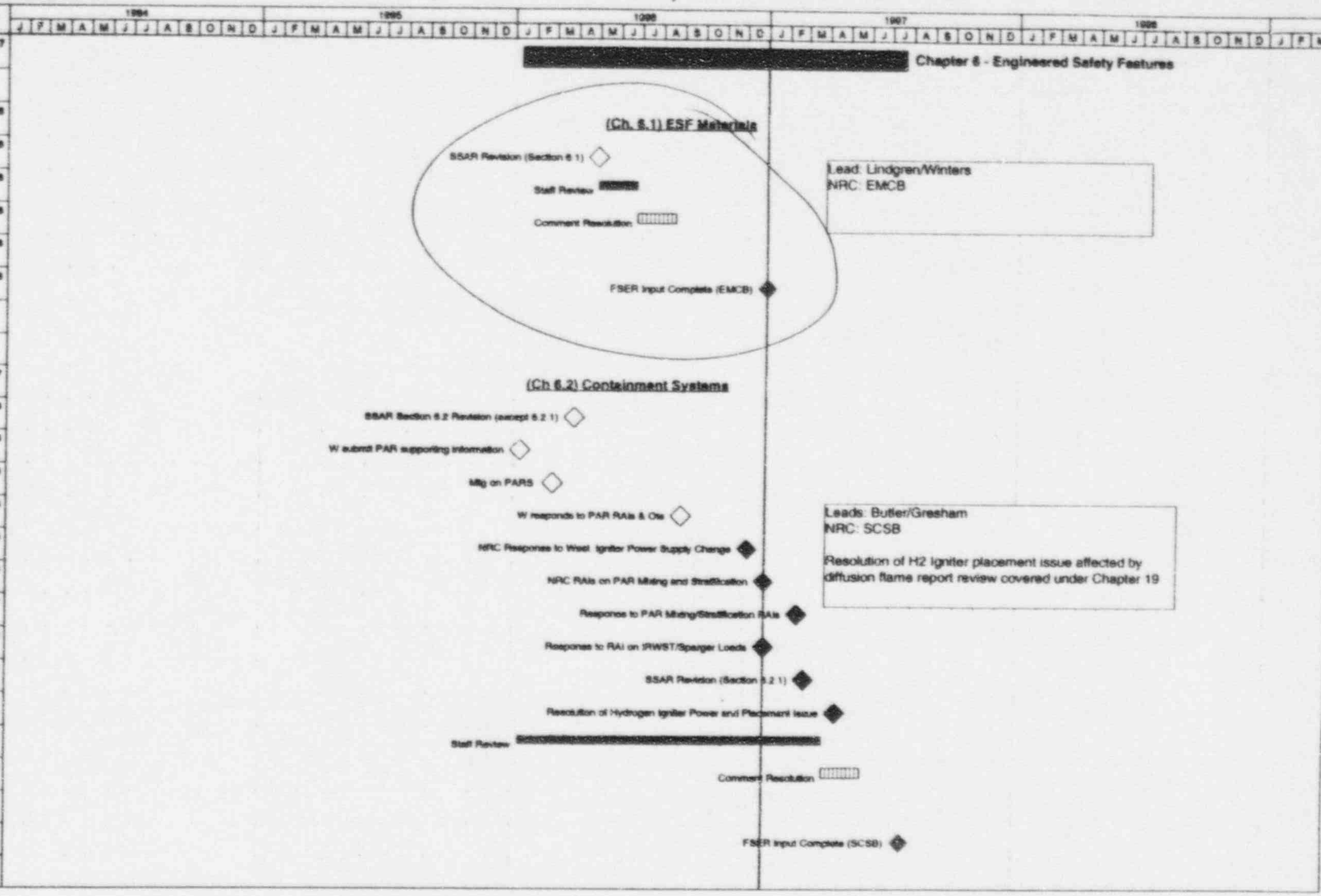
Project: AP600 Design Certification
Date: 12/30/96

W Task — W Milestone — Completed Milestone
NRC Task — NRC Milestone — Summary

4 of 10

AP600 DESIGN CERTIFICATION
(DRAFT ACTIVITY PLAN)

ID	Unique ID	Start	Finish
361	2086	1/11/96	7/21/97
362			
363	2437	4/30/96	12/31/96
364	2438	4/30/96	4/30/96
365	2804	4/30/96	6/24/96
366	2808	6/25/96	8/16/96
369	2809	12/31/96	12/31/96
370	2810	12/31/96	12/31/96
371			
372			
373	2730	1/11/96	7/21/97
374	2779	3/28/96	3/28/96
375	3280	1/11/96	1/11/96
376	3281	2/27/96	2/27/96
377	3282	[illegible]/96	6/20/96
378	3813	12/[illegible]	12/6/96
379	3584	12/31/96	12/31/96
380	3839	2/17/97	2/17/97
381	3585	12/31/96	12/31/96
382	3809	2/28/97	2/28/97
383	3810	4/15/97	4/15/97
384	3283	1/11/96	3/27/97
387	3287	3/28/97	5/22/97
388	3288	7/21/97	7/21/97
389	3289	7/21/97	7/21/97
390			



Project: AP600 Design Certification Date: 12/30/96	W Task	W Milestone	Completed Milestone		
	NRC Task	NRC Milestone	Summary		

4 of 10

6.1.2 Organic Materials

6.1.2.1 Protective Coatings

6.1.2.1.1 General

The AP600 is divided into four areas with respect to the use of protective coatings. These four areas are:

- Inside containment
- Exterior surfaces of the containment vessel
- Radiologically controlled areas outside containment
- Remainder of plant.

The considerations for protective coatings differ for these four areas and the coatings selection process accounts for these differing considerations.

The AP600 design considers the function of the coatings, their potential failure modes, and their requirements for maintenance. Local failure of the coatings on the exterior of the containment vessel and outside the containment does not prevent functioning of the engineered safety features required for safe shutdown of the plant during or after a design basis accident (DBA). Coatings inside the containment are evaluated to demonstrate that failure does not prevent functioning of the engineered safety features. Since failure of the coatings does not prevent functioning of the engineered safety features, the coatings are classified as nonsafety-related. Protective coatings are maintained to provide corrosion protection for the containment pressure boundary and for other safety-related system components inside containment. The coatings on the outside of the containment vessel are maintained to provide corrosion protection for the containment pressure boundary and to support passive cooling through their wetting ability and heat transfer properties. These functions are in addition to other functions (such as enhancing decontamination inside the containment and assisting in general housekeeping). The corrosion protection and decontamination functions of the coatings are nonsafety-related functions.

969

For information on coating design features, material and application requirements, and performance monitoring requirements, see subsection 6.1.2.1.6.

6.1.2.1.2 Inside Containment

Carbon Steel

Inorganic zinc primer is the basic coating applied to the containment vessel, structural carbon steel, and carbon steel equipment/components inside containment that need coating. Below the operating floor, most of the inorganic zinc primer is top coated with epoxy to enhance decontamination. The epoxy top coat also extends above the operating floor on structural modules and to a wainscot height of 7 feet above the operating floor on the containment vessel. Where practical, miscellaneous carbon steel items (such as stairs, gratings, ladders,

6.1.2.1.6 Design Features

A number of design features provide confidence that the coating systems inside the containment, on the exterior of the containment vessel and in potentially contaminated areas outside containment will perform as intended. These features enhance the ALARA program and enhance corrosion resistance. The features include:

- Specification of qualified coating materials
- Provision of coating specifications
- Provision of coating procedures
- Use of qualified painters
- Use of qualified coatings inspectors
- Documentation of coatings work
- Performance of as much coating work as practical under controlled shop conditions
- Specification of coating performance monitoring
- Specification of coating inspection and maintenance

-970

Testing and Application

The testing, application, and monitoring of nonsafety-related coatings are controlled by a program prepared by the Combined License applicant. The specified coatings used inside containment are tested for radiation tolerance per ASTM D4082 (Reference 1), for decontaminability per ASTM D4256 (Reference 2) and for performance under design basis accident conditions per ASTM D3911 (Reference 3). The coatings used in radiologically controlled areas outside containment are tested for radiation resistance and decontaminability but are not specified to be design basis accident tested. Where practical, the same coating materials are used in radiologically controlled areas outside containment as are used inside containment. This provides a high level of quality and optimizes maintenance painting over the life of the plant. The coatings used on the outside of the containment shell are not required to be tested for radiation tolerance, decontaminability, or performance under design basis accident conditions.

The coating manufacturer is required to manufacture the coatings under a suitable quality assurance program and to provide a product identity certification record. Coating specifications also require that the surfaces to be coated are properly prepared, coated, inspected and documented.

For coatings used inside containment, radiologically controlled areas outside containment, and on the outside of the containment shell, the coating applicator submits and follows acceptable procedures to provide confidence that correct coating practices are used, that the painters are qualified and certified in accordance with ASTM D4227 (Reference 4) and ASTM D4228 (Reference 5), and that the inspectors are qualified and certified in accordance with ASTM D4537 (Reference 6).

Due to the modularized construction, the majority of the containment coatings are shop applied to the containment vessel and to piping, structural and equipment modules. Fewer coating applicators are needed than would be if many individual items were coated by a wide variety of manufacturers and applicators and shipped to the site for individual installation. This restricted application of the coatings under controlled shop conditions provides additional confidence that the coatings will perform as designed and as expected.

Performance Monitoring

The performance of the installed coatings is monitored. The coatings performance monitoring program includes periodic inspections of the coatings inside containment and the exterior of the containment vessel during planned outages. Periodic inspections are also specified for coatings installed in the radiologically controlled areas outside containment. The monitoring program includes the planning of maintenance painting schedules so that the installed coatings can be maintained to perform as intended. The maintenance painting inside containment and on the exterior of the containment vessel is conducted during scheduled outages using qualified maintenance coating systems applied and inspected by qualified personnel.

6.1.2.2 Other Organic Materials

A listing of other organic materials in the containment is developed based on the specific type of equipment and the supplier selected to provide it. Materials are evaluated for potential interaction with engineered safety features to provide confidence that the performance of the engineered safety features is not unacceptably affected.

6.1.3 Combined License Information Items

6.1.3.1 Procedure Review

The Combined License applicants referencing the AP600 will address review of vendor fabrication and welding procedures or other quality assurance methods to judge conformance of austenitic stainless steels with Regulatory Guides 1.31 and 1.44.

6.1.3.2 Coating Program

The Combined License applicants referencing the AP600 will address preparation of a program to control testing, application, and monitoring of nonsafety-related coatings. A coating transport evaluation similar to that described in subsection 6.1.2.1.5 will be performed.

970

For the post-accident cooling solution in which the fission products released from the core are assumed to be dissolved, energy is emitted directly into the solution. All of the beta radiation is assumed to be absorbed by the water. Since the mass of water is relatively large compared to the penetrating capability of gamma radiation, it is also assumed that 100 percent of the gamma radiation energy is absorbed by the water.

The radiolytic decomposition of water is a reversible reaction. In the reactor vessel, where the products of radiolysis are continuously flushed away by the circulation of cooling solution, there is little chance for hydrogen and oxygen to accumulate. Consequently, recombination of hydrogen and oxygen is assumed not to occur because significant quantities of the two reactants are not available.

The post-accident cooling solution in the sump, however, is a deep and relatively static environment where the products of radiolysis are lost from solution primarily by molecular diffusion. Tests simulating post-accident sump conditions demonstrate that there is significant reverse reaction in the sump. Hence, there is an apparent reduction in the quantity of hydrogen produced per unit energy absorbed by the water.

The results of Westinghouse and Oak Ridge National Laboratory studies indicate maximum hydrogen yields of 0.44 molecules per 100 eV for core radiolysis and 0.3 molecules per 100 eV for solution radiolysis. The results of these studies are published in References 10, 11, and 12.

The analysis performed for the AP600 assumes a hydrogen yield of 0.5 molecules per 100 eV for both the core and the solution radiolysis cases. This value is conservative relative to the referenced studies and is consistent with the guidance of Regulatory Guide 1.7.

971

In a design basis loss of coolant accident there is expected to be no damage to the core and thus no release of activity from the core to the sump solution. The source term used for determining radiolysis production of hydrogen is conservatively based on guidance of Regulatory Guide 1.7 which states that 100 percent of noble gases, 50 percent of iodines, and 1 percent of other nuclides are assumed to be released from the core even though it is inconsistent with the limited amount of fuel cladding reaction that is determined to take place. Appendix 15A provides the core fission product inventory at shutdown.

Table 6.2.4-4 contains a summary of the assumptions used in the analysis of hydrogen produced from radiolysis. Produc[illegible]on rate of hydrogen as a function of time is shown [illegible]y in Figure 6.2.4-3 and the production of hydrogen is shown in Figure 6.2.4-4.

6.2.4.3.1.[illegible] [illegible]on of Metals

In the environment that would exist inside the containment following a postulated loss of coolant accident, aluminum and zinc corrode to form hydrogen gas. Table 6.2.4-4 lists the inventory of aluminum and zinc inside the containment.

971

Table 6.2.4-4 (Sheet 2 of 3)

ASSUMPTIONS USED TO CALCULATE HYDROGEN PRODUCTION FOLLOWING A LOSS OF COOLANT ACCIDENT

Corrosion of Materials

Aluminum inventory in containment 971

Component	Weight (lb)	Surface (sq. ft)
Excore detectors	25	8
Flux mapping system	120	84
Miscellaneous valve parts	230	86
RCDM connectors	190	42
Paint	140	18,000
Contingency	250	85
Other non-NSSS items	500	100
Total aluminum	1,455	

Table 6.2.4-4 (Sheet 3 of 3)

ASSUMPTIONS USED TO CALCULATE HYDROGEN PRODUCTION FOLLOWING A LOSS OF COOLANT ACCIDENT

Zinc inventory in containment — 971

Component	Weight (lb)	Surface (sq. ft)
Cable trays	310	2,100
Conduit	500	3,500
Hangers	24	170
Junction boxes	100	730
Paint	1,200	72,000
Gratings	680	41,000
HVAC ductwork	840	5,900
Stairs	13	800
Pipe supports	510	30,000
Contingency	1,050	39,000
Total zinc	5,227	

Aluminum corrosion rate	See Table 6.2.4-5
Zinc corrosion rate	See Table 6.2.4-5
Containment temperature	See Table 6.2.4-5
Solution pH	7 - 9.5

Initial Reactor Coolant Hydrogen Inventory

Hydrogen concentration in reactor coolant (cc at STP per kg)	40
Reactor coolant mass (lb)	353,000

FAX to DINO SCALETTI

January 29, 1997

CC: Sharon, please make copies for: Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre
Ed Cummins
Bob Vijuk

As I believe you requested, this is a reminder list of the Open Items where we have documented the difference between "W Status" and "NRC Status". In all cases, we believe the next action is at NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W".

Open Item Number	Westinghouse Submittal	Request for Status Change
4 (RAI 410.262)	9/10/96	1/9/97
21 (RAI 471.24)	5/20/96	1/9/97
30 (RAI 952.99)	5/13/96 12/17/96 - repeat	1/13/97
37 (RAI 260.74)	4/22/96	1/14/97
123 (M3.6.1-2)	4/30/96	1/15/97
134 (M3.11-1)	2/29/96 6/19/96 - more	1/16/97
135 (M3.11-2)	2/29/96 6/19/96 - more	1/17/97
137 (M3.11-4)	6/19/96	1/23/97
138 (M3.11-5)	2/29/96	1/24/97
139 (M3.11-6)	2/29/96 6/19/96 - more	1/24/97
140 (M3.11-7)	2/29/96	1/23/97
141 (M3.11-8)	2/29/96	1/24/97
586	6/19/96	1/28/97
969 (DSER 6.1.2-1)	4/30/96	1/29/97

Open Item Number	Westinghouse Submittal	Request for Status Change
970 (DSER 6.1.2-2)	4/30/96	1/29/97
971 (DSER 6.1.2-3)	4/30/96	1/29/97

Thanks for your help.

Jim

Jim Winters

2 l 2

FAX to DINO SCALETTI

January 30, 1997

CC: Sharon, please make copies for:

Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky
John Butler
Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre
Ed Cummins
Bob Vijuk

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134 (M3.11-1)	2/29/96 6/19/96 - more	1/16/97
135 (M3.11-2)	2/29/96 6/19/96 - more	1/17/97
137 (M3.11-4)	6/19/96	1/23/97
138 (M3.11-5)	2/29/96	1/24/97
139 (M3.11-6)	2/29/96 6/19/96 - more	1/24/97
140 (M3.11-7)	2/29/96	1/23/97
141 (M3.11-8)	2/29/96	1/24/97

Open Item Number	Westinghouse Submittal	Request for Status Change
586	6/19/96	1/28/97

Note that the status was changed for Items 969, 970 and 971, so they have been removed from the table.

Thanks for your help.

Jim

Jim Winters

FAX to DINO SCALETTI
& DIANE JACKSON

January 30, 1997

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

Don Lindgren
Don Hutchings
Brian McIntyre
Ed Cummins
Bob Vijuk

OPEN ITEM #586

This is out third request for information on Open Item 586. In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items for sections scheduled to have complete FSER drafts by now. On both January 29 and 30, we forwarded copies of some of the relevant documentation related to Open Item #586. This is the only open item associated with section 3.5.1. with an "NRC Status" of "Action W". We (NRC and W) promised each other in late 1996 that section 3.5.1.3 would be ready for a final FSER draft in December of 1996. We provided a revision to SSAR subsection 10.2.6 with Revision 5, February 29, 1996, (over 10 months ago). This was a very straightforward change and implemented NRC requests into the SSAR as we believed it was agreed. It is outside the scope of Design Certification for the SSAR to explicitly impose a time table for submittal of a turbine maintenance program. This will be part of the COL application process as required by subsection 10.2.6. It seems a reasonable request that NRC acknowledge receipt of the change. Our records show no outstanding Westinghouse action on this item (#586) or any other item associated with subsection 3.5.1.3. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Resolved". I will be calling you or Diane Jackson about 1:30 today to see if you have additional information on this item. Thank you.

Jim

Jim Winters
412-374-5290

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY ~~29~~ 30, 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:			

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

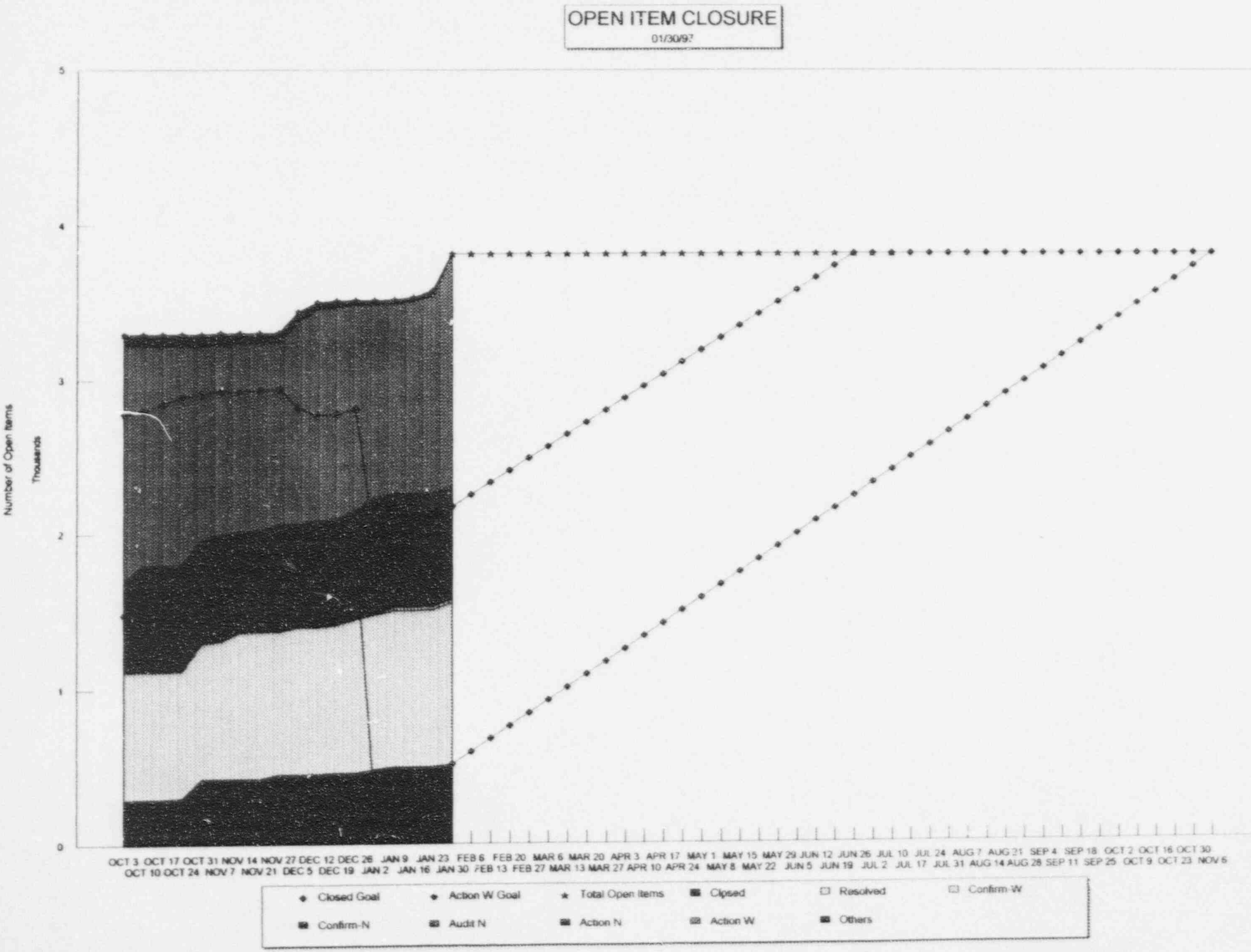
Here is an example table for Jeff. Recognize that this is not required by the BTP but if he wants a table like this, please find out if this example meets his needs.

Jim Winters

Example Table for AP600 Fire Hazards Analysis						
	Reactivity Control	Decay Heat Removal	Reactor Coolant Makeup	Reactor Pressure Control	Process Monitoring	Support Systems
AP600	Control Rods Core Makeup Tanks Borated Injection - CMT/Acc/IRWST	PRHR and PCCS ADS and CMT / Acc / IRWST and PCCS	CMT ADS and CMT / Acc / IRWST	Safety Valves	PAMS B1 Variables	Class 1E and DC UPS Protection and Safety Monitoring System (PMS)
1100 AF 01 / 1100 AF 11105 Neutron Detectors Cont. Level Instrument (B)	Control Rods Core Makeup Tanks Borated Injection - CMT/Acc/IRWST	PRHR and PCCS	CMT ADS and CMT / Acc / IRWST	Safety Valves	PAMS B1 Variables Cont. Level Instruments (A,C)	Class 1E and DC UPS PMS
1100 AF 01 / 1100 AF 11204 Cont. Level Instruments (A&C) Hot Leg 1 Level & Pressure Instruments (A&C)	Control Rods Core Makeup Tanks Borated Injection - CMT/Acc/IRWST	PRHR and PCCS	CMT ADS and CMT / Acc / IRWST	Safety Valves	PAMS B1 Variables Cont. Level Instruments (B) Hot Leg 2 Level & Pressure Instruments (B&D)	Class 1E and DC UPS PMS

Example Table for AP600 Fire Hazards Analysis						
	Reactivity Control	Decay Heat Removal	Reactor Coolant Makeup	Reactor Pressure Control	Process Monitoring	Support Systems
1100 AF 01 / 1100 AF 11206 PXS Valves Powered by Division B/D Cont. Isolation Valve (Inside Cont.) Hot Leg 2 Level & Pressure	Control Rods (Auto or Manual) Core Makeup Tanks (Valves Powered by A/C) Borated Injection - CMT/Acc/IRWST (Valves Powered by A/C)	PRHR and PCCS Cont. Isolation Valve (Outside Cont.)	CMT (Valves Powered by A/C) ADS and CMT / Acc / IRWST (Valves Powered by A/C)	Safety Valves	PAMS B1 Variables Hot Leg 1 Level & Pressure Instruments	Class 1E and DC UPS PMS

Example Table for AP600 Fire Hazards Analysis						
	Reactivity Control	Decay Heat Removal	Reactor Coolant Makeup	Reactor Pressure Control	Process Monitoring	Support Systems
1100 AF 01 / 1100 AF 11300B	Control Rods (Auto or Manual)	ADS and CMT / Acc / IRWST and PCCS (CMT A, IRWST Level Instruments A&C)	CMT	Safety Valves	PAMS B1 Variables	Class 1E and DC UPS
PRHR Control Valves (A&B)	Core Makeup Tanks (CMT A)		ADS and CMT / Acc / IRWST (CMT A, IRWST Level Instruments A&C)			PMS
CMT B and Level Instruments	Borated Injection - CMT/Acc/IRWST (CMT A, IRWST Level Instruments A&C)					
IRWST Level Instruments (B&D)						
PRHR Flow (B&C) and Outlet Temp. (C)					Core Exit T/Cs	
Pzr Pressure and Level Instruments (A&C)					Pzr Pressure and Level Instruments (B&D)	
Cold Leg Flow (A&C)					Cold Leg Flow (B&D)	
SG Level & Steamline Pressure Instruments (A&C)					SG Level & Steamline Pressure Instruments (B&D)	
Cont. Isol. Valves Inside Cont. (A/C)		Cont. Isol. Valves Outside Cont (B/D)				


OPEN ITEM CLOSURE
01/30/97
Number of Open Items
Thousands
5
4
3
2
1
0
OCT 3 OCT 17 OCT 31 NOV 14 NOV 27 DEC 12 DEC 26 JAN 9 JAN 23 FEB 6 FEB 20 MAR 6 MAR 20 APR 3 APR 17 MAY 1 MAY 15 MAY 29 JUN 12 JUN 26 JUL 10 JUL 24 AUG 7 AUG 21 SEP 4 SEP 18 OCT 2 OCT 16 OCT 30
OCT 10 OCT 24 NOV 7 NOV 21 DEC 5 DEC 19 JAN 2 JAN 16 JAN 30 FEB 13 FEB 27 MAR 13 MAR 27 APR 10 APR 24 MAY 8 MAY 22 JUN 5 JUN 19 JUL 2 JUL 17 JUL 31 AUG 14 AUG 28 SEP 11 SEP 25 OCT 9 OCT 23 NOV 6
Closed Goal
Action W Goal
Total Open Items
Closed
Resolved
Confirm-W
Confirm-N
Audit N
Action N
Action W
Others

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	JANUARY 30, 1997	NAME:	JIM WINTERS
TO:	DIANE JACKSON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

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

COMMENTS:

DIANE

THIS MARKUP SHOULD RESOLVE OPEN ITEM 586 AS WE DISCUSSED TODAY.

IT WILL BE INCLUDED IN SSAR REVISION 11 UNLESS WE HEAR FROM YOU. THIS SHOULD COMPLETE ALL WESTINGHOUSE ACTIONS ON SECTION 3.5

cc: LINDGREN
MCINTYRE
CUMMINS
REN-UITTRS
WINTERS
HUTCHINS
JEANNE EVANS.

Jim Winters

- Hydrogen gas pressure
- Hydrogen gas purity
- Generator winding overtemperature
- Generator ampere, voltage, and power

Additional generator protective devices are listed in Table 10.2-3.

10.2.6 Combined License Information on Turbine Maintenance and Inspection

The Combined License holder will implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis.

10.2.7 References

1. WSTG-4-P, Proprietary and WSTG-4-NP, Nonproprietary, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines," October 1984.

2. WCAP-11525, Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency, 1987.

3. NUREG-1275, Vol. 11, Operating Experience Feedback Report - Turbine-Generator Overspeed Protection Systems, Commercial Power Reactors, H. L. Ornstein, Nuclear Regulatory Commission, April 1995.

submit to the staff for review and approval within 3 years of obtaining a Combined License, and then

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	1/30/97	NAME:	JOHN BUTLER
TO:	TOM KENYON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office:
COMPANY:	NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 2

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

COMMENTS:

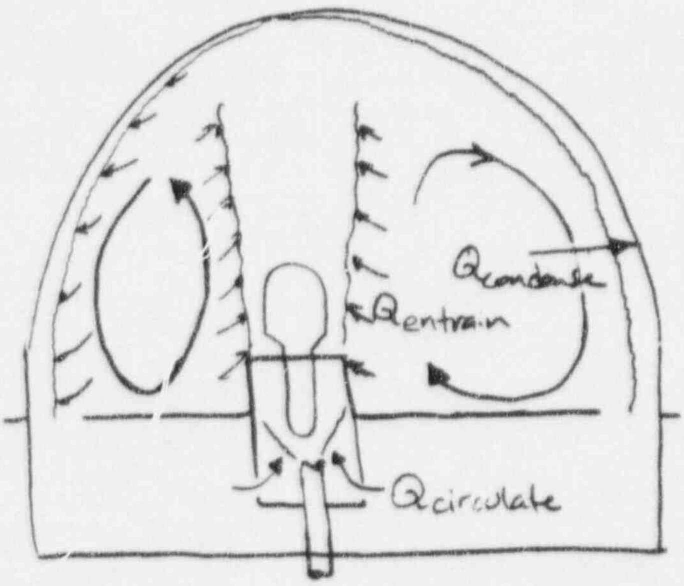
Tom,

A COUPLE OF PAGES TO ASSIST DISCUSSION ON WELL MIXED CONT. FOR AEROSOL REMOVAL

JCB

- Stratification with Entrainment into Plume & Wall Layer

Quasi-steady (like LST plateau and AP600 @ t ≳ 500s.)



Q_{steam} = constant ($\frac{ft^3}{sec}$)

Quasi-steady: $Q_{steam} = Q_{condense}$

For AP600 height,

$$Q_{entrain} \approx 10\ Q_{steam}$$

Any noncondensibles released are mixed with large volume of entrained air in rising plume. No mechanism exists to separate (eg Air+H_2) over time frames of interest.

As long as there is condensation on inner surface, downward circulation of wall layer will prevent stagnant pocket under dome.

External application of PCS water maintains dome at cool temperature before inner surface thermally saturates.

A quasi-steady axial gradient of steam will exist due to physics of rising plume and downward wall layer.

- Gradient can be quantified based on LST
- There is no stagnant region, entire cavity participates

Evidence from literature confirms that as long as there is cooling from the peak of a cavity, there will be downflow and circulation.

Therefore, it is appropriate to assume well-mixed volume in the containment to evaluate aerosol removal.

To: Huffman
From: "John C. Butler" <butlerjc@wcsmail.com>
Subject: Support Information for February 4th Meeting
Cc: McIntyre,VijukRon,CUMMINS.E.X,VijukBob,Winters
Bcc:
X-Attachments: C:\EUDORA\ATTACH\2563P.WPF; C:\EUDORA\ATTACH\DAYS7_3.WPF;

Bill,

Attached are two summary writeups concerning the changes to address the Post-72 hour and Spent Fuel Pool Cooling changes. It is hoped that these writeups will be useful to focus the discussions at the February 4th meeting. Also, these should allow you to assess which review areas may have an interest in attending the meeting.

I will place these in the folder that we maintain for faxes so that this will be docketed in a future letter.

AP600 POST-72 HOURS DESIGN CHANGES

In SECY-96-128, the NRC staff stated that local communities struggling with disaster response should not be given the additional burden of providing for nuclear power safety. In addition, the staff expressed concerned that equipment not under the plant operator's control may be susceptible to damage from environmental conditions. The staff recommended "the Commission approve the position that the site be capable of sustaining all design basis events with onsite equipment and supplies for the long term. After 7 days, replenishment of consumables such as diesel fuel oil from offsite suppliers can be credited. The equipment required after 72 hours need not be in automatic standby response mode, but must be readily available for connection and be protected from natural phenomena including seismic events (per GDC 2)."

Westinghouse has developed design changes to meet the staff position. A summary of the Westinghouse approach is provided below.

POST-72 HOUR CRITERIA

1. Long Term Functions — The functions required to support long term safety of the plant include the following:

   - Core cooling and reactivity control
   - Containment cooling and ultimate heat sink
   - Main control room (MCR) habitability
   - Post-accident monitoring
   - Spent fuel pool cooling

2. Protection from natural phenomena — To address the staff's concern about local communities struggling with disaster response, the onsite equipment is protected against those natural phenomena which have the potential for damage to a wide-spread area. Specifically, the equipment is protected against earthquakes, 145 mile-per-hour wind, and floods as required by site conditions. The 145 mile-per-hour design wind is approximately the 500-year recurrence wind for the worst-case contiguous US site and covers a hurricane similar to Andrew. Tornado hazards are not considered as a design basis since tornados only affect a relatively small area and do not affect the capability of the larger surrounding community to provide assistance and their probability of occurrence is very low.

3. Offsite Assistance — Equipment provided to perform these long term functions will be stored onsite. Consumables used by this equipment for 7 days are stored onsite. After 7 days, supplies of consumables (diesel fuel, water, etc.) may be from offsite sources.

4. Temporary equipment	Provisions are retained in the design for a safety-related connections for transportable equipment for passive containment cooling makeup and for safety-related electrical connections to supply power to the regulating transformers. This retains the previous capability of providing essential safety support functions from offsite equipment. This capability plus the fact that sufficient time is available for equipment repair justifies the non-seismic, non-Class 1E, safety Class D classification.
5. Equipment classification	The onsite equipment provided for post-72 hours is at least classified as non-seismic, Equipment Class D and is protected from natural events as described above. The equipment is analyzed or evaluated to show that it will withstand an SSE.
6. Building Design	The buildings that contain this equipment are at least classified as AP600 Class D and seismic Category 2. They will be analyzed to show they will withstand an SSE and a 145 mile-per-hour wind.
7. Operator Actions	Control of equipment required for post-72 hours is local at the components because of the long time before this equipment is required to operate.
8. Failure Tolerance	Redundancy is provided for active components. Separation of redundant components is not provided.
9. Startup Testing	The functions provided by this equipment will be tested during plant startup testing.
10. ITAACs	The functions provided by this equipment will be covered by Tier I descriptions and ITAACs.
11. Inservice Testing	System level ISTs will be defined to show system operation every 10 years.
12. Availability Controls	Operating procedures will be provided to require this equipment to be available during specified plant operating modes, to operate the equipment on a specified frequency to show that it is available and to return it to operation if it is found to be unavailable.
13. RAP	The equipment provided to perform these functions will be included in the RAP program.

DESIGN FEATURES

1. Post-accident monitoring

Two ancillary ac diesel generators (15 kW each) have been added in the south-east corner of the annex building at elevation 107'-2" to provide ac power for post-accident monitoring of variables identified as Class 1E post-accident monitoring in SSAR Table 7.5-1 and for MCR lighting. In addition, the ancillary generators provide power to the passive containment cooling system recirculation pumps to refill the passive containment cooling water storage tank (PCCWST), and thus the spent fuel pool, when all other sources of power are not available (See items 2 and 4 below.) The ancillary generators are not needed for post-accident monitoring or lighting for the first 72 hours following a loss of all ac and are not needed for refilling the PCCWST or spent fuel pool for the first 7 days following a loss of all ac. The ancillary generators are of a rugged design; therefore, they have a high probability of surviving a seismic event.

The structure housing the ancillary generators is designed so that the generators are protected from earthquakes to the SSE level and 145 mile-per-hour wind (hurricanes). The ancillary generators are located at elevation 107'-2" and therefore above the site maximum flood level of elevation 100'.

Fuel storage sufficient for 4 days of Class 1E post-accident monitoring and MCR lighting is located in the same room as the generators. The tank is connected to the diesel generators through manual, normally-closed valves. The fuel tank, piping, and valves are analyzed to show they will withstand an SSE. The tank includes provisions for venting to the outside atmosphere and for refilling from a truck or other mobile source of fuel. The room containing the ancillary diesel generators and the fuel tank is separated from the rest of the annex building by a three-hour rated fire barrier. Local indication of fuel level is provided for use during operation of the ancillary diesel generators. The tank is insulated and provided with two 1.25 kW heaters to maintain the fuel above the oil cloud point. Fuel oil lines from the tank to the diesels are also insulated.

Diesel generator control is local/manual from a control panel integral with the diesel package. The control panel includes provision for monitoring generator output voltage, current, and frequency (shaft speed). Provision for monitoring engine temperature and lubricating oil pressure is also provided at the control panel.

The diesel starting system will be a 12 Vdc automotive-type with dc starter motor, lead-acid battery, and float charger connected to the plant main ac power system. The battery will be kept in a continuous charge state by the float charger as long as the normal ac power source is available. A voltmeter for monitoring battery terminal voltage is provided on the local control panel.

Each ancillary generator output is connected to a distribution panel located in the same room as the generators. Each distribution panel has an incoming circuit breaker and outgoing feeder circuit breakers. The outgoing feeder circuit breakers are connected to cables which are routed to the divisions B and C voltage regulating transformers and to the passive containment cooling system recirculation pumps. Each distribution panel has the following outgoing connections:

- Connection for one Class 1E voltage regulating transformer to power the Class 1E post-accident monitoring loads and the normal and emergency lighting in the MCR.

- Connection for one recirculation pump for refilling PCCWST. The PCCWST can then be used to refill the spent fuel pool by gravity flow.

- Connection for local loads to support operation of the ancillary generator (lighting, fuel tank heater).

Room heating and cooling during normal plant operations will be by air supplied from the existing annex building HVAC system. A normal ventilation exhaust fan is added to meet building code requirements and NFPA 30 (for diesel room and storage tank ventilation).

The diesel-generators are located near a large doorway to the outside. When the diesels are operated, the outside door is opened, as required to maintain acceptable room temperature, to essentially make the room an outdoor area during diesel operation. Radiators and engine-driven fans discharge air out the door.

Fire protection in the room containing the ancillary diesel generators and their fuel tank is provided by an automatic dry pipe sprinkler system, hose stations, and portable fire extinguishers.

2. Containment cooling and ultimate heat sink	Changes to the containment cooling system include addition of water volume to the tank to provide containment cooling from 72 hours to 7 days, modifications to the standpipes and orifices to provide flow for 7 days, and changes to the recirculation system to add a second recirculation pump and allow the recirculation pumps to provide the motive force to refill the PCCWST from a mobile water source using power from the ancillary diesel generators.
3. Main control room habitability	Changes to the main control room habitability system include revisions to eliminate the need for portable cooling/ventilation units. In the post-72 hour period natural circulation of air with adjacent building areas will be utilized. The MCR access door will be opened after 72 hours. The expected control room operator doses are expected to be acceptable.

The basis for permitting the MCR doors to be opened at 72 hours into a LOCA is that the activity releases at that time would be low enough

such that there would be little benefit to maintaining strict isolation of the MCR. Of the original iodine inventory released to the containment, less than 0.2% would remain airborne and available for release to the environment from containment leakage. Additionally, credit is taken for lower X/Q values by this time which reduces the impact of releases on the calculated doses.

4. Spent fuel pool cooling	Additional safety-related water sources are provided for spent fuel cooling for at least 7 days. Piping and valves have been added to allow gravity-driven makeup water for the spent fuel pool to be provided from the PCCWST and the cask washdown pit.
5. Core cooling and reactivity control	Based on the anticipated containment leakage rate, water to containment for makeup of inventory losses due to containment leakage is not required for more than 30 days.

SUMMARY

The proposed AP600 plant design modifications completely address the NRC staff position which was endorsed by the commission. No equipment is required to be brought to the site and no water or consumable is required for at least 7 days. Additional details can be provided in the meeting proposed for February 4, 1997.

Spent Fuel Pool Design Changes

In SECY-96-128, on the Spent Fuel Pool Cooling System issue, the staff indicated that the AP600 design for spent fuel pool cooling does not conform to SRP guidance nor GDC 2 requirements, and that additional onsite capability to remove decay heat from the spent fuel pool should be provided.

Westinghouse is implementing design changes to meet the staff position, and to provide for full core offload as a normal refueling practice. A summary of the Westinghouse approach is provided below.

Spent Fuel Pool Criteria

1.	Safety-Related Spent Fuel Cooling	The design provides for on-site safety related inventory and makeup capability to keep the spent fuel covered for at least 7 days following a design basis event. Makeup after 7 days is provided as discussed in Post 72 Hour Design Changes. A safety related makeup path is retained from the PCS tank to the spent fuel pool, with makeup to PCS tank using onsite equipment and water from offsite.
2.	SRP Guidance	The design meets SRP guidance by providing Seismic I makeup water system, sources, and fuel pool building, and by meeting 10CFR20 offsite dose limits without credit for ventilation and filtration systems.
3.	GDC 2	The design meets GDC 2 by locating the spent fuel pool and the makeup water system and sources on the nuclear island which is designed to protect these features from the effects of natural phenomena including earthquakes, floods, tornadoes and hurricanes.
4.	Normal Heat Load	The design of the non-safety related spent fuel pool cooling system is based on full core offload as a normal heat load.

Design Features

1. Increased Safety Related Inventory and Makeup Capability — Piping connections are added from the PCS water storage tank and from the cask washdown pit to the spent fuel pool for gravity driven makeup to the spent fuel pool. These safety-related water sources and connections provide sufficient inventory to keep the spent fuel covered for at least seven days for design basis event scenarios.

2. Increased Design Heat Removal Capability — Piping connections are added between the normal residual heat removal system (RNS) and the spent fuel pool for additional cooling capacity during a full core offload. This provides the capacity and redundancy to satisfy SRP and URD pool temperature criteria with full core offload as the normal heat load case. The RNS piping is safety Class C and Seismic Class 1.

3. Post Accident Monitoring — A third safety-related level instrument is added to the spent fuel pool to provide additional post accident monitoring reliability for spent fuel pool level.

Summary:

The AP600 spent fuel cooling system meets GDC 2, GDC 4 and the alternate criteria of Section 9.1.3 of the standard review plan. Safety related makeup is provided to the pool with safety related on site source capability in excess of seven days. No ventilation system is required to meet 10CFR20 dose limits.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	January 30, 1997	NAME:	Jim Winters
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WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:
Bill

Here are the markups for Chapter 8 in partial satisfaction of open item 4615. It will be in Revision 11 of the SSAR unless we hear from you.

Jim Winters

cc: Lindgren
McIntyre
Cummins
Ron Vijuk
Winters
Hayes
Jeanne Evans.

operation and investment protection and to the hydrogen igniters located inside containment. Igniters Operation of the non-Class 1E dc and UPS system is not required for nuclear safety. See subsection 8.3.2.1.2.

The batteries for the Class 1E and non-Class 1E dc and UPS systems are sized in accordance with IEEE 485 (Reference 6). The operating voltage range of the batteries is 105 to 140 Vdc. The maximum equalizing charge voltage for batteries is 140 Vdc. The nominal system voltage is 125 Vdc.

8.3.2.1.1 Class 1E DC and UPS System

8.3.2.1.1.1 Class 1E DC Distribution

The Class 1E dc distribution is in compliance with applicable General Design Criteria, IEEE standards, and Regulatory Guides listed in subsection 8.1.4.3. The scope of compliance encompasses physical separation, electrical isolation, equipment qualification, effects of single active component failure, capacity of battery and battery charger, instrumentation and protective devices, and surveillance test requirements. The Class 1E dc components are housed in seismic Category I structures. For system configuration and equipment rating, see Class 1E dc one-line diagram, Figure 8.3.2-1. Nominal ratings of major Class 1E dc equipment are listed in Table 8.3.2-5.

There are four independent, Class 1E 125 Vdc divisions, A, B, C, and D. Divisions A and D are each comprising one battery bank, one switchboard, and one battery charger. The battery bank is connected to Class 1E dc switchboard through a set of fuses and a disconnect switch. Divisions B and C are each composed of two battery banks, two switchboards, and two battery chargers. The first battery bank in the four divisions, designated as 24-hour battery bank, provides power to the loads required for the first 24 hours following an event of loss of all ac power sources concurrent with a design basis accident (DBA). The second battery bank in divisions B and C, designated as 72-hour battery bank, is used for those loads requiring power for 72 hours following the same event. Each switchboard connected with a 24-hour battery bank supplies power to an inverter, a 125 Vdc distribution panel, and a 125 Vdc motor control center. Each switchboard connected with a 72 hour battery bank supplies power to an inverter. No load shedding or load management program is needed to maintain power during the required safety actuation periods.

A single spare battery bank with a spare battery charger is provided for the Class 1E dc and UPS system. In the case of a failure or unavailability of the normal battery bank and the battery charger, permanently installed cable connections allow the spare to be connected to the affected bus by plug-in locking type disconnect along with kirk-key interlock switches. The plug-in locking type disconnect and kirk-key interlock switches permit connection of only one battery bank and battery charger at a time so that the independence of each battery division is preserved. The spare battery and the battery charger can also be utilized as a substitute when offline testing, maintenance and equalization of an operational battery bank is desired.

The Class 1E dc motor control centers operate at 125 Vdc nominal two wire, ungrounded system. The dc motor control centers provide branch circuit protection for the dc motor-operated valves. Motor-operated valves are protected by thermal overload devices in accordance with Regulatory Guide 1.106. Motor overload condition is annunciated in the main control room. The loads fed from the motor control centers are protected against a short-circuit fault by fusible disconnect switches. Reduced-voltage motor controllers limit the starting current to approximately 250 percent of rated current for motors equal to or larger than 5 HP.

The Class 1E dc distribution panels provide power distribution and tripping capability between the 125 Vdc power sources and the assigned safeguard loads indicated on Figure 8.3.2-1.

8.3.2.1.1.2 Class 1E Uninterruptible Power Supplies

The Class 1E UPS provides power at 208 Y/120 Vac to four independent divisions of Class 1E instrument and control power buses. Divisions A and D each consist of one Class 1E inverter associated with an instrument and control distribution panel and a backup voltage regulating transformer with a distribution panel. The inverter is powered from the respective 24-hour battery bank switchboard. Divisions B and C each consist of two inverters, two instrument and control distribution panels, and a voltage regulating transformer with a distribution panel. One inverter is powered by the 24-hour battery bank switchboard and the other by the 72-hour battery bank switchboard. For system configuration and equipment rating, see Figures 8.3.2-1 and 8.3.2-2. The nominal ratings of the Class 1E inverters and the voltage regulating transformers are listed in Table 8.3.2-5. Under normal operation, the Class 1E inverters receive power from the associated battery bank. If an inverter is inoperable or the Class 1E 125 Vdc input to the inverter is unavailable, the power is transferred automatically to the backup ac source by a static transfer switch featuring a make-before-break contact arrangement. The backup power is received from the diesel generator backed non-Class 1E 480 Vac bus through the Class 1E voltage regulating transformer. In addition, a manual mechanical bypass switch is provided to allow connection of backup power source when the inverter is removed from service for maintenance.

In order to supply power during the post-72-hour period following a design basis accident, provisions are made to connect a transportable ac generator to the Class 1E voltage regulating transformers (divisions B and C only). This powers the post accident monitoring systems and the normal lighting in the main control room and remote shutdown area. See subsection 8.3.1.1.1 for post-72-hour power distribution details and Section 6.4 and subsection 9.5.3 for post-72-hour room ventilation and lighting details respectively.

8.3.2.1.2 Non-Class 1E DC and UPS System

The non-Class 1E dc and UPS system consists of the electric power supply and distribution equipment that provide dc and uninterruptible ac power to the plant non-Class 1E dc and ac loads that are critical for plant operation and investment protection and to the hydrogen igniters located inside containment. The non-class 1E dc and UPS system is comprised of two subsystems representing two separate power supply trains. The subsystems are located in

Table 8.3.1-1 (Sheet 4 of 5)

ONSITE STANDBY DIESEL GENERATOR ZOS MG02A NOMINAL LOADS

Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW)	Loading Method
53.	360	Non-1E Battery Charger EDS1-DC-1	117 kVA	88	AUTO
54.	360	Non 1E Battery Room A Exhaust Fan	0.5 hp	0.5	AUTO
55.	360	Containment Recirculation Fan A	150 hp	21	AUTO
56.	420	Containment Recirculation Fan D	150 hp	21	AUTO
57.	420	Non-1E Battery Charger EDS3-DC-1	117 kVA	88	AUTO
58.	480	Class 1E Div. A Battery Charger 1	78 kVA	26	AUTO
59.	480	Class 1E Div. C Battery Charger 1	78 kVA	24.5	AUTO
60.	480	Class 1E Div. C Battery Charger 2	78 kVA	15	AUTO
61.	480	Div. A/C Class 1E Battery Room Exhaust Fan A	5 hp	5	AUTO
62.	480	Supplemental Air Filtration System Fan A	15 hp	15	AUTO
63.	480	Supplemental Air Filtration System Electric Heater A	20 kW	20	AUTO
64.	480	Instrument Air Compressor A	200 hp	166	AUTO
65.	540	Backup Group 4A Pressurizer Heaters	246 kW	246	AUTO
~~66.~~	~~--~~	~~Hydrogen Igniters XFMR 1~~	~~10 kVA~~	~~5~~	~~MAN~~
676.	--	CRDM Fan 01A	40 hp	33	MAN
687.	--	CRDM Fan 01B	40 hp	33	MAN
698.	--	Spent Fuel Cooling Pump A	75 hp	62	MAN

Total Diesel Operating Loads (kW)
~~3582.5~~3577.5

Table 8.3.1-2 (Sheet 4 of 5)

ONSITE STANDBY DIESEL GENERATOR ZOS MG02B NOMINAL LOADS

Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW)	Loading Method
52.	360	Non-1E Battery Charger EDS2-DC-1	117 kVA	88	AUTO
53.	360	Non-1E Battery Room B Exhaust Fan 09B	0.5 hp	0.5	AUTO
54.	360	Containment Recirculation Fan B	150 hp	21	AUTO
55.	420	Containment Recirculation Fan C	150 hp	21	AUTO
56.	420	Non-1E Battery Charger EDS4-DC-1	117 kVA	88	AUTO
57.	480	Class 1E Div. B Battery Charger 1	78 kVA	24.5	AUTO
58.	480	Class 1E Div. B Battery Charger 2	78 kVA	15	AUTO
59.	480	Class 1E Div. D Battery Charger 1	78 kVA	26	AUTO
60.	480	Div. A/C Class 1E Battery Room Exhaust Fan B	1.5 hp	1.5	AUTO
61.	480	Supplemental Air Filtration System Fan B	15 hp	15	AUTO
62.	480	Supplemental Air Filtration System Electric Heater B	20 kW	20	AUTO
63.	480	Instrument Air Compressor B	200 hp	166	AUTO
64.	--	Backup Group 4B Pressurizer Heaters	246 kW	246	AUTO
~~65.~~	--	~~Hydrogen Igniters XFMR 2~~	~~10 kVA~~	5	~~MAN~~

Table 8.3.1-2 (Sheet 5 of 5)

ONSITE STANDBY DIESEL GENERATOR ZOS MG02B NOMINAL LOADS

Item No.	Time Seq. (sec)	Event or Load Description	Rating (hp/kW)	Operating Load (kW)	Loading Method
665.	--	CRDM Fan 01C	40 hp	33	MAN
676.	--	CRDM Fan 01D	40 hp	33	MAN
687.	--	Spent Fuel Cooling Pump B	75 hp	62	MAN

Total Diesel Operating Loads (kW)
~~3550~~3545

Notes:

1. Loads listed are for diesel generator ZOS MG02B.
2. Loads identified with automatic (AUTO) loading will be loaded without operator action. Loads identified with manual (MAN) loading will be energized at operator discretion based on system needs. Automatic loads may not be started until there is a system need (for example, the make-up pump may not be started until make-up flow is required.)
3. Time Sequence is counted from the time a diesel generator receives the start signal.
4. The "Operating Load" column shows the load input power requirement from diesel generator.
5. Motor operated valves (MOVs) pertaining to various systems will be energized on closure of the diesel generator breaker. Normally the MOV power requirement is for a very short duration (few seconds), hence the MOV load will not affect the diesel generator capacity rating.
6. On receipt of the diesel generator start signal, the engine accelerates to a set idle speed. Engine operates at the idle speed for a time period to allow bearing oil pressure build up, proper lubrication of the moving parts, and engine warmup. After a set time delay (to be determined based on vendor selection), the engine will ramp up to the rated operating speed.
7. On restoring the power supply to the diesel backed bus ES2 by closing diesel generator incoming breaker, the associated unit substation ECS EK 21 and 22 load center transformers are energized. The transformers draw magnetizing current and the no load losses (approx. 0.3 percent of the rating) from the bus.
8. Only a part of the building lighting load is automatically connected to the diesel generator bus. The remaining lighting load is connected via manual action at the operator's discretion.
9. Load Center ECS EK 23 transformer no load losses and magnetizing current is approximately 0.3 percent of the transformer rating.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
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COMMENTS:

BILL,

ITEM 1078 WAS THE DSER "CATCH ALL" COMMENT TO ENSURE COMPLETION OF RAIs 435.64 and 435.73. ATTACHED ARE THE TWO RAIs AND THE RESULTANT CHANGES TO 8.3.1.1.6 AND 8.3.1.1.7. PER MY INTERPRETATION OF OUR TELECON I CHANGED NRC STATUS FOR 1078 TO ACTION N. CALL IF IT SHOULD BE SOMETHING ELSE.

Jim Winters

cc: LINDGREN
MCINTYRE
WINTERS
HAYES

Question 435.64

Describe the provisions to prevent lightning from initiating fires that could damage safety-related equipment or fire protection equipment.

SSAR 8.3.1.1.7

Response:

The lightning protection system, consisting of air terminals and ground conductors, will be provided for the protection of the containment/shield building, cooling towers, switchyard, and other exposed structures and buildings housing safety-related and fire protection equipment in accordance with Lightning Protection Code NFPA 78-1989. Also, lightning arresters will be provided in each phase of the transmission lines and at the high-voltage terminals of the outdoor transformers. The isophase bus connecting the main generator and the main transformer and the medium-voltage switchgear will be provided with lightning arresters. In addition, surge suppressors will be provided to protect the plant instrumentation and monitoring system from lightning-induced surges in the signal and power cables connected to devices located outside.

Direct-stroke lightning protection for facilities is accomplished by providing a low-impedance path by which the lightning stroke discharge can enter the earth directly. The direct-stroke lightning protection system, consisting of air terminals, interconnecting cables, down conductors to ground, etc., will be provided external to the facility in accordance with the guidelines included in NFPA 78. The system will be connected directly to the station ground to facilitate dissipation of the large current of a direct lightning stroke. The lightning arresters and the surge suppressors connected directly to ground provide a low-impedance path to ground for the surges caused or induced by lightning. Thus, fire or damage to facilities and equipment resulting from a lightning stroke is avoided.

The design of direct-stroke lightning protection and the associated grounding depends on the lightning activity at the plant site and the soil resistivity of the ground. It is site specific and will be described by the combined license applicant.

SS' Revision: NONE

Question 435.73

SSAR 8.3.1.1.6

Address the grounding system and lightning protection features for the AP600.

Response:

The AP600 grounding system will comply with the guidelines provided in IEEE Standard 665-1987, "Guide for Generating Station Grounding." The grounding system consists of the following four subsystems:

- Station grounding grid
- System grounding
- Equipment grounding
- Instrument/computer grounding

The station grounding grid subsystem consists of buried, interconnected bare copper conductors and ground rods (Copperweld) forming a plant ground grid matrix. The subsystem will maintain a uniform ground potential and limit the step-and-touch potentials to safe values under all fault conditions.

The system grounding subsystem provides grounding of the neutral points of the main generator, main stepup transformers, auxiliary transformers, load center transformers, and onsite standby diesel generators. The main and diesel generator neutrals will be grounded through grounding transformers providing high-impedance grounding. The main stepup and load center transformer neutrals will be grounded solidly. The auxiliary (unit and reserve) transformer secondary winding neutrals will be resistance grounded.

The equipment grounding subsystem will provide grounding of the equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, MCCs, and control cabinets with two ground connections to the station ground grid.

The instrument/computer grounding subsystem will provide plant instrument/computer grounding through separate radial grounding systems consisting of isolated instrumentation ground buses and insulated cables. The radial grounding systems will be connected to the station grounding grid at one point only and will be insulated from all other grounding circuits.

For the lightning protection system description, see the response to Q435.64.

The design of the grounding grid system and the lightning protection system depends on the soil resistivity and lightning activity in the area. Therefore, the design of both systems is site-specific and is the responsibility of the combined license applicant.

SSAR Revision: NONE

field cables of the circuits so that the fault or overload currents are interrupted by the protective devices prior to a potential failure of a penetration. Penetrations are protected for the full range of currents up to the maximum short circuit current available.

Primary and backup protective devices protecting Class 1E circuits are Class 1E in accordance with IEEE 741 (Reference 10). Primary and backup protective devices protecting non-Class 1E circuits are non-Class 1E.

Penetration overcurrent protection coordination curves are generated based on the protection requirements specified by the penetration equipment manufacturer. When necessary, penetrations are protected for instantaneous overcurrent by current limiting devices such as current-limiting fuses, current-limiting breakers, or reactors.

8.3.1.1.6 Grounding System

The AP600 grounding system will comply with the guidelines provided in IEEE Standard 665-1987, "Guide for Generating Station Grounding." The grounding system consists of the following four subsystems:

- Station grounding grid
- System grounding
- Equipment grounding
- Instrument/computer grounding

The station grounding grid subsystem consists of buried, interconnected bare copper conductors and ground rods (Copperweld) forming a plant ground grid matrix. The subsystem will maintain a uniform ground potential and limit the step-and-touch potentials to safe values under all fault conditions.

The system grounding subsystem provides grounding of the neutral points of the main generator, main stepup transformers, auxiliary transformers, load center transformers, and onsite standby diesel generators. The main and diesel generator neutrals will be grounded through grounding transformers providing high-impedance grounding. The main stepup and load center transformer neutrals will be grounded solidly. The auxiliary (unit and reserve) transformer secondary winding neutrals will be resistance grounded.

The equipment grounding subsystem provides grounding of the equipment enclosures, metal structures, metallic tanks, ground bus of switchgear assemblies, load centers, MCCs, and control cabinets with two ground connections to the station ground grid.

The instrument/computer grounding subsystem provides plant instrument/computer grounding through separate radial grounding systems consisting of isolated instrumentation ground buses and insulated cables. The radial grounding systems are connected to the station grounding grid at one point only and are insulated from all other grounding circuits.

The design of the grounding grid system and the lightning protection system depends on the soil resistivity and lightning activity in the area. Therefore, the design of both systems is site-specific and is the responsibility of the combined license applicant.

8.3.1.1.7 Lightning Protection

The lightning protection system, consisting of air terminals and ground conductors, will be provided for the protection of exposed structures and buildings housing safety-related and fire protection equipment in accordance with Lightning Protection Code NFPA 78-1989. Also, lightning arresters are provided in each phase of the transmission lines and at the high-voltage terminals of the outdoor transformers. The isophase bus connecting the main generator and the main transformer and the medium-voltage switchgear is provided with lightning arresters. In addition, surge suppressors are provided to protect the plant instrumentation and monitoring system from lightning-induced surges in the signal and power cables connected to devices located outside.

Direct-stroke lightning protection for facilities is accomplished by providing a low-impedance path by which the lightning stroke discharge can enter the earth directly. The direct-stroke lightning protection system, consisting of air terminals, interconnecting cables, and down conductors to ground, are provided external to the facility in accordance with the guidelines included in NFPA 78. The system is connected directly to the station ground to facilitate dissipation of the large current of a direct lightning stroke. The lightning arresters and the surge suppressors connected directly to ground provide a low-impedance path to ground for the surges caused or induced by lightning. Thus, fire or damage to facilities and equipment resulting from a lightning stroke is avoided.

The design of direct-stroke lightning protection and the associated grounding depends on the lightning activity at the plant site and the soil resistivity of the ground. It is site specific and is the responsibility of the Combined License applicant.

8.3.1.2 Analysis

The ac power system is non-Class 1E and is not required for safe shutdown. Compliance with existing regulatory guides and General Design Criteria is covered in Table 8.1-1 of Section 8.1.

8.3.1.3 Raceway/Cable

8.3.1.3.1 General

The raceway system for non-Class 1E ac circuits complies with IEEE 422 (Reference 3) in respect to installation and support of cable runs between electrical equipment including physical protection. Raceway systems consist primarily of cable tray and wireway.

Westinghouse

FAX COVER SHEET

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COMMENTS:

Diane, The attached markups identify a revision to the Applications report to address a telecon action item. These markups will be included in a revision to the Application Report.

JB

cc: Woodcock
RAJIG

detailed discussion of how these features are used in containment modeling is given in Section 4-1.

The GOTHIC code also contains the options to model a large number of structures and components. These include, but are not limited to, heated and unheated conductors, pumps, fans, a variety of heat exchangers, and ice condensers. These components can be coupled to represent the various systems found in any typical containment.

The GOTHIC containment analysis code was modified by Westinghouse to include mechanistic convection heat and mass transfer correlations, a liquid film tracking model, a one-dimensional wall conduction model, and wall-to-wall radiant heat transfer to model heat removal by the PCS. The code with modifications, is called Westinghouse-GOTHIC and is abbreviated as WGOTHIC.

The three programs that make up the WGOTHIC code — solver version 4.0, preprocessor version number 4.0, and postprocessor version 4.0 — are based on GOTHIC code version 4.0 as described in NAI 8907-06 (Ref. 3). The preprocessor (input handler) and solver (numerical solution) programs contain the code modifications to incorporate the PCS models. Changes were made to the preprocessor program to assist the user in setting up model input. These changes were verified by hand. Changes to the solver program are described in the following sections.

3.3 THE WGOTHIC CLIME MODEL

A solution technique that includes wall-to-wall radiation necessitates a close coupling between the involved walls. This coupling is accomplished by assigning boundaries that define the portions of the various walls that radiate to each other. In keeping with the GOTHIC formulation (Refs. 1 and 2), that considers conductors or heat sinks to be energy source (or sink) terms, the code modifications made to include wall-to-wall radiant heat transfer can be thought of as the addition of a special type of conductor group. This new conductor group consists of a set of walls that radiate to each other and interface with GOTHIC fluid cells through mass and energy source terms. To distinguish this type of conductor from the existing GOTHIC terminology, the term *clime*, meaning a *region*, is used.

An AP600 clime is a horizontal slice consisting of: the heat and mass transfer source terms from the vessel volume to the wall, conduction through the vessel wall; heat and mass transfer source terms from the vessel wall to the air flow channel; radiation from the vessel wall to the baffle wall; heat transfer source terms from the baffle wall to the air flow channel; conduction through the baffle wall; radiation and convection heat transfer from the baffle wall to the environment. typical a ~~A simplified two-conductor~~ clime is depicted in Figure 3-3. The vessel volume, air flow channel volume, and environment volume are separate computational cells (fluid volumes) in WGOTHIC. The vessel and baffle wall are one-dimensional conductors representing solid walls

showing heat and mass source terms.

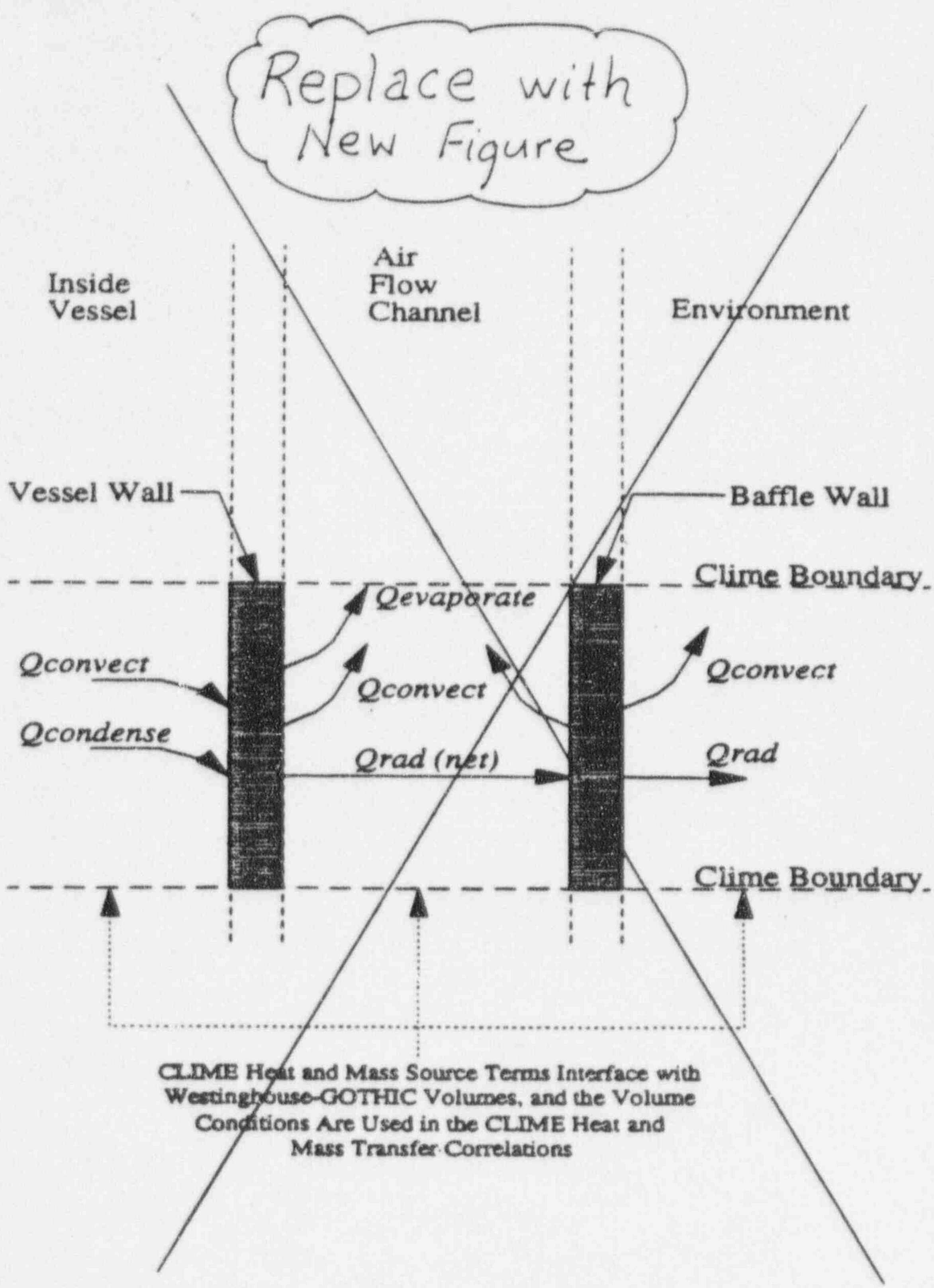

Figure 3-3 Westinghouse-GOTHIC Clime Wall Source Term Models

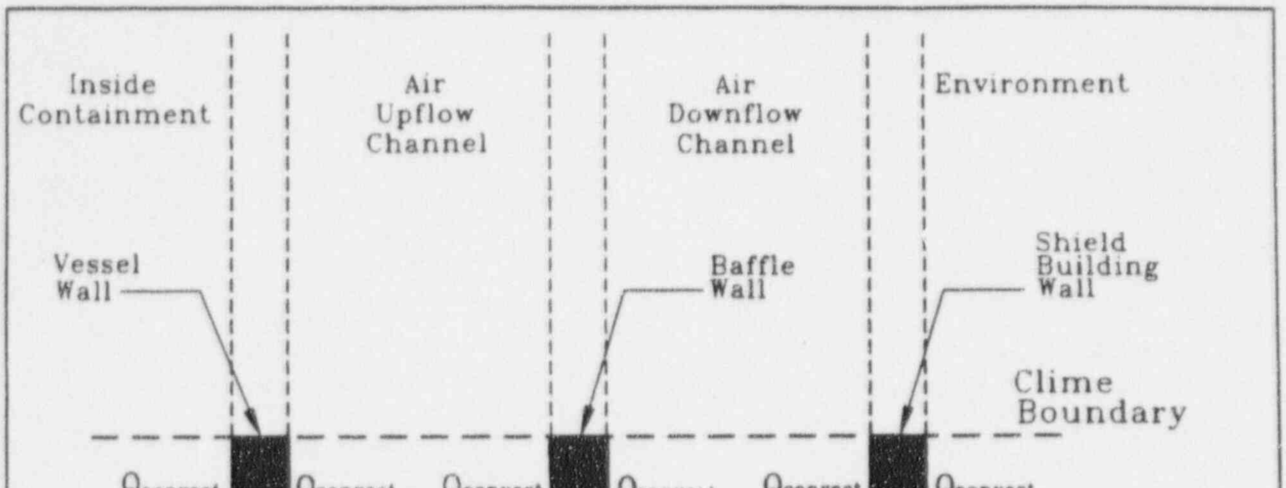

Figure 3-3 Climes Provide the Thermal Link From Inside Containment to Enviornment

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THESE ITEMS INCLUDE FIRST AND THIRD BULLETS FOR 309 IN JOHN'S NOTES FROM 11/19

LETTER, 12.b. OF YOUR 12/6 KEY ISSUES LETTER, (a) and (b) FOR 309 IN YOUR 1/3/97

LETTER AND (a) and (b) for 309 in YOUR 1/8/97 LETTER. THIS WILL BE IN

SSAR REVISION 11 UNLESS WE HEAR FROM YOU.

Jim Winters

CC: LINDGREN
MCINTYRE
CUMMINS
RON VIJUK
HUTCHINGS
BARRY SLOANE
JEANNE EVANS

- Fire barrier separation is not provided within the remote shutdown workstation fire area because the remote shutdown workstation is not required for safe shutdown unless a fire requires evacuation of the main control room.

Complete — necessary to define a fire area — which define fire zones

- Fire barrier separation is not provided within the primary containment fire area (including the middle and upper annulus zones of the shield building) because of the need to satisfy other design requirements, such as allowing for pressure equalization within the containment following a high-energy line break. Fire protection features within the containment fire area provide confidence that one train of safe shutdown equipment will remain undamaged following a fire. The quantity of combustible materials is minimized. The use of canned reactor coolant pump motors has eliminated the need for an oil lubrication system. Redundant trains of safe shutdown components are separated whenever possible by existing structural walls, or by distance. The fire protection system provides appropriate fire detection and suppression capabilities.

in any fire zone

If cables of a safety-related division must pass through a fire zone of an unrelated division, they are protected by fire barriers.

Outside of the primary containment and the main control room, the arrangement of plant equipment and routing of cable are such that safe shutdown can be achieved with all components (except those protected by 3-hour fire barriers) in any one fire area rendered inoperable by fire.

Openings and penetrations through fire barriers are protected in accordance with the guidelines of BTP CMEB 9.5-1.

The fire protection analysis contains a description of plant fire areas, fire zones, fire barriers, and the protection of fire barrier openings, as well as a description of the separation between redundant safe shutdown components.

Electrical Cable Design, Routing, and Separation

Electrical cable (including fiber optic cable) and methods of raceway construction are selected in accordance with BTP CMEB 9.5-1. Metal cable trays are used. Rigid metal conduit or metal raceways are used for cable runs not embedded in concrete or buried underground. Flexible metallic tubing is used in short lengths for equipment connections.

The insulating and jacketing material for electrical cables are selected to meet the fire and flame test requirements of IEEE Standard 383 (Reference 3).

The design, routing, and separation of cable and raceways are further described in Section 8.3.

Control of Combustible Materials

The plant is constructed of noncombustible materials to the extent practicable. The selection of construction materials and the control of combustible materials are in accordance with BTP CMEB 9.5-1 and NFPA 803.

Any damage which the fire is capable of causing is assumed to occur immediately. No credit is taken for proper operation of equipment or proper positioning of valves which are not protected from the effects of a postulated fire.

INSERT 1

Zone of Influence

INSERT 2

A postulated fire does not exceed the boundary of the fire area. For fire areas outside the main control room, remote shutdown workstation, and containment fire areas, all equipment in any one fire area is assumed to be rendered inoperable by the fire and re-entry into the fire area for repairs and operator actions is assumed to be impossible. However, no credit is taken for complete fire damage in cases in which complete damage is beneficial and partial damage is not. Chases for electrical cables, piping or ducts that pass through the fire area but are separated from it by 3-hour fire barriers are outside ~~the zone of influence for~~ that fire area.

~~Inside the containment fire area, potential fire damage is evaluated by fire zone~~. All equipment in any one fire zone is assumed to be rendered inoperable by the fire unless the fire protection analysis demonstrates otherwise. Class 1E electrical cables that are located in or pass through the fire zone but are separated from it by a 3-hour fire barrier are outside ~~the zone of influence for~~ that fire zone.

INSERT 3

INSERT 4

Independence of Affected Fire Areas

Only systems, components, and circuits free of fire damage are credited for achieving safe shutdown for a given fire. Systems, components, and circuits outside the zone of influence are considered free of fire damage if the effects of the fire do not prevent them from performing their required safe shutdown functions.

Event Assumptions

Plant accidents and severe natural phenomena are not assumed to occur concurrently with a postulated fire. Furthermore, a concurrent single active component failure (independent of the fire) is not assumed.

Offsite Power

A loss of offsite power is assumed concurrent with the postulated fire only when the safe shutdown evaluation indicates the fire could initiate the loss of offsite power.

Availability of Nonsafety-Related Systems

Only safety-related components and systems are assumed to be available to perform safe shutdown functions. (This is more stringent than required by BTP CMEB 9.5-1.) Fire protection and smoke control systems are assumed to function as designed to detect and mitigate the effects of the fire.

INSERT 1

Fire Barriers

As described in subsection 9.5.1.2.1.1, non-combustible fire barriers are provided in accordance with BTP CMEB 9.5-1 and NFPA 803 (Reference 2). The equivalent fire barrier ratings are shown in Figures 9A-1 through 9A-5. Fire barriers or equivalent structural features form the boundaries of fire areas. For most fire zones in containment, fire barriers separate redundant equipment. If cables of a safety-related division must pass through or adjacent to a fire area or fire zone of an unrelated division, they are protected by fire barriers.

INSERT 2

Fire Areas

Fire areas are three dimensional spaces designed to contain a fire that may exist within them. They are surrounded by fire barriers, structure equivalent to fire barriers, fire barrier penetration protection, and other devices, such as those within the heating and air conditioning ducts, that isolate a fire to within the fire area.

INSERT 3

Outside containment, zone of influence is not defined. A fire outside containment is assumed to affect its entire fire area. Inside the containment fire area, the zone of influence is defined as the entire fire zone containing the fire.

In containment, fire zones are usually bounded by physical structures equivalent to a 3-hour fire barrier. In some cases, other fire protection features apply, such as distance or lack of fuel. For example, fire zone 1100 AF 11300A has no physical barrier between it and fire zone 1100 AF 11300B. This is due to the fact that all combustibles are at the extreme ends of these fire zones and are separated by more than 40 feet. There will be no communication of a fire from one fire zone to the other. Other examples include fire zones 1100 AF 11301 and 1100 AF 11302 which are open at their tops into fire zone 1100 AF 11500. Fire zone 1100 AF 11500 is the open upper containment. With no fuel sources over fire zones 1100 AF 11301 and 1100 AF 11302, there will be no fire communication between these zones and fire zone 1100 AF 11500.

INSERT 4

Fire Zones

Fire zones are three dimensional spaces within fire areas. Fire zones are identified uniquely to indicate that they have fire protection features or attributes different than other fire zones in a given area. For example, this difference may be due to different sprinkler coverage due to different fuel loadings. In containment, fire zones are identified to establish "zones of influence".

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Jim Winters

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monitoring, and therefore requires no nuclear safety evaluation. Redundant safety-related isolation dampers are provided in the supply, return, and exhaust ducts penetrating the main control room. Therefore, there are no single active failures which would prevent isolation of the main control room envelope. Redundant main control room supply air radiation monitors are provided. The nuclear island nonradioactive ventilation system is designed so that safety-related systems, structures, or components are not damaged as a result of a seismic event.

9.4.1.4 Tests and Inspections

within a tolerance of ± 10 percent of design flow rate

The nuclear island nonradioactive ventilation system is designed to permit periodic inspection of system components. Each component is inspected prior to installation. Components of each system are accessible for periodic inspection during normal plant operation. A system air balance test and adjustment to design conditions is conducted in the course of the plant preoperational test program. Airflow rates are measured and balanced in accordance with the guidelines of SMACNA HVAC systems, Testing, Adjusting and Balancing (Reference 19) except the supplemental air fultration units which are balanced in accordance with the guidelines of ASME N510 (Reference 3). Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability.

The supplemental air filtration unit, HEPA filters, and charcoal adsorbers are field tested in accordance with ASME N510 to verify that these components do not exceed a maximum allowable bypass leakage rate. Used samples of charcoal adsorbent are periodically tested to verify a minimum charcoal efficiency of 90 percent in accordance with Regulatory Guide 1.140, except that test procedures and test frequency are conducted in accordance with ASME N510.

The ductwork for the supplemental air filtration subsystem and portions of the main control room/technical support center HVAC subsystem that maintain the integrity of the main control room/technical support center pressure boundary during conditions of abnormal airborne radioactivity are tested for leak tightness in accordance with ASME N510, Section 6. The remaining supply and return/exhaust ductwork is field tested for ~~leak tightness~~ leakage in accordance with SMACNA HVAC Duct Leakage Test Manual (Reference 18).

9.4.1.5 Instrumentation Applications

The nuclear island nonradioactive ventilation system is controlled by the plant control system except for the main control room isolation dampers, which are controlled by the protection and safety monitoring system. Refer to subsection 7.1.1 for a description of the plant control and plant safety and monitoring systems.

Temperature controllers are provided in the return air ducts to control the room air temperatures within the predetermined ranges. Temperature indication and alarms for the main control room return air, Class 1E electrical room return air, air handling unit supply air, supplemental filtration unit inlet air and charcoal adsorbers are provided to inform plant operators of abnormal temperature conditions.

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Bill,

ADS Roadmap. Formal letter to follow.

JB

AP600 Hydrodynamic Load Roadmap

The approach for resolving the hydrodynamic loading issue for the AP600 consisted of a testing program, and an analytical program. The testing program was performed by ENEA in the Vapore test facility, and consisted of a full scale single sparger submerged in a non-prototypic test tank full of water. The sparger was connected to a steam supply system capable of supplying saturated steam and steam/water two-phase mixture, to simulate different full flow operating conditions for the ADS system. The test program was divided into phase A (steam discharge only) and phase B (both steam and two-phase mixture discharge through the sparger). The first part of phase B testing (B1) was intended to duplicate the design conditions for the sparger discharge in the AP600. The following is a summary of the results from the phase B1 testing:

1. Choked flow occurred at more than one point in the ADS flow path simultaneously, and included at the open ADS valves, at the inlet to the sparger, at the sparger arm inlet from the body, at the sparger arm holes (discharge holes).

2. Sparger operation was smooth with steam and two-phase fluid, and the resultant pressure peaks appeared to be within the expected magnitudes.

3. The sparger arm geometry created a strong mixing current in the quench tank during the blowdown when the tank was subcooled. However, complete mixing to the bottom of the tank did not occur.

4. Blowdown in the fully heated tank (212°F) expelled a significant amount of water from the tank.

5. The pressure pulses measured in the quench tank when the water was hot (212°F) were significantly reduced compared with blowdown in cold water (90°F).

6. No instability in steam condensation was observed at elevated pool temperatures > 179°F to saturation temperature.

7. Air and water clearing loads were not dominant.

8. No significant pressure oscillations due to chugging were observed.

A. Information that documents that the ADS testing performed at VAPORE adequately simulated and provided blowdown conditions that are directly applicable to the AP600 plant is listed below.

- WCAP-14303, "Facility Description Report — AP600 Automatic Depressurization System Phase B1 Tests", AP600 Document Number RCS-T3R-001, Rev. 0, March 1995.

  This document provides a detailed description of the VAPORE facility. It is a good reference document for details on the physical layout, the instrumentation, and components of the VAPORE test facility.

- WCAP-14324, "Final Data Report for ADS Phase B1 Tests", AP600 Document Number RCS-T2R-100, Rev. 0, April 1995.

  This document provides detailed information on the actual performance of the Phase B1 testing, and provides the verified test data. Sections of interest include:

AP600 Hydrodynamic Load Roadmap

- Section 1.2, describes the test matrix

- Section 2.0, pp 2-1 to 2-21, provides a concise description of the test facility.

- Section 4.0 provides reduced test data on each test run including the blowdown flow rate vs. time, the fluid quality vs. time, and the pressure drops through the facility at different times. Table 4.5-1 on p. 4-151 provides a summary of the quasi-steady state pressure and fluid quality upstream of the test facility ADS package, the ADS package delta-P, the pressure and fluid quality as discharged from the sparger, and the flow rate achieved in each test run.

- Appendix C provides a set of selected at plots for each test which includes the pressure pulse time histories for quench tank pressure sensors and their associated power density spectrum.

- WCAP-14305, "AP600 Test Program Test Analysis Report", AP600 Document Number RCS-T2R-110, Rev. 1, June 1995.

  This document analyses the ADS Phase B1 test data to obtain needed information on ADS behavior, pressure drops, flow rates and flow splits through the stages, and two phase multipliers. These results are used to validate the computer models of the ADS in the NOTRUMP and WCOBRA/TRAC computer codes used in SSAR Chapter 15 analyses. This information does not directly apply to the analysis of the ADS/IRWST interaction, but is referenced to illustrate the detailed review performed on the actual test data.

- WCAP-14676, "AP600 Automatic Depressurization System Stage 1, 2 and 3 Cold Flow Test", AP600 Document Number RCS-T2R-020, Rev. 0, July 1996.

  This document presents delta-P data and determines the overall and individual component resistances in the ADS valve package experienced with single-phase cold water flow. This information does not directly apply to the analysis of the ADS/IRWST interaction, but is referenced for completeness.

- WCAP-14727, "AP600 Scaling and PIRT Closure Report", AP600 Document Number PXS-GSR-020, Rev. 0, September 1996.

  - Section 5.0 of this document presents a concise summary of the ADS Phase B1 test and the phenomena important for LOCA type analysis.

  - Section 5.2 presents the ADS Phase B1 test scaling basis for each component and is of interest in documenting the applicability of the data.

  - Section 5.3 discusses the scaling distortions of the ADS facility.

  - Section 5.4 describes any unanticipated phenomena which occurred during the test program.

  - Section 5.5 compares the ADS Phase B1 tests run with the performance of the ADSs during the transient simulated at the SPES-2 and OSU integral systems tests.

AP600 Hydrodynamic Load Roadmap

B) Information describing how the ADS test data on the quench tank pressure pulses was utilized to establish the pressure pulse design criteria for the plant, and how was this information incorporated into the plant IRWST structural design is provided in the following documents. These documents were developed as part of the AP600 FOAKE program but are available for review at the Westinghouse Rockville office or at the Energy Center Site in Monroeville, PA.

- MT01-S3C-012, Rev. 0, AP600 Document, April 1996, "ADS Discharge Investigation and IRWST Hydrodynamic Global Analysis".

  This report documents the selection of pressure loads for the structural analysis of the IRWST. The selection was based on the expected operating conditions of the ADS, being conservatively enveloped by the test data from the B1 testing based on the following parameters: mass flow rate vs. quality, and quality and mass flow rate derivative. Two tests were selected as representative for enveloping expected operating conditions, one representing the frequency spectrum up to 40 hz, the other for higher frequencies. Selected short time periods were then selected from each test and used as pressure time history input loading in the ANSYS model for the IRWST. The displacement time history of critical locations are also documented.

- AP600 letter MIS/FOK0019, Gordon K. Ashley II (SciEnTec) to R. Hundal (W), "Generation of an Acoustic Source Function for the Ansaldo Pressure Trace IRW330", dated October 22, 1996.

  This report evaluated the pressure traces used by the ANSALDO analysis, to determine the extent of test tank influence in the pressure trace used in the analysis. The intent was to develop a pure acoustic pressure source by determining the part of the frequency content of the measured pressure trace which was due to the test tank. The results of this investigation was that the test tank frequency could not be determined with certainty since there was no value for the sonic velocity determined n the test tank. Based on the assumptions made by ANSALDO for sonic speed of 1440 m/s resulting from near rigid test tank walls, the test tank acoustic resonance frequency is approximately 50 hz. However, with a sonic frequency of 872 m/s, based on measurements in the Kraftwerk Union GKM II-M test tank which was expected to have similar wall stiffness to the ANSALDO tank, the test tank frequency will be approximately 30 hz. The measured pressure response in the test tank contained peaks at both of these frequencies. It was, however, concluded that these acoustic resonance peaks included in the source loading used by ANSALDO for the analysis of the IRWST, provide additional conservatism in terms of the source loading.

- WCAP-14766, "The Hydrodynamic Effect of the Automatic Depressurization System (ADS) Discharge on the AP600 RCS," December 1996.

  This report documents a detailed analysis of the steam generator 1, which is located adjacent to the wall of the IRWST and therefore sees the greatest loads from the ADS discharge/sparger operation. The report documents the evaluation based on both monolithic concrete properties and cracked concrete properties, and evaluates the loads transferred to the steam generator via supports, snubbers and bumpers. A linear analysis was performed to evaluate the hypothetical effects of supports without gaps, and a non-linear analysis showed the effects of expected gaps in the supports. The conclusions of this study are as follows:

AP600 Hydrodynamic Load Roadmap

1. The high-frequency, low amplitude input is not sufficient to overcome the inertia of the steam generator, and therefore does not produce significant steam generator movements.

2. There is no bumper/SG interaction if the gap exceed 0.013 inch at one bumper, or 0.005 inch at each bumper.

3. The SG snubbers are not active during an ADS event if the "dead band" for the snubber assembly exceeds 0.007 inch.

4. There is no appreciable amplification of the input motion by the SG or the SG internal structures.

5. The loads and displacements generated by the response to the ADS hydrodynamic loading on the reactor coolant system are less than 10 percent of those produced by the SSE. The combined ADS and SSE loads adds less than 0.3 percent to those produced by the SSE alone, and are not considered significant. Therefore, the ADS generated hydrodynamic loads can be ignored as design basis events.

** TX CONFIRMATION REPORT ** AS OF JAN 31 '97 14:51 PAGE.01

AP600 DESIGN CERT

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	STATUS
01	1/31	14:49	#23:NRC	G3--S	02"37	05	OK

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The diesel generator engine fuel oil system consists of an engine-mounted, engine-driven fuel oil pump that takes fuel from the fuel oil day tank, and pumps through inline oil filters to the engine fuel injectors and a separate recirculation circuit with a fuel oil cooler. The recirculation circuit discharges back to the fuel oil day tank that is maintained at the proper fuel level by the diesel fuel oil storage and transfer system.

The design of the onsite standby diesel generators does not ensure functional operability or maintenance access or support plant recovery following design basis events. Maintenance accessibility is provided consistent with the system nonsafety-related functions and plant availability goals.

The piping and instrumentation diagrams for the onsite standby diesel generator units and the associated subsystems are shown on Figures 8.3.1-4 and 8.3.1-5.

The onsite standby power supply system is shown schematically on one line diagram, Figure 8.3.1-1.

INSERT 8.3-7-1

8.3.1.1.2.2 Generator

Each generator is a direct-shaft driven, air-cooled self ventilated machine. The generator enclosure is open drip-proof type that facilitates free movement of ventilation air The generator component design is in compliance with the NEMA MG-1 (Reference 1) requirements.

Each generator produces its rated power at 4160 V, 60 Hz. Each generator continuous rating is based on supplying the electrical ac loads listed in Tables 8.3.1-1 or 8.3.1-2. The loads shown on Tables 8.3.1-1 and 8.3.1-2 represent a set of nonsafety-related loads which provide shutdown capability using nonsafety-related systems. The generators can also provide power for additional investment protection ac loads. The plant operator would normally provide power to these loads by deenergizing one of those system components that are redundantly supplied by both the diesel generators. The diesel generator design is compatible with the step loading requirements identified in Tables 8.3.1-1 and 8.3.1-2. The generator exciter and voltage regulator systems are capable of providing full voltage control during operating conditions including postulated fault conditions.

Each generator has a set of potential and current transformers for protective relaying and metering purposes.

The following generator protection functions are provided via relays that are mounted on the local generator control panel:

> Differential (87), overcurrent (50/51), reverse power (32), underfrequency (81), under/over voltage (27/59), loss of excitation (40), ground fault (51g), negative sequence (46), synchronization check (25), voltage balance (60).
>
> Note: The number in the parentheses identifies the ANSI device designation.

INSERT 8.3-7-1

The onsite diesel generators will be procured in accordance with an equipment specification which will include requirements based upon the manufacturer's standards and applicable recommendations from documents such as NUREG - 0660 (Reference 15). Control of moisture in the starting air system by the equipment described above will be based upon manufacturer's recommendations. Dust and dirt in the diesel generator room is controlled by the diesel generator building ventilation system described in subsection 9.4.10. Personnel training is addressed as part of overall plant training in subsection 13.2.1. Automatic engine prelube by the equipment described above will be based upon manufacturer's recommendations. Testing, test loading and preventive maintenance is addressed as part of overall plant testing and maintenance in Chapter 13. Instrumentation to support diagnostics during operation are shown on Figure 8.3.1-4. The overall diesel building ventilation deign is described in subsection 9.4.10 and the combustion air systems are described above. The fuel oil storage and handling system is described in subsection 9.5.4. High temperature insulation will be based upon manufacturer's recommendations. Engine oil cooling by the equipment described above will be based upon manufacturer's recommendations. Response to the effects of engine vibration will be based upon manufacturer's recommendations. Diesel building floor coatings are described in subsections 6.1.2.1.4 and 6.1.3.2.

8.3.4 References

1. NEMA MG-1, "Motors and Generators," 1987.

2. IEEE Standard 317, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," 1983.

3. IEEE Standard 422, "Guide for the Design and Installation of Cable Systems in Power Generating Stations," 1986.

4. ICEA Standard Publication P-54-440, "Ampacities of Cables in Open-Top Cable Trays," 1986.

5. National Electrical Code (NEC), 1990.

6. IEEE Standard 485, "IEEE Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations," 1983.

7. IEEE Standard 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," 1981.

8. IEEE Standard 308, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," 1980.

9. IEEE Standard 946, "IEEE Recommended Practice for the Design of Safety-Related dc Auxiliary Power Systems for Nuclear Power Generating Stations," 1985.

10. IEEE Standard 741, "IEEE Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," 1990.

11. IPCEA Standard Publication P-46-426-1962, "Power Cable Ampacities, Volume I - Copper Conductors."

12. IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Storage Batteries for Generating Stations and Substations," 1987.

13. Young, G. L. et al., "Cable Separation - What Do Industry Programs Show?," IEEE Transactions of Energy Conversion, September 1990, Volume 5, Number 3, pp 585-602.

14. WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," September 1993.

15. NUREG-0660, "NRC Action Plan Developed as a result of the TMI-2 accident," May 1980.

- The bulbs are not seismically qualified.

9.5.3.3 Safety Evaluation

The areas that require lighting for safe shutdown are the main control room, and the remote shutdown area when the main control room is not accessible.

- Lighting fixtures in the main control room and remote shutdown area are seismic Category II.
- Emergency and panel lighting circuits up to the lighting fixture are routed in seismic Category I raceways.
- Panel Lighting circuits up to the lighting fixture are treated as Class 1E and Classified as associated. This is acceptable to the Class 1E power supply because of the over current protective device coordination.
- Bulbs are not seismically qualified. However, the bulbs con only fail open and therefore do not represent a hazard to the Class 1E power sources.
- Power to normal and emergency lighting in the main control room and in the remote shutdown area is supplied from the redundant divisions of Class 1E dc and UPS system through two series fuses for isolation. The fuses protect the batteries from failures of the non-1E lighting circuits. The Class 1E batteries provided in the Class 1E dc and UPS system are capable of powering the emergency lighting in these rooms for 72 hours when the normal ac sources are not available. Operation beyond 72 hours is described in subsection 8.3.1.1.1.

9.5.3.4 Test and Inspections

The ac lighting circuits are normally energized and require no periodic testing. The 8-hour battery pack lighting is inspected and tested periodically.

9.5.3.5 Combined License Information for Plant Lighting

This section has no requirements to be provided in support of Combined License application.

9.5.4 Standby Diesel and Auxiliary Boiler Fuel Oil System

This subsection describes the features of the standby diesel and auxiliary boiler fuel oil system. Both the standby diesel generators and the auxiliary boiler are supplied by a combined storage system of fuel oil storage tanks. Two above-ground fuel oil storage tanks for the combined system service are provided. These tanks store diesel grade fuel suitable for either service. The standby diesel generators are described in subsection 8.3.1.1.2 and the auxiliary boiler is described in subsection 10.4.10.