Westinghouse Electric Corporation Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

DCP/NRC1286
NSD-NRC-98-5605
Docket No.: 52-003

March 6, 1998

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

ATTENTION: T. R. QUAY

SUBJECT: AP600 RESPONSE TO FSER OPEN ITEMS, SSAR SECTION 6.2 MARKUP, AND AP600 RESPONSE TO INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (ITAAC) RELATED TO PXS/RCS AND CONTAINMENT

REFERENCE: 1. Westinghouse letter to NRC, letter number DCP/NRC1247, dated February 20, 1998.
2. Westinghouse letter to NRC, letter number DCP/NRC1282, dated March 5, 1998.

Dear Mr. Quay:

Enclosure 1 of this letter contains the Westinghouse responses to FSER open items on the AP600. A summary of the enclosed responses is provided in Table 1. Included in the table is the FSER open item number, the associated OITS number, and the status that will be designated in the Westinghouse status column of OITS.

Enclosure 2 provides revisions to SSAR section 6.2 that supersede the package of markups provided in Attachment 1 of Reference 1. This package of SSAR section 6.2 markups corrects the footer revision number to reflect the revision number shown in the controlled copy of the SSAR (13 pages affected), and reflects changes to SSAR Revision 20. Also note that this enclosure is consistent with the markups provided via Reference 2 to address relocation of the MSLB break.

1/1

Enclosure 3 provides the Westinghouse response to ITAAC questions related to comments on Passive Core Cooling System (PXS) and RCS.

E004

Enclosure 4 provides the Westinghouse response to ITAAC questions related to Containment.

A summary of the ITAAC responses contained in Enclosures 3 and 4 is provided in Table 2. Included in the table is the ITAAC open item number, the associated OITS number, and the status that will be designated in the Westinghouse status column of OITS.

9803130129 980306
PDR ADOCK 05200003
A PDR

Note that ITAAC open item 640.167 (OITS #6511) was revised following the January 15, 1998 ITAAC meeting by Westinghouse letter DCP/NRC1227, dated January 23, 1998. The response provided in the January 23, 1998 letter address the issues that were raised at the meeting. Therefore, OITS item 6511 is closed from the Westinghouse perspective and will be noted in OITS as "Action N" in the Westinghouse status column.

The NRC should review the enclosures and inform Westinghouse of the status to be designated in the "NRC Status" column of OITS.

Please contact me on (412) 374-4334 when you receive this package to confirm receipt.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Enclosures

cc: W. C. Huffman, NRC (Enclosure 1)
J. E. Lyons, NRC (Enclosure 1)
T. J. Kenyon, NRC (Enclosure 1)
J. M. Sebrosky, NRC (Enclosure 1)
D. C. Scaletti, NRC (Enclosures 1 and 2)
J. N. Wilson, NRC (Enclosures 3 and 4)
N. J. Liparulo, Westinghouse (w/o Enclosures)

Table 1
List of FSER Open Items Included in Letter DCP/NRC1286

FSER Open Item	OITS Number	Westinghouse status in OITS
Informal PRA Question	6623	Confirm W
480.1146F (R1)	6570	Confirm W
640.107 (R1)	5861	Confirm W
720.423F (R2)	6135	Confirm W
9.5.1-1	6618	Confirm W
9.5.1-4	6621	Confirm W
9.5.1-5	6622	Confirm W

Table 2 List of ITAAC Open Items Included in Letter DCP/NRC1286		
ITAAC Open Item	**OITS Number**	**Westinghouse status in OITS**
PXS/RCS:		
640.110 (R1)	5864	Confirm W
640.117 (R1)	5871	Confirm W
640.121 (R1)	5875	Confirm W
640.123 (R1)	5877	Confirm W
640.124 (R1)	5878	Confirm W
640.164 (R1)	6192	Confirm W
N/A	6632	Confirm W
N/A	6633	Confirm W
Containment:		
640.56 (R1)	5238	Confirm W
640.177 (R0)	6610	Action N
640.181 (R0)	6614	Action N
N/A	6627	Confirm W

Enclosure 1 to Westinghouse
Letter DCP/NRC1286

March 6, 1998

Informal PRA Question (OITS #6623)

Item #8 of NRC letter dated 2/19/98 reads:

Reconcile the containment isolation valve tables in PRA Chapter 24 and SSAR Chapter 6. Confirm any differences between the tables had no impact on the results of the screening analysis found in the PRA.

Response:

This item was discussed during a 2/25/98 telecon with NRC (J. Sebrosky and R. Palla). Westinghouse agreed to address this item by adding a discussion into PRA Appendix C.

PRA Revision:

The information below will be incorporated into Table C-1 of PRA Appendix C for PRA Revision 11:

System	Description of Change	Potential Impact on PRA Results
Containment Isolation	Some valves were renumbered, a few valves were removed or the valve type changed, eight penetrations are no longer in the design (4 PCS containment pressure instrument lines, SGS SG blowdown out, CVS water in to CMTs and accumulators, CAS breathing air in, WLS RCDT out), and two SFW penetrations have been added to the design.	Does not adversely affect the PRA results or insights. The penetrations that were screened from further analysis in PRA Chapter 24 remain screened out. There are no changes to the penetrations that would cause one to switch to require further evaluation in the fault tree models. One of the penetrations that was evaluated in the fault tree models (WLS, RCDT out) no longer exists. One of the penetrations that was evaluated in the fault tree models (CAS, instrument air) could now be screened from further evaluation because the line size decreased to 2", which meets one of the screening criteria presented in PRA Chapter 24. The two SFW penetrations were evaluated by the SFW system design changes earlier in Table C-1. These SFW penetrations would not effect the results of the CIB fault tree. Thus, it is concluded that the PRA fault tree models conservatively include two penetrations that need not be included in the model due to design changes. The PRA results are conservative relative to these changes.

Question: 480.1146F (OITS #6570) REVISION 1 TO RESPONSE

BASES Figure B 3.6.8-1 of TS 3.6.8 should be incorporated into the TS and not just the bases, since it establishes the required completion time to achieve containment closure.

Response: REVISION 1

The figure provides allowable closure times for four representative sets of plant conditions. However other conditions (RCS temperature, IRWST temperature, etc.) affect the actual times to steaming. The final presentation of this information (curves, calculations, etc.) will be implemented by the COL ~~holder~~ applicant. The Bases are updated to reflect additional insights regarding this figure.

SSAR Revision:

Technical Specification 3.6.8 BASES, **SSAR Section 5.4 and SSAR Section 6.3**.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Containment Penetrations

BASES

BACKGROUND

Containment closure capability is required during shutdown operations when there is fuel inside containment. Containment closure is required to maintain within containment the cooling water inventory. Due to the large volume of the IRWST and the reduced sensible heat during shutdown, the loss of some of the water inventory can be accepted. Further, accident analyses have shown that containment closure capability is not required to meet offsite dose requirements. Therefore, containment does not need to be leak tight as required for MODES 1 through 4.

In MODES 5 and 6, the LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no requirement for containment leak tightness, compliance with the Appendix J leakage criteria and tests are not required.

In Modes 5 and 6, there is no potential for steam release into the containment immediately following a accident. Pressurization of the containment could only occur after heatup of the IRWST due to PRHR HX operation (MODE 5 with RCS intact) or after heatup of the RCS with direct venting to the containment (MODE 5 with reduced RCS inventory or MODE 6 with the refueling cavity not fully flooded) or after heatup of the RCS and refueling cavity (MODE 6 with refueling cavity fully flooded). The time from loss of normal cooling until steam release to the containment for four representative sets of plant conditions ~~those different MODES~~ is shown in Figure B 3.6.8-1 as a function of time after shutdown. Because local manual action may be required to achieve containment closure it is assumed that the containment hatches, air locks and penetrations must be closed prior to steaming into containment.

The containment equipment hatches, which are part of the the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least [four] bolts. Good engineering practice dictates that bolts required by this LCO be approximately equally spaced. Alternatively, if open, each equipment hatch can be

INSERT 440.1146F-1

(continued)

480.1146F-2 (R1)

Insert 480.1146F-1

Figure B 3.6.8-1 provides allowable closure times for four representative sets of plant conditions. The time to steaming is dependent on various plant parameters (RCS temperature, IRWST temperature, etc.) and plant configuration (RCS Pressure Boundary Intact, RCS Open, etc.). Therefore, the actual representation of the time to steaming may be different than that provided in Figure B 3.6.8-1. In determining the minimum time to steaming, conservative assumptions regarding core decay heat, RCS configuration, and initial RCS inventory are used to minimize the calculated time to steaming.

Per Reference 3, the most risk significant events during shutdown are events that lead to a loss of RNS cooling. Of these, the limiting events that lead to steaming to containment are the loss of shutdown cooling events that are presented in Reference 2. For such events, time to steaming is dependent on the postulated RCS configuration (intact versus open), and is based on the response of the plant considering features such as the operation of the 4th stage ADS valves if necessary, status of the upper internals, status of refueling cavity, etc. Conservative assumptions regarding these features are made in the determination of the minimum time to steaming. The risk of overdraining the RCS has been significantly reduced in the AP600 due to the automatic protection features associated with the hot leg level instruments which isolate letdown on low hot leg water level. For such events, a loss of RNS cooling does not occur.

The assumptions used in determining the required closure time for the various containment openings should be conservative, and should be consistent with the plant operating procedures, staffing levels, and status of the containment openings. The evaluation should consider the ability to close the containment for the limiting loss of shutdown cooling event, and considering the possibility of a station blackout.

480.1146F(R1)-3

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.8.2

Each of the two equipment hatches is provided with a set of hardware, tools, equipment, and self-contained power source for moving the hatch from its storage location and installing it in the opening. The required set of hardware and tools shall be visually inspected to ensure that they can perform the required functions. The equipment and power source shall be inspected and/or operated as necessary to verify that the hatch can be installed. The power source shall be verified as containing sufficient energy to install the hatch from the storage location.

The 7 day Frequency is adequate considering that the hardware, tools, equipment, and power sources are dedicated to the associated equipment hatch and not used for any other functions.

The SR is modified by a Note which only requires that the surveillance be met for an open equipment hatch. If the equipment hatch is installed in position, then the availability of the means to install the hatch is not required.

SR 3.6.8.3

This Surveillance demonstrates that at least one valve in each open penetration actuates to its isolation position on manual initiation or on an actual or simulated containment isolation signal. The 24 month Frequency maintains consistency with other similar valve testing requirements. The OPERABILITY requirements for the Containment Isolation function are specified in LCO 3.3.2.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.

2. WCAP-14837, Revision ~~1~~ 2, "AP600 Shutdown Evaluation Report."

3. AP600 Probabilistic Risk Assessment.

manual isolation valve upstream of the residual heat removal pumps. These valves are provided for pump maintenance.

The normal residual heat removal system suction header is continuously sloped from the reactor coolant system hot leg to the pump suction. This eliminates any local high points where air could collect and cause low net positive suction head, pump binding and a loss of residual heat removal capability.

The discharge of each residual heat removal pump is directed to its respective residual heat removal heat exchanger. The outlet of each residual heat removal heat exchanger is routed to the common discharge header, which contains a normally closed, motor-operated isolation valve. For pump protection, a miniflow line with an orifice is included from downstream of the residual heat removal heat exchanger to upstream of the residual heat removal pump suction. This line is sized to provide sufficient pump flow when the pressure in the reactor coolant system is above the residual heat removal pump shutoff head.

Once inside containment, the common discharge header contains a check valve that acts as a containment isolation valve. Downstream of the check valve, the discharge header branches into two lines, one to each passive core cooling system direct vessel injection nozzle. These branch lines each contain a stop check valve and check valve in series that serve as the reactor coolant system pressure boundary. A line to the chemical and volume control system demineralizers branches from one of the direct vessel injection lines. This line is used for shutdown purification of the reactor coolant system. Another line branches from the same direct vessel injection line to the in-containment refueling water storage tank which is used when cooling the tank.

One safety relief valve is located on the normal residual heat removal system suction header inside containment. This valve provides low temperature overpressure protection of the reactor coolant system. Subsection 5.4.9 describes the sizing basis of this valve. Another safety relief valve outside of containment provides protection against excess pressure for the piping and components.

When the normal residual heat removal system is in operation, the water chemistry is the same as that of the reactor coolant. Sampling may be performed using the normal residual heat removal heat exchangers channel head drain connections. Sampling of the reactor coolant system using these connections is available at shutdown. Sampling of the in-containment refueling water storage tank is available during normal plant operation.

Shutdown and

5.4.7.2.1 Design Features Addressing Mid-Loop Operations

The following is a summary of the specific AP600 design features that address Generic Letter (GL) 88-17 regarding mid-loop operations. In addition, these features support improved safety during shutdown as discussed in reference 10.

Loop Piping Offset - As shown in Figure 5.3-6, the reactor coolant system hot legs and cold legs are vertically offset. This permits draining of the steam generators for nozzle dam insertion with hot leg level much higher than traditional designs. The reactor coolant system

must be drained to a level which is sufficient to provide a vent path from the pressurizer to the steam generators. This is nominally 80 percent level in the hot leg. This loop piping offset also allows a reactor coolant pump to be replaced without removing a full core.

Step-nozzle Connection - The normal residual heat removal system employs a step-nozzle connection to the reactor coolant system hot leg. The step-nozzle connection has two effects on mid-loop operation. One effect is to substantially lower the RCS hot leg level at which a vortex occurs in the residual heat removal pump suction line due to the lower fluid velocity in the hot leg nozzle. This increases the margin from the nominal mid-loop level to the level where air entrainment into the pump suction begins.

Another effect of the step-nozzle is that, if a vortex should occur, the maximum air entrainment into the pump suction has been shown experimentally to be no greater than 5 percent. This level of air ingestion will make air binding of the pump much less likely.

No Normal Residual Heat Removal Throttling During Mid-Loop - The normal residual heat removal pumps are designed to minimize susceptibility to cavitation. The plant piping configuration, piping elevations and routing, and the pump net positive suction head characteristics allow the normal residual heat removal pumps to be started and operated at their full design flow rates with saturated conditions in the reactor coolant system. The normal residual heat removal system operates without the need for throttling a residual heat removal control valve when the level in the reactor coolant system is reduced to a mid-loop level.

Self-Venting Suction Line - The residual heat removal pump suction line is sloped continuously upward from the pump to the reactor coolant system hot leg with no local high points. This eliminates potential problems with refilling the pump suction line if a residual heat removal pump is stopped when cavitating due to excessive air entrainment. With the self-venting suction line, the line will refill and the pumps can be immediately restarted once an adequate level in the hot leg is re-established.

Insert A →

Hot Leg Level Instrumentation - The AP600 reactor coolant system contains level instrumentation in each hot leg with indication in the main control room. In addition, the wide-range pressurizer level instrumentation (used during cold plant operations) ~~is available to measure to the bottom of the hot legs. There is~~ which provides continuous level indication in the main control room from the normal level in the pressurizer ~~to the range of the~~ two narrow-range hot leg level instrumentation are available. Alarms are provided to alert the operator when the reactor coolant system hot leg level is approaching a low level. The isolation valves in the line used to drain the reactor coolant system close on a low reactor coolant system level during shutdown operations. Operations required during mid-loop are performed by the operator in the main control room. The level monitoring and control features significantly improve the reliability of the AP600 during mid-loop operations.

Reactor Vessel Outlet Temperature - Reactor coolant system hot leg wide range temperature instruments are provided in each hot leg. The orientation of the wide range thermowell-mounted resistance temperature detectors enable measurement of the reactor coolant fluid in

Insert A

Wide Range Pressurizer Level - A nonsafety-related independent pressurizer level transmitter, calibrated for low temperature conditions, provides water level indication during startup, shutdown, and refueling operations in the main control room and in the remote shutdown workstation. The upper level tap is connected to an ADS valve inlet header above the top of the pressurizer. The lower level tap is connected to the bottom of the hot leg. This provides level indication for the entire pressurizer and a continuous reading as the level in the pressurizer decreases to mid-loop levels during shutdown operations.

reactor coolant system hot leg level is approaching a low level. The isolation valves in the line used to drain the reactor coolant system close on a low reactor coolant system level during shutdown operations. Operations required during mid-loop are performed by the operator in the main control room. The level monitoring and control features significantly improve the reliability of the AP600 during mid-loop operations.

Reactor Vessel Outlet Temperature - Reactor coolant system hot leg wide range temperature instruments are provided in each hot leg. The orientation of the wide range thermowell-mounted resistance temperature detectors enable measurement of the reactor coolant fluid in the hot leg when in reduced inventory conditions. In addition, at least two incore thermocouple channels are available to measure the core exit temperature during midloop residual heat removal operation. These two thermocouple channels are associated with separate electrical divisions.

ADS Valves - The automatic depressurization system first-, second-, and third-stage valves connected to the top of the pressurizer, are open whenever the core makeup tanks are blocked during shutdown conditions while the reactor vessel upper internals are in place. This provides a vent path to preclude pressurization of the reactor coolant system during shutdown conditions when decay heat removal is lost. This also allows the in-containment refueling water storage tank to automatically provide injection flow if it is actuated on a loss of decay heat removal.

The capability to restore containment integrity during shutdown conditions is provided. The containment equipment hatches are equipped with guide rails that allow reinstallation of the hatches to re-establish containment integrity. The containment design also includes penetrations for temporary cables and hoses needed for shutdown operations.

Procedures direct the operator in the proper conduct of midloop operation and aid in identifying and correcting abnormal conditions that might occur during shutdown operations.

5.4.16 References

1. Eshleman, R. L., "Flexible Rotor-Bearing System Dynamics, Part I. Critical Speeds and Response of Flexible Rotor Systems," Flexible Rotor System Subcommittee, Design Engineering Division. American Society of Mechanical Engineers, 1972.

2. Hagg, A. C. and Sankey, G. O., "The Containment of Disk Burst Fragments by Cylindrical Shells," ASME Journal of Engineering for Power, April 1974, pp. 114-123.

3. ASTM-A-609-78, Standard Specification for Longitudinal Beam Ultrasonic Inspection of Carbon and Low-alloy Steel Castings.

4. ASTM-E-165-80, Practice for Liquid Penetrant Inspection Method.

5. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors."

6. ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

7. ANSI N278.1-1975, Self-Operated and Power-Operated Safety-Relief Valves Functional Specification Standard.

8. ANSI B16.41-1983, Functional Qualification Requirements for Power Operated Active Valve Assemblies for Nuclear Power Plants.

9. ANSI B16.34-1981, Valves - Flanged and Buttwelding End.

10. WCAP 14837, "AP600 Shutdown Evaluation Report."

A. Increase in heat removal by the secondary system

1. Inadvertent opening of a steam generator power-operated atmospheric steam relief or safety valve

2. Steam system piping failure.

B. Decrease in heat removal by the secondary system

1. Loss of Main Feedwater Flow
2. Feedwater system piping failure.

C. Decrease in reactor coolant system inventory

1. Steam generator tube rupture

2. Loss of coolant accident from a spectrum of postulated reactor coolant system piping failures

3. Loss of coolant due to a rod cluster control assembly ejection accident (This event is enveloped by the reactor coolant system piping failures.)

D. Shutdown Events (Reference 3)

1. Loss of Startup Feedwater

2. Loss of normal residual heat removal system with reactor coolant system pressure boundary intact

3. Loss of normal residual heat removal system during mid-loop operation

4. Loss of normal residual heat removal system with refueling cavity flooded.

The events listed in groups A and B are non-LOCA events where the primary protection is provided by the passive core cooling system passive residual heat removal heat exchanger. For these events, the passive residual heat removal heat exchanger is actuated by the protection and monitoring system for the following conditions:

- Steam generator low narrow range level, coincident with startup feedwater low flow
- Steam generator low wide range level
- Core makeup tank actuation
- Automatic depressurization actuation
- Pressurizer water level - High 3
- Manual actuation

6.3.7.7 Automatic Depressurization System Actuation at 24 Hours

A timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This prevents discharging the Class 1E dc power sources such that they are no longer able to operate the automatic depressurization system valves. If power becomes available to the dc batteries and they are no longer discharging prior to activation of the timer, then the automatic depressurization system actuation would be delayed. If the plant does not need actuation of the automatic depressurization system based on having stable pressurizer level, full core makeup tanks, and high and stable in-containment refueling water storage tank levels, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the automatic depressurization system and allow for its actuation later should the plant conditions unexpectedly degrade.

6.3.8 Combined License Information

6.3.8.1 Containment Cleanliness Program

The Combined License applicants referencing the AP600 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages.

6.3.9 References

1. WCAP-8966, "Evaluation of Mispositioned ECCS Valves," September 1977.

2. WCAP-13594 (Proprietary), WCAP-13662 (Nonproprietary), "FMEA of Advanced Passive Plant Protection System."

3. WCAP-14837, "AP600 Shutdown Evaluation Report."

AP600

Question 640.107 (OITS 5861)

REACTOR COOLANT SYSTEM

In ITAAC Table 2.1.2-4, the acceptance criterion for Item 9a for the reactor coolant system flow rate post-fuel load is listed as 189,600 gpm, which is the vessel thermal design flow rate indicated in SSAR Table 4.4-1. This ITAAC acceptance criterion is inconsistent with the vessel minimum measured flow rate of 193,000 gpm specified in technical specification LCO 3.4.1 and SR 3.4.1.3. A clarification should be made in ITAAC to ensure that the ITAAC analysis to determine the post-fuel load RCS flow takes into account appropriate allowance for flow measurement uncertainties.

Response (Revision 1):

The inspection, test, analysis column of item 9a in Table 2.1.2-4 has been modified to account for measurement uncertainty. The flow rate specified in the acceptance criteria has been revised to be consistent with the SSAR and technical specification.

SSAR Revision:

None.

ITAAC Revision:

Modify **Table 2.1.2-4**

9.a) The RCS provides circulation of coolant to remove heat from the core.	Testing and analysis to measure RCS flow with four reactor pumps operating at no-load RCS pressure and temperature conditions will be performed. Analyses to convert the measured pre-fuel load flow rate will be performed. ***This analysis accounts for flow measurement uncertainties.***	The calculated post-fuel load RCS flow rate is $\geq$ ~~189,600~~ 193,200 gpm.

Question 720.423F (OITS - 6135)

In meeting the RTNSS criteria, credit was taken for external reactor vessel cooling (ERVC) as a strategy for retaining molten core debris in-vessel. This results in the majority of core melt accidents (~90 percent) being arrested in-vessel, thereby avoiding RPV failure and associated containment challenges from ex-vessel phenomena. Successful RCS depressurization and reactor cavity flooding are prerequisites for ERVC, and credit for these aspects of ERVC in the focussed PRA is appropriate since both functions are fulfilled by safety-related systems. However, the nonsafety-related RPV thermal insulation system is also required for successful ERVC. The thermal insulation system limits thermal losses during normal operations, but provides the engineered pathway for supplying water cooling to the vessel and venting steam from the annulus during severe accidents. Attributes of the system include specific RPV/insulation clearances and water/steam flow areas based on scaled tests, integral ball-and-cage check valves and buoyant steam vent dampers which change position during flood-up of the reactor cavity, and insulation panel and support members designed to withstand the hydrodynamic loads associated with ERVC.

If credit for ERVC is reduced, large release frequency and CCFP would increase proportionally since all RPV breaches are assumed to lead to early containment failure in the PRA. Under the most limiting assumption of no credit for ERVC, the large release frequency would approach the core melt frequency and CCFP would approach 1.0. In view of the reliance on ERVC to meet the Commission's large release frequency goals, the staff will require an appropriate level of regulatory oversight of the RPV thermal insulation system. This oversight should provide reasonable assurance that the as-built insulation system conforms with design specifications contained in Chapter 39 of the PRA, and that the operability of the system is confirmed through periodic surveillance.

The RPV insulation design description and functional requirements are not currently included in the SSAR, ITAAC, or reliability assurance program. The design description and functional requirements for the RPV insulation should be added to the SSAR, and important criteria associated with the insulation design should be incorporated into the ITAAC, including information related to the necessary clearances/flow areas, and the check valves and steam vent dampers. The system should be included as a risk-significant SSC in the reliability assurance program, and reliability/availability controls and goals should be provided, consistent with maintenance rule guidelines, to assure that operability of the system and moving parts is maintained.

Westinghouse Response (Revision 2):

Functional requirements for the reactor vessel insulation was incorporated in section 5.3.5 of Revision 14 of the AP600 SSAR. Based on discussions with the NRC staff, more information was requested to be included in the SSAR. SSAR Section 5.3.5 is modified to include the design bases and design description for the reactor vessel insulation and is attached. In addition, the reactor vessel insulation is included as a risk-significant SSC in the reliability assurance program as shown in the proposed revision to SSAR section 17.4. The AP600 Certified Design Material is also revised to include appropriate ITAACs for the reactor vessel insulation per the response to RAI 720.442F.

SSAR Section 5.3.5 is further revised based on discussions held with the staff at the January 22, 1998 chapter closeout meeting and subsequent discussions.

SSAR Revision:

Revised SSAR Sections 5.3.5, 17.4 attached. See the response to RAI 720.442F for changes to the Certified Design Material.

5.3.5 Reactor Vessel Insulation

5.3.5.1 Reactor Vessel Insulation Design Bases

Reactor vessel insulation is provided to minimize heat losses from the primary system. Nonsafety-related reflective insulation similar to that in use in current pressurized water reactors is utilized. The AP600 reactor vessel insulation contains design features to promote in-vessel retention following severe accidents. In the unlikely event of a beyond design basis accident, the reactor cavity is flooded with water, and the reactor vessel insulation allows heat removal from core debris via boiling on the outside surface of the reactor vessel. The reactor vessel insulation permits a water layer next to the reactor vessel to promote heat transfer from the reactor vessel. This is accomplished by providing:

- A means of allowing water free access to the region between the reactor vessel and insulation.
- A means to allow steam generated by water contact with the reactor vessel to escape from the region surrounding the reactor vessel.
- A support frame to prevent the insulation panels from breaking free and blocking water from cooling the reactor vessel exterior surface.

The reactor vessel insulation and its supports are designed to withstand bounding pressure differentials across the reactor vessel insulation panels during the period that the reactor vessel is externally flooded with water and the core retained in the reactor vessel through heat removal from the vessel wall accomplished by the water. This is accomplished by providing a minimum flow area of 7.5 ft^2 in the portions of the flow paths required to vent steam. The flow path from the reactor loop compartment to the reactor cavity provides an open flow path for water to flood the reactor cavity. The reactor vessel insulation inlet assembles are designed to minimize the pressure drop during ex-vessel cooling to permit water to cool the vessel.

5.3.5.2 Description of Insulation

A schematic of the reactor vessel, the vessel insulation and the reactor cavity is shown in Figure 5.3-7. The insulation is mounted on a structural frame that is supported from the wall of the reactor cavity. The vertical insulation panels are designed to have a minimum gap between the insulation and reactor vessel not less than 2 inches when subjected to the dynamic loads in the direction towards the vessel that result during ex-vessel cooling. A nominal gap (with no deflection) of more than twice the minimum gap is provided.

The conical design of the bottom portion of the vessel insulation is constructed of flat panels. This provides a single point of contact with the spherical portion of the vessel in the event that an insulation panel becomes dislodged. This prevents hot spots from developing on the reactor vessel and permits sufficient flow to maintain in-vessel retention. The nominal gap

between the conical portion of the insulation and the spherical portion of the reactor vessel is not less than 9 inches.

The structural frame supporting the insulation is designed to withstand the bounding severe accident loads without exceeding deflection criteria. The fasteners holding the insulation panels to the frame are also designed for these loads.

At the bottom of the insulation are water inlet assemblies. Each water inlet assembly is normally closed to prevent an air circulation path through the vessel insulation. The inlet assemblies are self-actuating passive devices. The inlet assemblies open when the cavity is filled with water. This permits ingress of water during a severe accident, while preventing excessive heat loss during normal operation.

The total flow area of the water inlet assemblies have sufficient margin to preclude significant pressure drop during ex-vessel cooling during a severe accident. The minimum total flow area for the water inlets assemblies is 6 ft^2. Due to the relatively low approach velocities in the flow paths leading to the reactor cavity, and due to the relatively large minimum flow area through each water inlet assembly, with an area of at least 7 in^2, the water inlet assemblies are not susceptible to clogging from debris inside containment.

Near the top of the lower insulation segment are steam vent dampers. These dampers are normally closed to prevent reactor vessel heat loss, and a small buildup of steam pressure under the insulation will cause them to open to the vent position. The steam vent dampers are passive, self-actuated devices and will operate when steam is generated under the insulation with the cavity filled with water.

Extensive maintenance of the vessel insulation is not normally required. Periodic verification that the vessel insulation moving parts can be performed during refueling outages.

5.3.5.3 Description of External Vessel Cooling Flooded Compartments

Ex-vessel cooling during a severe accident is provided by flooding the reactor coolant system loop compartment including a vertical access tunnel, the reactor coolant drain tank room, and the reactor cavity. Water from these compartments replenishes the water that comes in contact with the reactor vessel and is boiled and vented to containment. The opening between the vertical access tunnel and the reactor coolant drain tank room is approximately 100 ft^2. Figure 5.3-8 depicts the flooded compartments that provide the water for ex-vessel cooling. ~~The opening between the reactor coolant drain tank room and the reactor cavity is free from obstruction such that it does not preclude a minimum flow area of 6 ft^2 to permit water to flood the reactor cavity compartment.~~ *The doorway between the reactor cavity compartment and the reactor coolant drain tank room consists of a normally closed door and a damper above the door. The door and damper arrangement, shown in Figure 5.3-9, maintains the proper air flow through the reactor cavity during normal operation. The damper prevents air from flowing into the reactor coolant drain tank compartment, but opens to permit flooding of the reactor cavity from the reactor coolant drain tank compartment. The damper*

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opening has an minimum flow area of 8 ft². It is constructed of light-weight material to minimize the force necessary to open the damper and permit flooding and continued water flow through the opening during ex-vessel cooling.

5.3.5.4 Determination of Forces on Insulation and Support System

The expected forces that may be expected in the reactor cavity region of the AP600 plant during a core damage accident in which the core has relocated to the lower head and the reactor cavity is reflooded have been conservatively established based on data from the ULPU test program (Reference 5). The particular configuration (Configuration III) reviewed closely models the full-scale AP600 geometry of water in the region near the reactor vessel, between the reactor vessel and the reactor vessel insulation. The ULPU tests provide data on the pressure generated in the region between the reactor vessel and reactor vessel insulation. These data, along with observations and conclusions from heat transfer studies, are used to develop the functional requirements with respect to in-vessel retention for the reactor vessel insulation and support system. Interpretation of data collected from ULPU Configuration III experiments in conjunction with the static head of water that would be present in the AP600 is used to estimate forces acting on the rigid sections of insulation. Further evaluation of the forces on the reactor vessel insulation and supports is provided in the AP600 Probabilistic Risk Assessment.

5.3.5.5 Design Evaluation

A structural analysis of the AP600 reactor cavity insulation system demonstrates that it meets the functional requirements discussed above. The analysis encompassed the insulation and support system and included a determination of the stresses in support members, bolts, insulation panels and welds, as well as deflection of support members and insulation panels.

The results of the analyses show that the insulation is able to meet its functional requirements. The reactor vessel insulation provides an engineered pathway for water-cooling the vessel and for venting steam from the reactor cavity.

The reactor vessel insulation is purchased equipment. The purchase specification for the reactor vessel insulation will require confirmatory static load analyses.

5.3.6 Combined License Information

5.3.6.1 Pressure-Temperature Limit Curves

The pressure-temp. curves shown in Figures 5.3-2 and 5.3-3 are generic curves for AP600 reactor vessel design, and they are the limiting curves based on copper and nickel material composition. However, for a specific AP600, these curves will be plotted based on material composition of copper and nickel. Use of plant-specific curves will be addressed by the Combined License applicant during procurement of the reactor vessel.

5.3.6.2 Reactor Vessel Materials Surveillance Program

The Combined License applicant will address a reactor vessel reactor material surveillance program based on subsection 5.3.2.6.

5.3.6.3 Reactor Vessel Materials Properties Verification

The Combined License applicant will address verification of plant-specific belt line material properties consistent with the requirements in subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3.

5.3.6.4 Reactor Vessel Insulation

The Combined License applicant will address verification that the reactor vessel insulation is consistent with the design bases established for in-vessel retention.

5.3.7 References

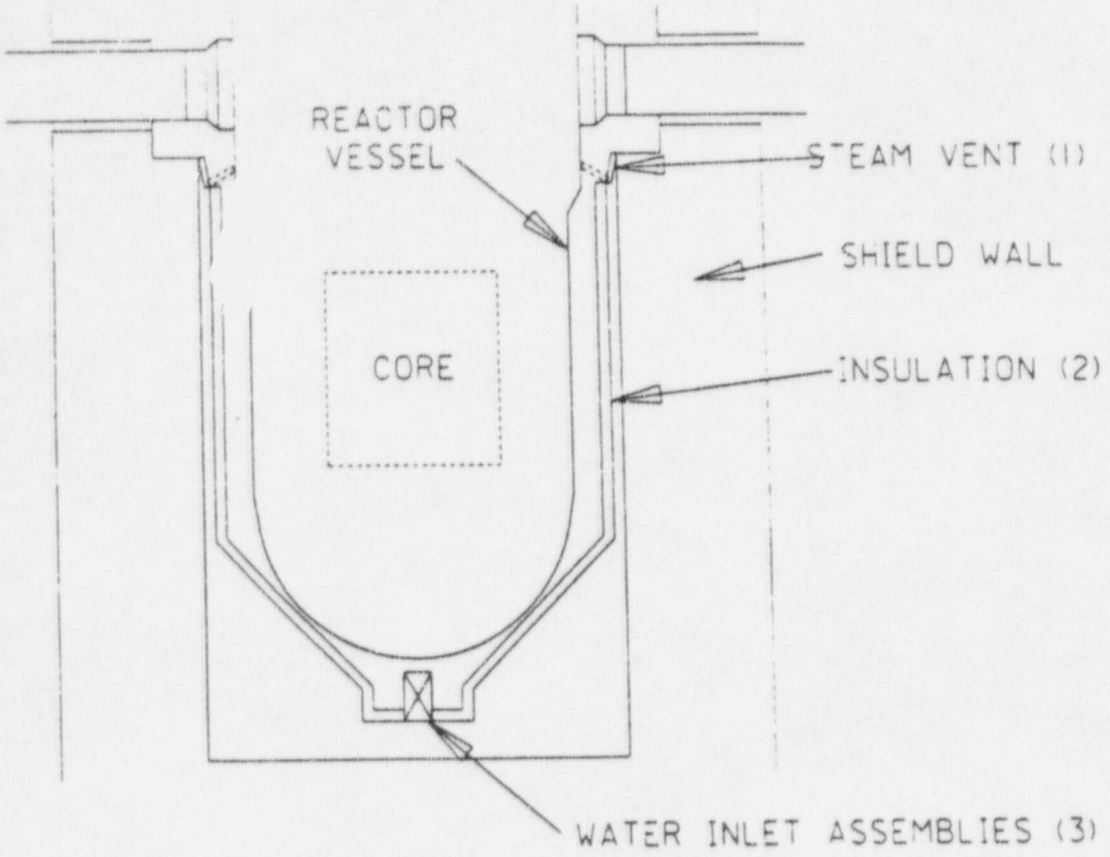
1. ASTM E-185-82, Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

2. Soltesz, R. G., et al., "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Volume 5 -Two-Dimensional Discrete Ordinates Techniques," WANL-PR-(LL)-034, August 1970.

3. SAILOR RSIC Data Library Collection DLC-76, "Coupled, Self-Shielded, 47 Neutron, 20 Gamma-Ray, P3, Cross Section Library for Light Water Reactors."

4. NRC Policy Issue, "Pressurized Thermal Shock," SECY-82-465, November 23, 1982.

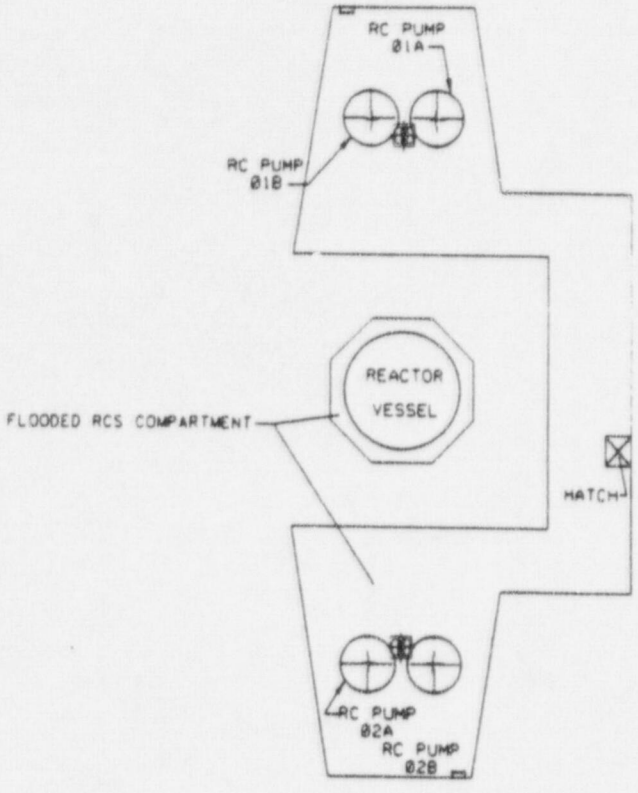
5. Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.



(1) - Minimum steam vent flow area provided in subsection 5.3.5.1
(2) - Minimum gap between insulation and vessel insulation provided in subsection 5.3.5.2
(3) - Minimum flow area provided in subsection 5.3.5.2

Figure 5.3-7

Schematic of Reactor Vessel Insulation



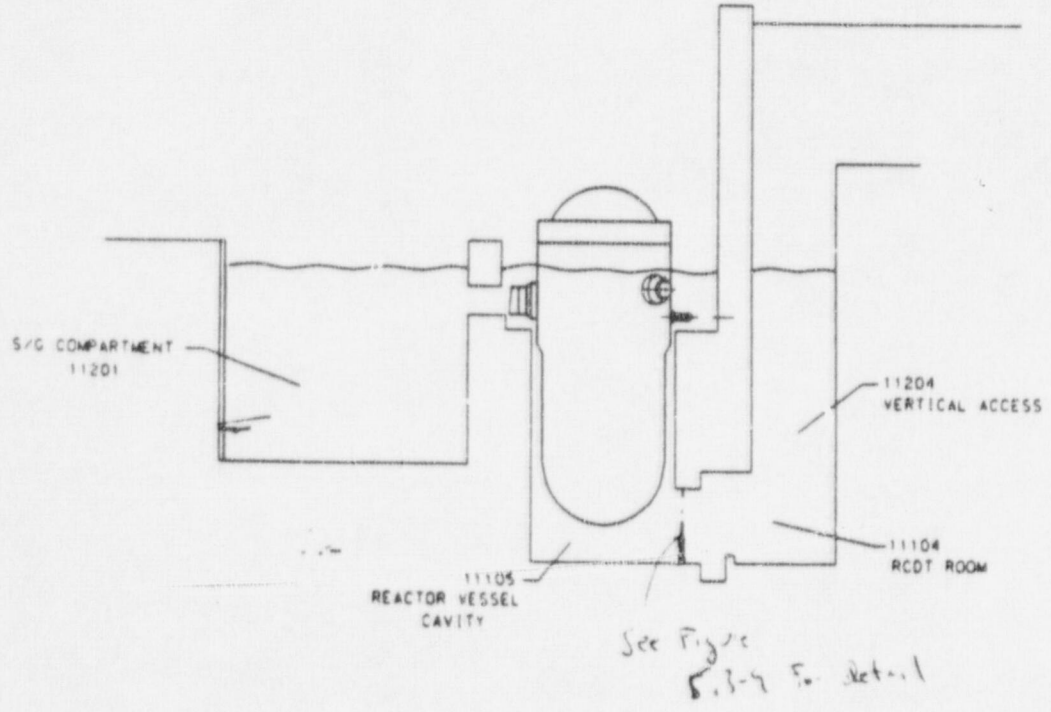
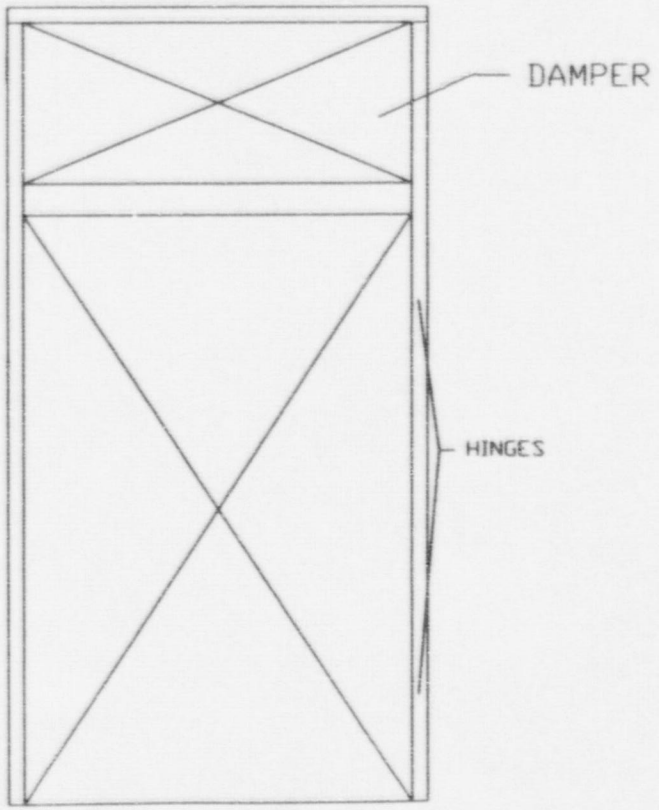


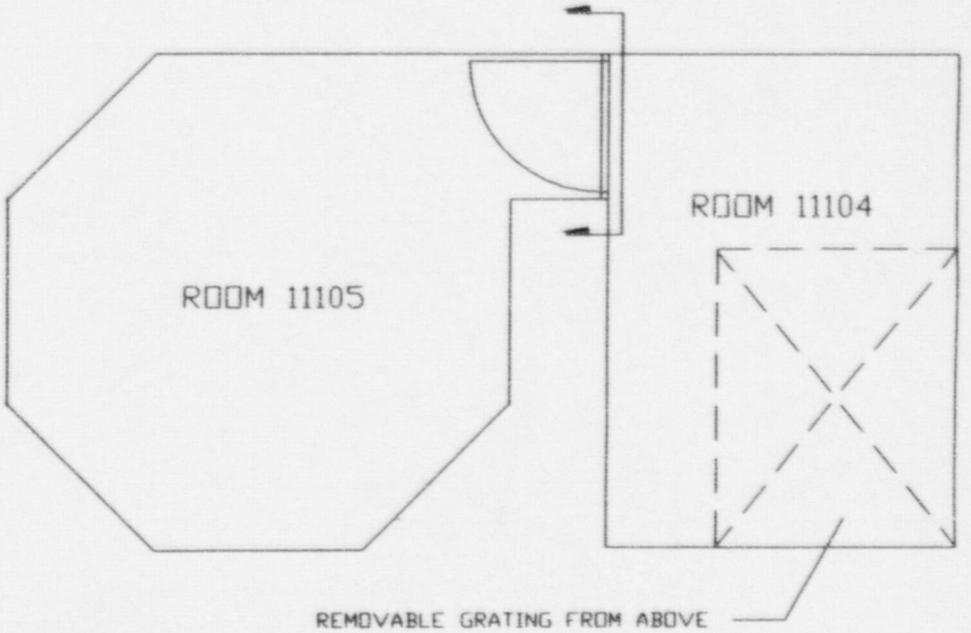
Figure 5.3-8

RCS Fluidal Compartments During Ex-Vessel Cooling

SSAR Figure 5.3-9

DOORWAY BETWEEN RCDT ROOM AND REACTOR CAVITY COMPARTMENT





LOOKING SOUTH FROM RCDT ROOM (11104) INTO REACTOR CAVITY (11105)

Table 17.4-1 (Sheet 7 of 9)

RISK-SIGNIFICANT SSCs WITHIN THE SCOPE OF D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
System: Reactor Coolant System (RCS)		
ADS Stages 1/2/3 Motor-Operated Valves	EP, L2	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
ADS 4th Stage Squib Valves	RAW/CCF	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
Pressurizer Safety Valves	EP	These valves provide overpressure protection of the RCS.
Reactor Vessel Insulation Water Inlet and Steam Vent Devices	EP	These devices provide an engineered flow path to promote in-vessel retention of the core in a severe accident.
System: Normal Residual Heat Removal System (RNS)		
RNS Pumps	EP	These pumps provide shutdown cooling of the RCS. They also provide an alternate RCS lower pressure injection capability following actuation of the ADS. The operation of these pumps is RTNSS-important during shutdown reduced-inventory conditions. RNS valve realignment is not required for reduced-inventory conditions.
RNS Motor-Operated Valves	RRW/FVW	These MOVs align a flowpath for nonsafety-related makeup to the RCS following ADS operation.
System: Spent Fuel Cooling System (SFS)		
SFS Pumps	EP	These pumps provide flow to the heat exchangers for removal of the design basis heat load.

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FSER Open Item 9.5.1-1 (OITS 6618)

Fire areas containing redundant equipment and circuits necessary to achieve and maintain cold shutdown using the normal systems specified in Table 9.5.1-4 of the SSAR have not been provided with automatic suppression in accordance with the specified criteria, and is therefore unacceptable.
Section 9.5.1.3 of the SSAR states that manual partial opening and closing of one of the first stage automatic depressurization (ADS) valves to reduce RCS pressure to the allow initiation of the normal residual heat removal system (RNS) will be used in the event the chemical volume and control system (CVS) is not available. The use of the first stage ADS results in a controlled venting of the RCS to the in containment refueling water storage tank (IRWST) and does not result in an increase in containment pressure or temperature. The use of the first stage ADS as described above is acceptable as an alternative shutdown method provided the normal shutdown equipment and circuits are provided with automatic suppression or are located in separate fire areas.

Response:

Westinghouse and the NRC met on 02/24/98 and agreed to finalize this issue by:

A. Commitment:
Add automatic suppression to the room containing the CVS charging pumps and the electrical penetration room above it.

Comment:
Westinghouse will add automatic suppression to the CVS make-up pump room fire area 1200-AF-01, room 12255 and to the non-1E equipment / penetration room in fire area 1230-AF-02, room 12321.

B. Commitment:
Revise Table 9.5.1-4 to indicate the extent of fire area separation for the RNS function.

Comment:
The definition of the term "separation" has been added to Table 9.5.1-4 as note 5.

C. Commitment:
Provide the NRC with a description of the fire department connection to the fire protection system to the extent it exists.

Comment:
As stated in the 02/24/98 meeting by the NRC, AP600 has no fire truck connection to the main fire water tanks.



SSAR Revision:

A. See revisions to SSAR 9A.3.1.2.6.1, 9A.3.1.3.1.1, Table 9A-3, 9.5.1.2.1.5, Table 9.5.1-4, Figure 9.5.1-1 (sheet 3 of 3) and 3.4.1.2.2.2. Note: Changes made in response to FSER Open Item 9.5.1-5 are also incorporated in the attachment.

B. See revision to Table 9.5.1-4 note 5 in the attachment.

C. None.

9.5.1.2.1.5 Manual Fire Suppression Systems

Manual fire suppression capability is provided in areas that do not require an automatic suppression system. Plant areas that have an automatic suppression system also have manual backup fire suppression capability.

Manual fire suppression capabilities include the yard main fire hydrants and hose stations described in subsection 9.5.1.2.1.3.

Standpipe and Hose Systems

Standpipe systems are provided for each building in accordance with NFPA 14 (Reference 2) requirements for Class III service. Wet standpipe systems are used except inside containment. Individual standpipes are at least 4 inches in diameter for multiple hose connections and 2.5 inches in diameter for single hose connections.

Hose stations are located to facilitate access for firefighting, as described in the fire protection analysis. Areas that contain, or could present a fire exposure event to, safety-related equipment are within reach of at least one effective hose stream. Alternative hose stations are provided for an area where the fire could block access to a single hose station serving that area. To the maximum extent practical, hose stations are located outside of high-radiation areas.

Each hose station has not more than 100 feet of 1.5-inch woven-jacket lined fire hose. Nozzles are provided at each station.

Seismic Standpipe System

The standpipe system serving areas containing equipment required for safe shutdown following a safe shutdown earthquake is designed and supported so that it can withstand the effects of a safe shutdown earthquake and remain functional. The seismically analyzed standpipe system is illustrated on Figure 9.5.1-1. This system also supplies water to automatic suppression systems inside containment ***and in the nonradiologically controlled portion of the auxiliary building*** (see Appendix 9A).

The seismic standpipe system is operated in the same manner during normal plant operation or following a safe shutdown earthquake. It is supplied with water from the safety related passive containment cooling system storage ~~tank~~ and normally operates independently of the rest of the fire protection system. The supply line draws water from a portion of the storage tank, using water allocated for fire protection. This volume of water is sufficient to supply two hose streams, each with a flow of 75 gallons per minute, for 2 hours.

The portion of the system outside containment is a wet standpipe system that is pressurized by the static head of water in the passive containment cooling system tank. The portion of the system inside containment is a dry standpipe system. The supply valve is normally closed for containment isolation. During shutdown periods when the containment is occupied, when operation of containment automatic suppression systems is required, or when containment access is required to fight a fire, the valve is opened to pressurize the system.



In the unlikely event that the water supply from the passive containment cooling system is unavailable or additional water is needed, the seismic standpipe system can be supplied from the fire main by opening the normally closed cross-connect valve with the plant fire main.

A passive containment cooling ancillary water storage tank is provided to supply the seismic standpipe system following a safe shutdown earthquake and after actuation of the passive containment cooling system. The tank is designed and supported so that it can withstand the effects of a safe shutdown earthquake and remain functional. A dedicated portion of the storage capacity of the tank is sufficient to supply two hose streams, each with a flow of 75 gallons per minute, for 2 hours. Normally much more water is available. (Refer to subsection 6.2.2 for additional information.)

A failure of the seismic standpipe system does not prevent successful operation of the passive containment cooling system. A leak in the standpipe system could result in the loss of only a limited amount of water from the passive containment cooling system storage tank, even if no action were taken to isolate the leak. The volume of water allocated for fire protection is not required for passive containment cooling.

Portable Fire Extinguishers

Portable fire extinguishers are provided throughout the plant. Portable extinguishers are readily accessible for use in high radiation areas but are not located within those areas unless the fire protection analysis indicates that a specific requirement exists.

Table 9.5.1-4

CAPABILITIES USED TO ACHIEVE COLD SHUTDOWN FOLLOWING A FIRE

Function	System Capability	Fire Protection
RCS Reactivity Control		
Short Term	Control Rods	separation *(6)*
Long Term	(1)	(1)
RCS Makeup	(1)	(1)
RCS Pressure Control		
Increase	Pressurizer heaters	*fire suppression*~~separation~~
Decrease	~~Auxiliary spray~~ (2)	(2)
Decay Heat Removal (high temperature)	SFW pumps feeding CST water to SG	fire suppression
	SG PORV discharge to atm.	separation
Decay Heat Removal (cold temperature)	RNS pumps circulating RCS	separation
	CCS cooling RNS	fire suppression
	SWS cooling CCS	fire suppression
Process Monitoring	RCS monitoring instruments (PMS)	separation
	Non-1E Instrumentation and Control (3)	*fire suppression or* separation
Support Systems	Instrument Air	fire suppression
	Standby Diesel Generators	fire suppression and separation
	Non-1E AC Power *and control* (3)	*fire suppression or* separation
	~~Non-1E Room Ventilation Fans (4)~~	~~separation~~

Notes: (1) CVS makeup from the BAT provides RCS makeup and boration. ***Automatic suppression is provided in the CVS makeup pump room and in the Non-Class 1E equipment/penetration room.*** If the CVS is damaged by a fire, one CMT can provide this capability~~ without heating up the containment~~.



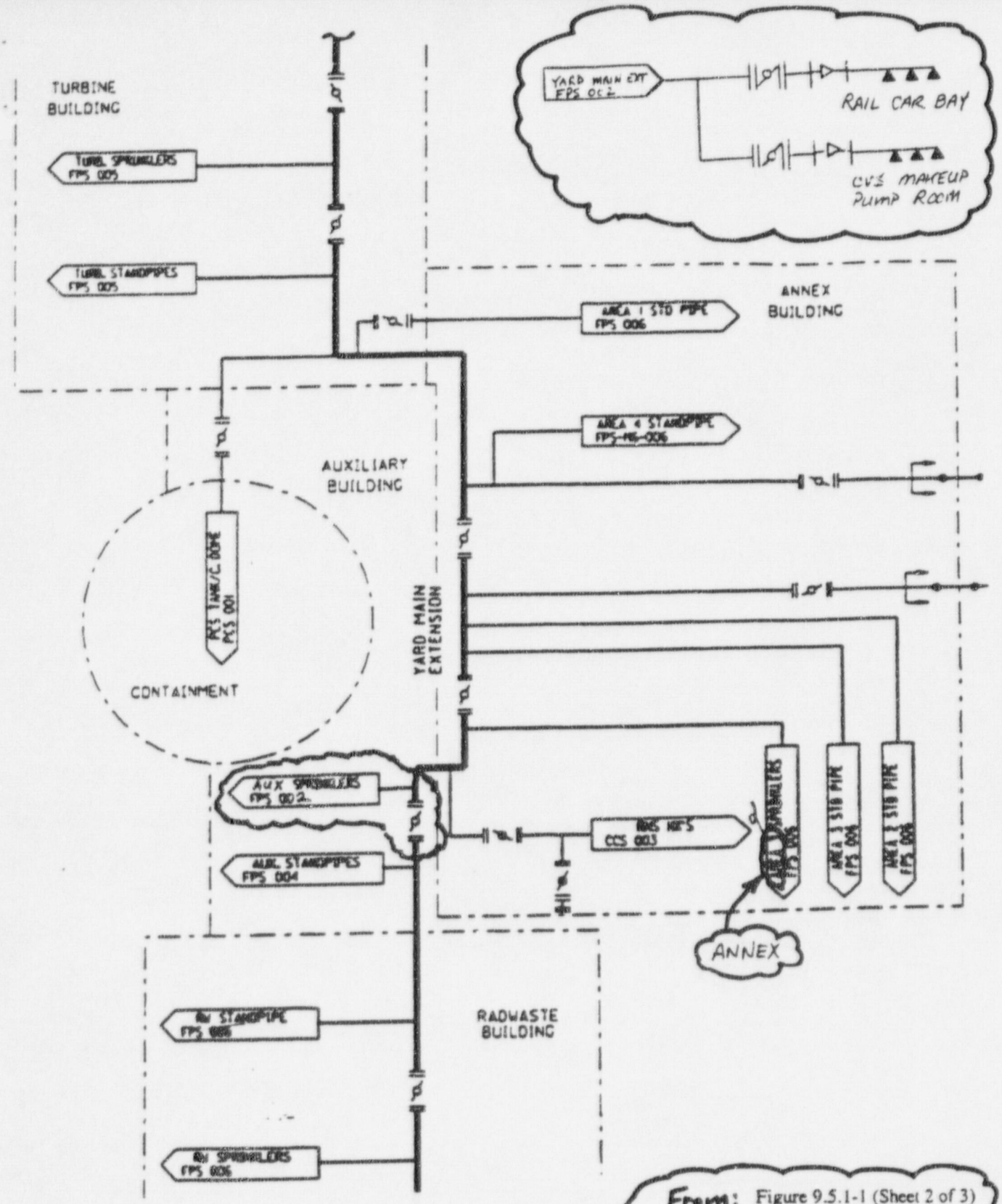
(2) CVS auxiliary spray provides pressurizer pressure reduction. ***Automatic suppression is provided in the CVS makeup pump room and in the Non-Class 1E equipment/penetration room.*** If the CVS is damaged by a fire, one ADS stage 1 valve used in a low capacity throttled vent mode of operation can slowly depressurize the RCS without loss of RCS pressure boundary ~~or heating up of the containment~~.

(3) The portions of the non-1E AC power and the non-1E instrumentation and control system required are those needed to operate ***cold shutdown*** components; local control is sufficient (switchgear/control cabinet).

(4) Portions of the non-1E heating and ventilating systems are required to ventilate the main control room, non-1E switchgear rooms, and the required portions of the non-1E instrumentation and control system (see note 3).

(5) The term "separation" means that fire barriers provide separation of redundant components, including equipment, electrical cables, instrumentation and controls, except in the main control room, the remote shutdown workstation and containment. See section 9A.3.1.1 for discussion on containment and section 9A.3.1.2.5 for discussions on the main control room and remote shutdown workstation.

(6) Separation is provided for the reactor trip function. The reactor trip breakers are separated.

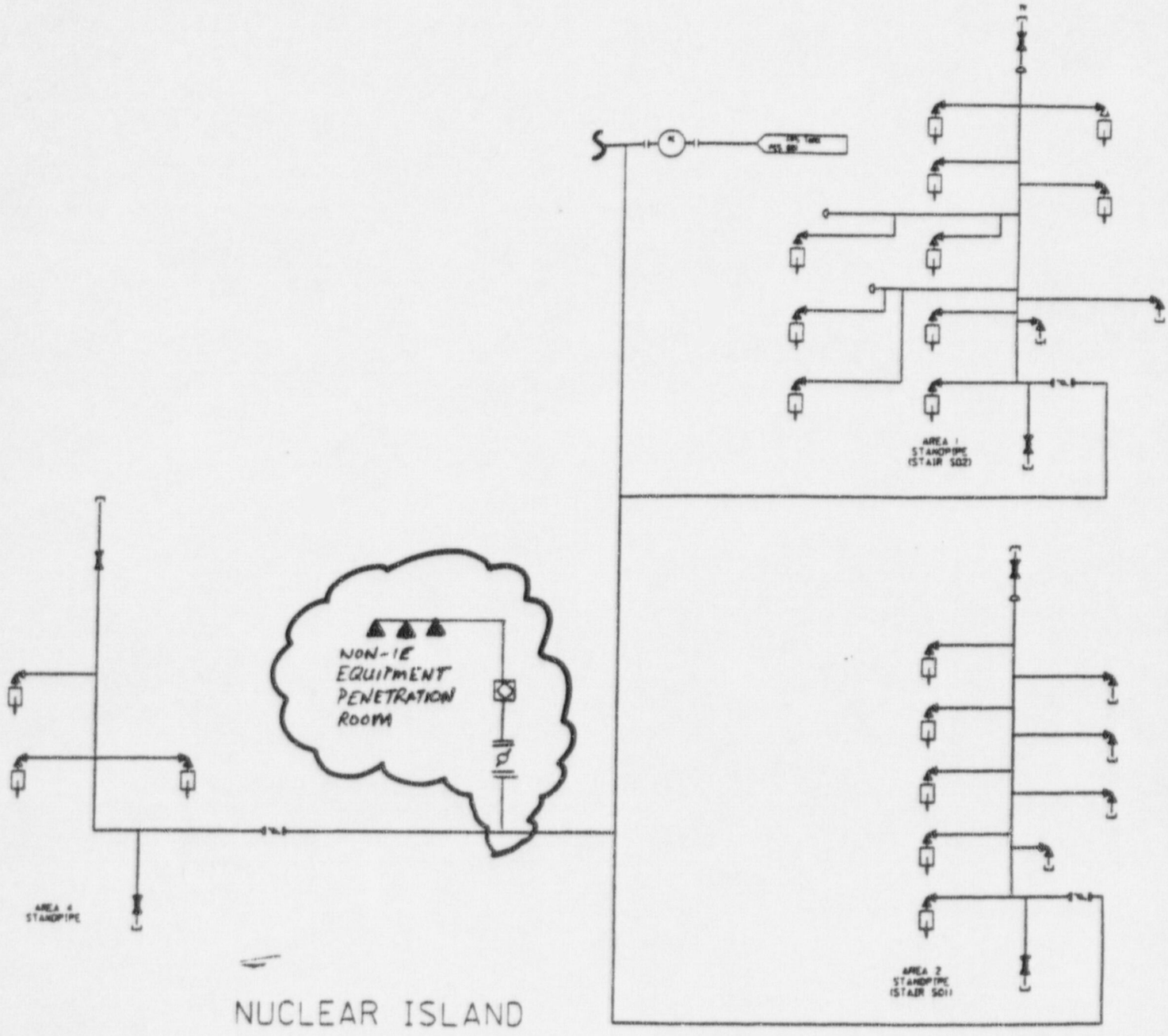


From: Figure 9.5.1-1 (Sheet 2 of 3)

Fire Protection System Piping and Instrumentation Diagram (REF FPS 002)

Revision: 18
December 5, 1997

FSERDI-9.5.1-1--7



From Figure 9.5.1-1 (Sheet 3 of 3)

Fire Protection System
Piping and Instrumentation Diagram

(REF) FPS 004

Revision: 17
October 31, 1997

FSER OI 9.5.1-1--8

9A.3.1.2.6 Non-Class 1E Electrical Rooms

9A.3.1.2.6.1 Fire Area 1230 AF 02

The fire area is comprised of the following room:

Room No.
12321 Non-Class 1E equipment/penetration room

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

Fire Detection and Suppression Features

- Fire detectors
- ***Preaction sprinklers***
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The equipment room HVAC subsystem of the annex/auxiliary building non-radioactive ventilation system (VXS) servicing this fire area stops upon detection of smoke in the supply duct. Combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to control the spread of fire and smoke. Other VXS subsystems continue to provide ventilation to the unaffected fire areas. This subsystem may be restarted and manually realigned to the once-through smoke exhaust ventilation mode to minimize the potential migration of smoke. If the exhaust fire-smoke damper for this fire area is operable, the damper may be reopened to further reduce the migration of smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers and operating the ventilation system in the smoke exhaust ventilation mode.

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~~ ***by the preaction sprinkler system or manually*** using hose streams or portable extinguishers.

Combustible materials in this fire area are listed in Table 9A-3, and primarily consist of cable insulation for cables associated with the electrical equipment and containment penetrations in this fire area. There are small concentrations of cable overhead, at the electrical penetrations, and at the electrical cabinets located along the east and west walls of room 12321. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. ***An automatic***

suppression system is provided to increase the availability of non-safety related systems required to achieve cold shutdown.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

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 *is considered in the evaluation of internal flooding in Section 3.4*. ~~An evaluation of the consequences of a break in a fire protection line is not required because no such lines pass through or terminate in this fire area.~~

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 B and D cables are routed through this area. In the event of a fire, the division B and D cabling in this area can be damaged. This damage can result in loss of control of equipment serviced by these cables. Other components in divisions B and D are not affected.

This fire can also disable the division B and D 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.

Spurious DAS actuation of squib valves is prevented by the use of a squib valve controller circuit which requires multiple hot shorts for actuation, physical separation of potential hot short locations, and provisions for operator action to remove power from the fire area. No postulated fire can spread to the hot short locations before the operator can remove power from the fire area.

Following detection of a fire in the non-Class 1E equipment/penetration room, the operators can close the automatic depressurization system stage 4 block valves, then remove DAS actuation power. This operator action will prevent spurious actuation of squib valves resulting from multiple hot shorts in the non-Class 1E equipment/penetration room.

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

9A.3.1.3.1.1 Fire Area 1200 AF 01

This fire area includes most of the radiologically controlled areas of the auxiliary building outside the fuel handling area. This fire area contains one of the two normal residual heat removal pumps, the liquid radwaste system, spent fuel pool cooling system, chemical and volume control system makeup pump, and lower annulus areas. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 1200 AF 12241	12241	Lower annulus east
• 1200 AF 12241	12242	Lower annulus southeast
• 1200 AF 12241	12243	Lower annulus southwest
• 1200 AF 12461	12461	Corridor
• 1210 AF 12151	12151	Demineralizer/filter room
• 1210 AF 12151	12155	Gaseous radwaste system equipment room
• 1210 AF 12151	12156	Liquid radwaste system equipment room
• 1210 AF 12151	12158	Degasifier discharge pump room
• 1210 AF 12151	12258	Degasifier column
• 1210 AF 12171	12171	Effluent holdup tank room A
• 1214 AF 12152	12152	Primary sample room
• 1214 AF 12154	12154	Auxiliary building sump room
• 1214 AF 12154	12254	Spent fuel pool cooling system penetration room
• 1215 AF 12161	12161	Corridor
• 1215 AF 12161	12163	Normal residual heat removal pump room B
• 1216 AF 12166	12166	Waste holdup tank room A
• 1216 AF 12167	12167	Waste holdup tank room B
• 1216 AF 12169	12168	Corridor
• 1216 AF 12169	12169	Corridor
• 1216 AF 12169	12268	Liquid radwaste system pump room
• 1216 AF 12264	12264	Chemical waste tank room
• 1216 AF 12169	12265	Waste monitor tank room C
• 1216 AF 12172	12172	Effluent holdup tank room B
• 1220 AF 12251	12251	Demineralizer/filter access area
• 1220 AF 12251	12255	Chemical and volume control system makeup pump room
• 1220 AF 12269	12259	Pipe chase
• 1220 AF 12271	12272	Spent fuel pool cooling system pump room A
• 1220 AF 12271	12273	Spent fuel pool cooling system heat exchanger room A
• 1220 AF 12271	12274	Spent fuel pool cooling system pump room B
• 1220 AF 12271	12275	Spent fuel pool cooling system heat exchanger room B
• 1224 AF 12252	12252	Radiation chemistry laboratory
• 1225 AF 12261	12261	Corridor
• 1225 AF 12261	12271	Liquid radwaste system pump room
• 1235 AF 12361	12361	Corridor
• 1250 AF 12561	12551	Access corridor
• 1250 AF 12561	12561	Component cooling water system valve room

The equipment and piping in this fire area normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- ***Wet pipe sprinklers (Fire Zone 1220 AF 12251, room 12255)***
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The radiologically controlled area ventilation system (VAS) serves this fire area on a once-through basis. In the event of a fire the system continues to ventilate the fire area unless the operator decides to shut down the system, or until heat from the fire is sufficient to close the fire dampers. Fire dampers close automatically on high temperature to control the spread of fire and smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers. 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 ***by the wet pipe sprinkler system or*** manually using hose streams or portable extinguishers.

Combustible materials in this large fire area are listed in Table 9A-3, and consist primarily of cable insulation for cables associated with the mechanical equipment and instrumentation in this fire area. There are small concentrations of lubricants in fire zones containing pumps. There are small concentrations of paper and plastic in the radiation chemistry laboratory fire zone. Concentrations of paper or plastic anti-contamination clothing may also be present in some fire zones. There are small concentrations of cable in the overhead cable trays in many fire zones. This is a light hazard fire area and the rate of fire growth is expected to be slow. Three-hour fire barriers provide adequate separation from adjacent fire areas and the fire is contained within the fire area. ***An automatic suppression system is provided to increase the availability of non-safety related systems required to achieve cold shutdown.***

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

The consequences of inadvertent operation of an automatic suppression system ***in this fire area or of fire suppression systems*** that drains to this fire area from the radwaste building; or of a break in a fire protection line in this fire area, are considered in the evaluation of internal flooding in 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 serviced by these cables. Other components in divisions A and C are not affected.

The spent fuel pool cooling system containment isolation valve is conservatively assumed to be disabled as a result of a fire in this fire area. The redundant spent fuel pool cooling system containment isolation valve located inside containment is outside of this fire area and is 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.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.



9A.3.1.3.1.2 Fire Area 1200 AF 02

This fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 1200 AF 12562	12462	Cask washdown pit
• 1200 AF 12562	12463	Cask loading pit
• 1200 AF 12562	12472	New fuel storage pit
• 1200 AF 12562	12563	Spent fuel storage pool
• 1200 AF 12562	12564	Fuel transfer canal
• 1230 AF 12371	12371	Rail car bay/filter storage area
• 1230 AF 12371	12374	Waste disposal container area
• 1236 AF 12372	12372	Resin transfer pump/valve room
• 1236 AF 12373	12373	Spent resin tank room
• 1246 AF 12471	12471	Solid waste system valve/piping area

The spent fuel storage pool, spent fuel handling systems and components, and the solid radwaste rooms normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- ***Wet pipe sprinklers (Fire Zone 1230 AF 12371, room 12371 rail car bay only)***
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The radiologically controlled area ventilation system serves this fire area on a once-through basis. In the event of a fire the system continues to ventilate the fire area unless the operator decides to shut down the system, or until heat from the fire is sufficient to close the fire dampers. Fire dampers close automatically on high temperature to control the spread of fire and smoke. After the fire, smoke is removed from the fire area by reopening the fire dampers. The radiologically controlled area ventilation system exhausts the smoke and hot gases 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 ***by the wet pipe sprinkler system or*** manually using hose streams or portable extinguishers.

Combustible materials in this large fire area are listed in Table 9A-3, and consist primarily of cable insulation for the cables associated with the mechanical equipment and instrumentation in this fire area.

There are small concentrations of lubricants associated with equipment such as the overhead cranes and the fuel handling machine. ***Diesel fuel may be present when there is a truck in the rail car bay.*** Concentrations of paper or plastic anti-contamination clothing may also be present. There are small concentrations of cable in the overhead cable trays. This is ***generally*** a light hazard fire area and the rate of fire growth is expected to be slow. ***Concentrations of transient combustibles in the rail car bay may produce a rapidly growing fire, so an automatic suppression system is provided in that area.*** Three-hour fire barriers provide adequate separation from adjacent fire areas. The roof and building exterior walls are unrated because there are no significant exposure fire hazards in nearby outdoor areas. ***Automatic or*** ~~Mm~~anual fire suppression activities prevent the fire from propagating beyond the fire area.

The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Fire Protection System Integrity

An evaluation of the consequences of inadvertent operation of an automatic suppression system ***or*** ~~is not required because there are no such systems in this fire area. The consequences~~ of a break in a fire protection line in this fire area ~~were~~ ***are*** considered in the evaluation of internal flooding in Section 3.4.

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 serviced by these cables. Other components 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.

No fire in this fire area can cause spurious actions which could cause a breach in the reactor coolant boundary or defeat safety-related decay heat removal capability or cause an increase in shutdown reactivity of the reactor.



9A.3.2.1 Fire Area 2000 AF 01

This fire area contains the main condenser, lubrication equipment, turbine-generator and auxiliaries, switchgear rooms, electrical equipment room, feedwater pumps, chemical feed equipment, chiller area, plant air compressors, digital-electrohydraulic skid, main steam piping, office area, and a sampling laboratory. The fire area is subdivided into the following fire zones:

Fire Zone	Room No.	
• 2030 AF 20300	20300	Elevation 100′-0″ (base slab) general floor area
• 2030 AF 20300	20306	Steam generator blowdown heat exchanger area
• 2030 AF 20300	20309	Circulating water pipe trench
• 2038 AF 20300	----	Main feedwater pump area
• 2039 AF 20301	20301	Chemical storage area
• 2040 AF 20400	20400	Elevation 117′-6″ general floor area
• 2040 AF 20400	20409	Condensate polishing area
• 2040 AF 20400	20410	Access to annex building
• 2040 AF 20400	20411	Access to auxiliary building lower nuclear island nonradioactive ventilation equipment room
• 2050 AF 20500	20500	Elevation 135′-3″ general floor area
• 2050 AF 20500	20512	Access to annex building
• 2050 AF 20500	20513	Access to auxiliary building main steam isolation valve compartment B
• 2050 AF 20500	20514	Access to auxiliary building upper nuclear island nonradioactive ventilation system equipment room
• 2050 AF 20500	20515	Access to auxiliary building main steam isolation valve compartment A
• 2050 AF 20502	----	Digital-electrohydraulic skid
• 2052 AF 20504	20510	HVAC equipment area
• 2053 AF 20505	20505	Office area at elevation 149′-0″
• 2053 AF 20506	20509	Variable frequency driver power converter room
• 2057 AF 20503	20511	Generator seal oil unit
• 2060 AF 20600	20600	Elevation 161′-0″ general floor area
• 2060 AF 20600	20603	Restroom
• 2060 AF 20600	20604	Restroom
• 2060 AF 20600	20605	Surge tank platform
• 2063 AF 20601	20601	Tool room/storage area
• 2063 AF 20602	20602	Office area/engineering workstation at elevation 171′-0″

This fire area designation consists of the entire turbine building floor areas except those special areas contained within fire rated enclosures and discussed below as separate fire areas. In zones within this area where there is a potential for spills of oil or other combustibles, such as the digital-electrohydraulic skid, condensate polishing area, chemical storage area and the generator seal oil unit, spill control curbs are provided to limit the spread of fire should one occur. There are no systems in this fire area which normally contain radioactive material.

Fire Detection and Suppression Features

- Fire detectors
- Automatic suppression for the oil spill areas around the turbine-generator, the generator seal oil unit, the main feedwater pump area, the digital-electrohydraulic skid, and the chemical storage area.

  Automatic suppression for the following equipment: the component cooling water pumps, the service water pumps, the start-up feedwater pumps and MCCs and control equipment at elevation 135'-3"(in the area defined by column 13.1 to 14 and P.1 to O).

  Automatic suppression is ~~also~~ provided over elevation 100'-0" down the corridor between the condenser and the main feedwater pumps defined by column 13.1 to column 18, continuing down another corridor defined by column L.2 to column K.5 ***and for the equipment access area in fire zone 2030-AF-20300, room 20300***

  See Table 9A-3 for identification of the specific types of automatic suppression systems for this fire area.

- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The general area heating and ventilation subsystem of the turbine building ventilation system (VTS), as described in SSAR subsection 9.4.9, uses roof mounted exhaust ventilators to pull in air through wall louvers. Dedicated roof mounted smoke and heat vents are also provided. The smoke and heat vents and, if available, the roof mounted exhaust ventilators, vent smoke to outside areas and prevent migration of smoke to adjacent fire areas. The dedicated smoke and heat vents provide additional assurance that excessive smoke and heat cannot buildup at the turbine building ceiling. The design of the smoke and heat vents is in conformance with the guidelines of NFPA 204M.

Fire Protection Adequacy Evaluation

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

Combustible materials in this fire area are listed in Table 9A-3. This area contains fire zones with a low quantity of combustible materials and other zones which have moderate to high quantities of combustible materials. Zones containing a low quantity of low heat release combustibles are generally



protected by manual suppression. Zones containing moderate to high quantities of high heat release combustibles, such as oil spill areas, are protected by automatic suppression.

The limited amount of combustible material or automatic fire suppression activities prevent fire from propagating to other fire zones in the area and prevent fire from propagating through the exterior walls or roof of the turbine building to other fire areas. The south end of the turbine building is formed by the 3-hour fire barrier exterior walls of the annex and auxiliary building. There are no concentrations of combustibles in the turbine building adjacent to these walls. Therefore, a turbine building fire will not propagate through these exterior walls of the annex or auxiliary building.

The turbine building ventilation system (VTS) serves the turbine building only and therefore does not contribute to the spread of the fire or smoke to fire areas outside the turbine building. The turbine building ventilation system does not contribute to the spread of fire or smoke to other fire areas within the turbine building because fire dampers isolate the other fire areas.

9A.3.4.12 Fire Area 4032 AF 02

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

Room No.	
40357	Containment access corridor elevation 107'-2"

This fire area is the corridor used to transport equipment and personnel through the annex building and auxiliary building to and from containment. As such, low levels of radioactive material may be present within the fire area.

Fire Detection and Suppression Features

- Fire detectors
- ***Wet pipe sprinklers (Elevation 107'-2")***
- Hose station(s)
- Portable fire extinguishers

Smoke Control Features

The auxiliary/annex building subsystem of the radiologically controlled area ventilation system (VAS) servicing this fire area alarms in the main control room upon detection of smoke in the area or in the supply or exhaust duct. Fire dampers close automatically on high temperature to isolate this fire area. The balance of the VAS auxiliary/annex building ventilation subsystem remains in operation unless plant operators determine that there is a need to manually shut down the subsystem. The balance of VAS remains in operation. These actions control the spread of fire and smoke. After the fire, smoke is removed from this fire area by reopening the fire dampers. The ventilation exhaust system serving the area exhausts smoke 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 ***by the wet pipe sprinkler system or*** manually using hose streams or portable extinguishers.

Combustible materials in this fire area are listed in Table 9A-3 and primarily consist of electrical cable insulation and ordinary combustible materials such as wood and paper. ***Diesel fuel may be present when there is a truck in the containment access corridor.*** Combustibles are relatively uniformly distributed throughout the fire area ***except when transient combustibles are present.*** ~~This is a light hazard fire area and the rate of fire growth is expected to be slow.~~ ***This is an ordinary hazard fire area when transient combustibles are present and the rate of fire growth may be rapid.*** Minimum two-hour fire barriers are provided and the building exterior wall is not rated. ***An automatic suppression system is provided to address transient combustibles.***

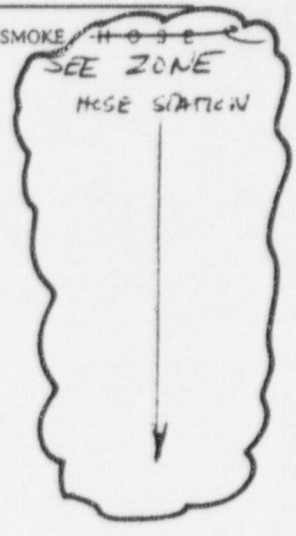


The ventilation system does not contribute to the spread of the fire or smoke as described in the Smoke Control Features section above.

Table 9A-3 (Sheet 2 of 25)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area[2][3]	Floor Area, Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[1] (Hours)	Detect. Cap.	Fixed Suppression Capability[3]
1100 AF 11500			CABLE INS	C	12000	1.2E+08					HOSE STATION
OPERATING DECK AND			PAPER	C	500	3.9E+06					
REFUELING CAVITY		VOLATILES		E	55	7.5E+06					
			LUBE OIL	E	10	1.5E+06					
		11150	NET CAT.	D	TOTAL:	1.4E+08	12000	9			
1200 AF 12341			CABLE INS	C	4000	4.1E+07					NONE
MIDDLE ANNULUS			RUBBER	D	1200	1.5E+07					
		1845	NET CAT.	D	TOTAL:	5.5E+07	30000	22			
1200 AF 12541			CABLE INS	C	500	5.1E+06					NONE
UPPER ANNULUS		1685	NET CAT.	C	TOTAL:	5.1E+06	3000	1			
1270 AF 12701			CABLE INS	C	500	5.1E+06					NONE
PCS VALVE ROOM		800	NET CAT.	C	TOTAL:	5.1E+06	6400	5			
1250 AF 12555			CABLE INS	C	4000	4.1E+07					HOSE STATION
VES AIR STORAGE/			PAPER	C	1000	7.7E+06					
OPERATING DECK			TRASH	B	1000	7.7E+06					
STAGING AREA			CLOTH	B	500	4.0E+06					
			WOOD	C	500	4.2E+06					
			PLASTIC	D	500	6.6E+06					
			RUBBER	D	100	1.2E+06					
			VOLATILES	E	10	1.4E+06					
		1200	NET CAT.	D	TOTAL:	7.4E+07	61000	49			
FIRE AREA TOTAL		27363	NET CAT.	D	TOTAL:	4.5E+08	16000	12			
1200 AF 01 STATION	YES								3	SMOKE	
1200 AF 12241			CABLE INS	C	200	2.0E+06					
LOWER ANNULUS		1800	NET CAT.	C	TOTAL:	2.0E+06	1100	1			
1200 AF 12461			CABLE INS	C	1000	1.0E+07					
CORRIDOR			TRASH	B	200	1.5E+06					
		480	NET CAT.	C	TOTAL:	1.2E+07	24000	19			
1210 AF 12151			CABLE INS	C	8000	8.2E+07					
DEMIN./FILTER.			LUBE OIL	E	10	1.5E+06					
WLS & WGS			VOLATILES	E	10	1.4E+06					
EQUIPMENT ROOMS		PAPER	C		100	7.7E+05					
		1890	NET CAT.	D	TOTAL:	8.5E+07	45000	34			
1210 AF 12171			CABLE INS	C	2000	2.0E+07					
EFFLUENT HOLDUP		PAPER	C		500	3.9E+06					
TANK ROOM A			VOLATILES	E	10	1.4E+06					
		785	NET CAT.	D	TOTAL:	2.6E+07	33000	25			

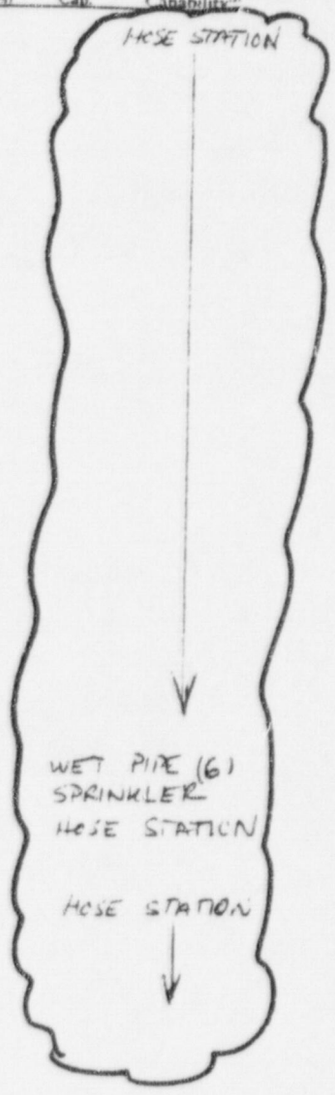


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Table 9A-3 (Sheet 3 of 25)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area[2,3]	Floor Area, Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[4] (Hours)	Detect. Cap.	Fixed Suppression Capability[5]
1214 AF 12152 PRIMARY SAMPLE ROOM			CABLE INS	C	500	5.1E+06					
			PAPER	C	100	7.7E+05					
			PLASTIC	D	200	2.6E+06					
		280	NET CAT.	D	TOTAL:	8.5E+06	30000	22			
1214 AF 12154 AUX. BLDG SUMP AND SFS PENETRATION ROOM			CABLE INS	C	200	2.0E+06					
		190	NET CAT.	C	TOTAL:	2.0E+06	11000	8			
1215 AF 12161 RNS PUMP ROOM B AND CORRIDOR			CABLE INS	C	1500	1.5E+07					
		LUBE OIL		E	5	7.6E+05					
		785	NET CAT.	D	TOTAL:	1.6E+07	20000	16			
1216 AF 12166 WASTE HOLDUP TANK ROOM A			CABLE INS	C	200	2.0E+06					
			VOLATILES	E	10	1.4E+06					
		280	NET CAT.	E	TOTAL:	3.4E+06	12000	9			
1216 AF 12167 WASTE HOLDUP TANK ROOM B			CABLE INS	C	200	2.0E+06					
			VOLATILES	E	10	1.4E+06					
		300	NET CAT.	E	TOTAL:	3.4E+06	11000	9			
1216 AF 12169 WLS PUMP ROOM/ CORRIDOR			CABLE INS	C	1000	1.0E+07					
			LUBE OIL	E	5	7.6E+05					
			VOLATILES	E	10	1.4E+06					
		475	NET CAT.	E	TOTAL:	1.1E+07	23000	17			
1216 AF 12172 EFFLUENT HOLDUP TANK ROOM B			CABLE INS	C	2000	2.0E+07					
		PAPER	C		500	3.9E+06					
			VOLATILES	E	10	1.4E+06					
		795	NET CAT.	D	TOTAL:	2.6E+07	32000	24			
1216 AF 12264 WASTE MONITOR TANK ROOM C & CHEM. WASTE TANK ROOM			CABLE INS	C	500	5.1E+06					
			VOLATILES	E	10	1.4E+06					
		660	NET CAT.	E	TOTAL:	6.5E+06	9800	7			
1220 AF 12251 DEMIN/FILTER ACCESS AREA AND CVS MAKEUP PUMP ROOMS			CABLE INS	C	7000	7.1E+07					
			PAPER	C	1000	7.7E+06					
			PLASTIC	D	200	2.6E+06					
		LUBE OIL		E	5	7.6E+05					
			VOLATILES	E	10	1.4E+06					
		1750	NET CAT.	D	TOTAL:	8.4E+07	48000	37			
1220 AF 12269 EL 92'-6" PIPE CHASE			CABLE INS	C	100	1.0E+06					
		800	NET CAT.	C	TOTAL:	1.0E+06	1300	1			
1220 AF 12271 SFS EQUIPMENT ROOMS			CABLE INS	C	5000	5.1E+07					
			LUBE OIL	E	10	1.5E+06					
			VOLATILES	E	10	1.4E+06					
		1190	NET CAT.	D	TOTAL:	5.4E+07	45000	34			



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Table 9A-3 (Sheet 4 of 25)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area[2,5]	Floor Area, Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[4] (Hours)	Detect. Cap.	Fixed Suppression Capability[5]
1224 AF 12252			CABLE INS	C	800	8.2E+06					HOSE STATION
RADIOACTIVE			PAPER	C	500	3.9E+06					
CHEMISTRY			PLASTIC	D	500	6.6E+06					
LABORATORY			VOLATILES	E	5	6.8E+05					
		285	NET CAT.	D	TOTAL:	1.9E+07	68000	55			
1225 AF 12261			CABLE INS	C	3000	3.1E+07					
WLS PUMP ROOM			LUBE OIL	E	5	7.6E+05					
AND CORRIDOR			VOLATILES	E	5	6.8E+05					
		865	NET CAT.	D	TOTAL:	3.2E+07	37000	28			
1235 AF 12361			CABLE INS	C	2000	2.0E+07					
CORRIDOR			TRASH	B	500	3.9E+06					
		480	NET CAT.	C	TOTAL:	2.4E+07	51000	39			
1250 AF 12561			CABLE INS	C	1000	1.0E+07					
CCS VALVE ROOM AND			VOLATILES	E	5	6.8E+05					
ACCESS CORRIDOR		630	NET CAT.	D	TOTAL:	1.1E+07	17000	13			
FIRE AREA TOTAL:		14720	NET CAT.	D	TOTAL:	4.3E+08	29000	22			
1200 AF 02	YES								3/0	SMOKE	~~HOSE STATION~~ SEE ZONE
1200 AF 12562			CABLE INS	C	10000	1.0E+08					HOSE STATION
FUEL HANDLING AREA			PAPER	C	1500	1.2E+07					
			WOOD	C	1000	8.4E+06					
			TRASH	B	1000	7.7E+06					
			CLOTH	B	500	4.0E+06					
			PLASTIC	D	500	6.6E+06					
			LUBE OIL	E	15	2.3E+06					
			VOLATILES	E	5	6.8E+05					
		4725	NET CAT.	D	TOTAL:	1.4E+08	30000	23			
1230 AF 12371			CABLE INS	C	3000	3.1E+07					WET PIPE (6) SPRINKLER HOSE STATION
RAIL CAR BAY/			PAPER	C	1000	7.7E+06					
FILTER STORAGE AREA			WOOD	C	1000	8.4E+06					
			TRASH	B	1000	7.7E+06					
			LUBE OIL	C	10	1.5E+06					
			VOLATILES	E	5	6.8E+05					
		1460	NET CAT.	D	TOTAL:	5.7E+07	39000	29			
1236 AF 12372			CABLE INS	C	200	2.0E+06					HOSE STATION
RESIN TRANSFER PUMP/			LUBE OIL	E	5	7.6E+05					
VALVE ROOM		80	NET CAT.	E	TOTAL:	2.8E+06	35000	26			
1236 AF 12373			CABLE INS	C	200	2.0E+06					HOSE STATION
SPENT RESIN TANK ROOM		70	NET CAT.	C	TOTAL:	2.0E+06	29000	22			
1246 AF 12471			CABLE INS	C	200	2.0E+06					HOSE STATION
WSS VALVE/PIPING AREA		90	NET CAT.	C	TOTAL:	2.0E+06	23000	17			
FIRE AREA TOTAL:		6425	NET CAT.	D	TOTAL:	2.1E+08	32000	24	2.2E+08	34000	26
1230 AF 12371			CABLE INS	C	3000	3.1E+07					
RAIL CAR BAY/			PAPER	C	1000	7.7E+06					
FILTER STORAGE AREA			WOOD	C	1000	8.4E+06					
			TRASH	B	1000	7.7E+06					
			LUBE OIL	E	10	1.5E+06					
			FUEL OIL	E	100	1.4E+07					
		1460	NET CAT.	D	TOTAL:	7.0E+07	48000	37			

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Table 9A-3 (Sheet 11 of 25)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area[2]	Floor Area, Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[5] (Hours)	Detect. Cap.	Fixed Suppression Capability[6]
1230 AF 01 STATION	YES								3	SMOKE	H O S E
CORRIDOR			CABLE INS	C	2500	2.6E+07					
FIRE AREA TOTAL:		770	NET CAT.	C	TOTAL:	2.6E+07	33000	24			
1230 AF 02 STATION	YES								3	SMOKE	H O S E
NON-1E EQUIPMENT/ PENETRATION ROOM			CABLE INS	C	3500	3.6E+07					
FIRE AREA TOTAL:		870	NET CAT.	C	TOTAL:	3.6E+07	41000	30			
1232 AF 01 STATION	YES								3	SMOKE	H O S E
REMOTE SHUTDOWN WORKSTATION			CABLE INS	C	1500	1.5E+07					
			PAPER	C	1000	7.7E+06					
			PLASTIC	D	500	6.6E+06					
FIRE AREA TOTAL:		410	NET CAT.	C	TOTAL:	3.0E+07	72000	63			
1240 AF 01 STATION	YES								3	SMOKE	H O S E
NON-1E EQUIPMENT/ PENETRATION ROOM			CABLE INS	C	3500	3.6E+07					
FIRE AREA TOTAL:		800	NET CAT.	C	TOTAL:	3.6E+07	45000	34			
1242 AF 01 STATION	YES								3	SMOKE	H O S E
1242 AF 12401A			CABLE INS	C	4000	4.1E+07					
MCR MAIN CONTROL			PAPER	C	3500	2.7E+07					
AREA/TAGGING			PLASTIC	D	1000	1.3E+07					
ROOM/VESTIBULE		1545	NET CAT.	C	TOTAL:	8.1E+07	52000	41			
1242 AF 12401B			CABLE INS	C	2000	2.0E+07					
MCR SHIFT SUPR/			PAPER	C	3500	2.7E+07					
CLERK/OPERATOR			PLASTIC	D	1000	1.3E+07					
AREAS		845	NET CAT.	C	TOTAL:	6.1E+07	72000	63			
FIRE AREA TOTAL:		2390	NET CAT.	C	TOTAL:	1.4E+08	59000	48			

PREVCTION SPRINKLER

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Table 9A-3 (Sheet 12 of 25)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area[2]	Floor Area, Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[4] (Hours)	Detect. Cap.	Fixed Suppression Capability[5]
1242 AF 02 STATION	YES								3	SMOKE	H O S E
DIVISION A ELECTRICAL PENETRATION ROOM			CABLE INS	C	2000	2.0E+07					
FIRE AREA TOTAL:		450	NET CAT.	C	TOTAL:	2.0E+07	45000	34			
1243 AF 01 STATION	YES								3	SMOKE	H O S E
REACTOR TRIP SWITCHGEAR 1			CABLE INS	C	500	5.1E+06					
FIRE AREA TOTAL:		95	NET CAT.	C	TOTAL:	5.1E+06	54000	42			
1243 AF 02 STATION	YES								3	SMOKE	H O S E
REACTOR TRIP SWITCHGEAR 2			CABLE INS	C	500	5.1E+06					
FIRE AREA TOTAL:		95	NET CAT.	C	TOTAL:	5.1E+06	54000	42			
1250 AF 01 STATION	NO								3	SMOKE	H O S E
VBS MCR/A&C EQUIPMENT ROOM			CABLE INS	C	12000	1.2E+08					
			CHARCOAL	C	5000	7.3E+07					
			LUBE OIL	E	20	3.0E+06					
			VOLATILES	E	10	1.4E+06					
FIRE AREA TOTAL:		3575	NET CAT.	D	TOTAL:	2.0E+08	56000	44			
2000 AF 01	NO								0	SEE ZONE	SEE ZONE
2030 AF 20300 ELEVATION 100'-0" SPRINKLERS (BASE SLAB) STATION GENERAL FLOOR AREA			CABLE INS	C	70000	7.1E+08				HEAT	WET PIPE SPRINKLERS (6)
			LUBE OIL	E	3000	4.5E+08					
			PLASTIC	D	10000	1.3E+08					H O S E
			VOLATILES	E	250	3.4E+07					
			TRASH	B	500	3.9E+06					
		32568	NET CAT.	E	TOTAL:	1.3E+09	41000	31			
2038 AF 20300 PREACTION MAIN FEEDWATER SPRINKLERS PUMP AREA			CABLE INS	C	2000	2.0E+07				HEAT	
			LUBE OIL	E	1500	2.3E+08					
			PLASTIC	D	100	1.3E+06					
			VOLATILES	E	55	7.5E+06					
			TRASH	B	200	1.5E+06					
		1550	NET CAT.	E	TOTAL:	2.6E+08	166000	124			

CABLE INS	C	70000	7.1E+08		
LUBE OIL	E	3000	4.5E+08		
PLASTIC	D	10000	1.3E+08		
VOLATILES	E	250	3.4E+07		
FUEL OIL	E	100	1.4E+07		
TRASH	B	500	3.9E+06		
NET CAT.	E	TOTAL:	1.4E+09	41000	31

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Table 9A-3 (Sheet 19 of 25)

FIRE PROTECTION SUMMARY

Fire Area/ Zone[1]	Safety Area[2]	Floor Area, Sq Ft	Combust. Material[3]	Fire Sev. Cat.	Amount	Heat Value (Btu)	Comb. Load, Btu/ Sq Ft	Equiv. Dur. (Min)	Boundary Fire Res.[4] (Hours)	Detect. Cap.	Fixed Suppression Capability[5]
4031 AF 05 STATION	NO								2/0	SMOKE	H O S E
4031 AF 40300 ACCESS CORRIDOR, ACCESS AREA, SECURITY ROOM #2			CABLE INS	C	1000	1.0E+07					
			PAPER	C	1200	9.2E+06					
			PLASTIC	D	500	6.6E+06					
			WOOD	C	500	4.2E+06					
			CLOTH	B	200	1.6E+06					
		1920	NET CAT.	D	TOTAL:	3.2E+07	17000	12			
4031 AF 40303 CORRIDOR AND RESTROOM			CABLE INS	C	2000	2.0E+07					
			PAPER	C	1000	7.7E+06					
			PLASTIC	D	500	6.6E+06					
			WOOD	C	500	4.2E+06					
			CLOTH	B	500	4.0E+06					
		1600	NET CAT.	D	TOTAL:	4.3E+07	27000	20			
FIRE AREA TOTAL:		3520	NET CAT.	D	TOTAL:	7.5E+07	21000	16			
4031 AF 06 STATION	NO								2	SMOKE	H O S E
CENTRAL ALARM STATION & SECURITY ROOM #1			CABLE INS	C	1075	1.1E+07					
			PAPER	C	200	1.5E+06					
			PLASTIC	D	175	2.3E+06					
			WOOD	C	500	4.2E+06					
			CLOTH	B	100	8.0E+05					
			TRASH	B	20	1.5E+05					
FIRE AREA TOTAL:		640	NET CAT.	D	TOTAL:	2.0E+07	31000	23			
4032 AF 01 STATION	NO								2/0	SMOKE	H O S E
HEALTH PHYSICS AREA			CABLE INS	C	4000	4.1E+07					
			WOOD	C	3400	2.9E+07					
			PLASTIC	D	500	6.6E+06					
			RUBBER	D	50	6.1E+05					
			CLOTH	B	1000	8.0E+06					
			PAPER	C	4000	3.1E+07					
			TRASH	B	400	3.1E+06					
			VOLATILES	E	10	1.4E+06					
FIRE AREA TOTAL:		6280	NET CAT.	D	TOTAL:	1.2E+08	19000	14			
4032 AF 02 STATION	NO								2/0	SMOKE	H O S E
CONTAINMENT ACCESS CORRIDOR			CABLE INS	C	4800	4.9E+07					
			PAPER	C	5000	3.9E+07					
			PLASTIC	D	200	2.6E+06					
			VOLATILES	E	50	6.8E+06					
FIRE AREA TOTAL:		2210	NET CAT.	D	TOTAL:	9.7E+07	44000	33			

WET PIPE SPRINKLER

CABLE INS	C	4800	4.9E+07		
PAPER	C	5000	3.9E+07		
PLASTIC	D	200	2.6E+06		
FUEL OIL	E	100	1.4E+07		
VOLATILES	E	50	6.8E+06		
NET CAT.	D	TOTAL:	1.1E+08	50000	39

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Table 9A-3 (Sheet 25 of 25)

FIRE PROTECTION SUMMARY

Notes:

1. The first four digits of the fire area and fire zone numbers indicate the building, level and building area in which the fire area/zone is located. When the third or fourth digit is a zero, the fire area/zone spans more than one level or building area. The last two digits in a fire area number are a sequence number only. The last five digits in a fire zone number coincide with the room number of a prominent room in the fire zone.

2. A YES indication in the Safety Area column means that one or more safety-related components are located in the fire area.

3. Estimated quantities of combustible materials are shown. Where the presence of transient combustibles is anticipated, their presence is indicated by the listing of volatiles or trash. The units and heat of combustion values for the combustible materials are shown in Table 9A-1.

4. The boundary fire resistance for each fire area represents the minimum resistance, in hours, for the surrounding walls, floor, and ceiling, except that:

   - A non-rated barrier capable of qualifying as a three-hour barrier is considered to have a resistance of three hours, provided that penetrations are adequately sealed.
   - Stairwells, elevator shafts and the like, which are enclosed by two-hour (minimum) fire barrier walls, may comprise a portion of the boundary of a fire area having a three hour resistance.
   - Building exterior walls below grade (soil on the outside) are considered to have a fire resistance of at least three hours even though they are not fire-rated.

   A boundary fire resistance designation such as "3/0" indicates that part of the fire area boundary consists of nonrated building exterior walls/roof above grade, but that the minimum resistance of the fire barriers separating the fire area from adjacent fire areas is three hours. For detailed information about the fire resistance of fire area boundaries see Figures 9A-1 through 9A-6.

5. The fixed suppression capability is indicated for each fire area. Unless otherwise indicated, sprinkler and spray systems are automatic. Where a hose station is indicated, one or more hose streams are available. In addition to fixed suppression capability, portable extinguishers are provided throughout the plant.

6. Partial suppression coverage in this zone. See SSAR 9A.3 Fire Protection Analysis results for details.

FSER01 9.5.1-1-27



Changes to SSAR 3.4.1.2.2.2:

Auxiliary Building Level 2 (Elevation 82′-6″)

- **Nonradiologically Controlled Area**

  Level 2 of the nonradiologically controlled area has two Class 1E battery rooms, four divisional Class 1E dc electrical equipment rooms, and one Class 1E reactor coolant pump trip switchgear room. The doors to these rooms are not water tight.

  Level 2 contains an arrangement of fire protection and potable water piping similar to level 1.

  The potential for flooding on this level is limited to fire fighting activities. Fire fighting in these rooms is accomplished by manual means from two fire hose stations located adjacent to the two stairwells. The maximum flowrate to this area from the two hose stations is assumed to be 250 gpm.

  The drains, elevator shafts, and stairwells drain water spilled on this level to level 1. Therefore, no significant accumulation of water occurs on level 2.

- **Radiologically Controlled Area**

  The radiologically controlled area on level 2 contains a few containment isolation valves. The horizontal pipe chase at elevation 92′-6″ contains two normally closed normal residual heat removal system isolation valves. One spent fuel pool cooling system containment isolation valve is located, above 92′-6″, in the adjacent vertical pipe chase. The area on the north side of the lower annulus contains two chemical and volume control system and two liquid radwaste automatically operated containment isolation valves above elevation 82′-6″. These valves are required to close or remain closed during a safe shutdown operation.

  Two chemical and volume control system valves used to isolate the chemical and volume control system makeup pump suction from the demineralized water storage tank are located in the makeup pump compartment at 82′-6″. These safety-related valves close or remain closed to prevent boron dilution events. They are not required for safe shutdown.

  Potential sources of flooding for this area include the chemical and volume control system and the fire protection system, including an automatic suppression system in the CVS makeup pump room. Flow from a component rupture or from fire fighting activities on level 2 drains to level 1 as described below.

  To protect the above valves from flooding, the makeup pump compartment at elevation 82′-6″ drains, via the floor grating located in the corridor adjacent to the stairwell, directly to elevation 66′-6″. Flooding in the lower annulus drains directly to elevation 66′-6″ via the floor grating and various openings to the tank rooms. The stair and elevator shaft on the east wall are additional

flow paths to level 1. The horizontal pipe chase at elevation 92′-6″ drains under the door directly to elevation 66′-6″ via the vertical pipe chase. As a result of these drain paths, there is no significant accumulation of water in the makeup pump compartment, lower annulus or the horizontal pipe chase from any postulated pipe ruptures. The containment isolation valves are above the maximum flood level in these areas. The chemical and volume control system makeup pumps and the normal residual heat removal valves in the valve compartment are nonsafety-related defense-in-depth equipment and are expected to remain functional following the flooding event since the pump motors and valve operators are above the calculated flood level of 6 inches.

Auxiliary Building Level 3 (Elevation 100′-0″)

- **Nonradiologically Controlled Area**

Level 3 of the nonradiologically controlled area includes the remote shutdown workstation room, one reactor coolant pump trip switchgear room, four divisional Class 1E I&C rooms, one equipment room, and the valve/piping penetration room. The division A, B, C and D I&C rooms and the electrical room also include containment electrical penetrations. The doors are not water tight.

The level 3 Class 1E and non-Class 1E electrical areas contains only fire protection system piping. Fire hose stations are provided near each of the two stairwells ***and normally dry fire protection piping, supplied from the passive containment storage tank, serves the preaction sprinkler system in the non-1E equipment/penetration room***. The potential for flooding in the electrical areas on this level is limited to fire fighting activities. ***The maximum flowrate to this area from either automatic or manual fire fighting activities is assumed to be 250 gpm.*** The floor drains, stairwells, and elevator shaft drain water spilled on this level down to level 1. Therefore, no significant accumulation of water occurs in this area.

The valve/piping penetration room on level 3 is physically separated from the electrical rooms. The valve/piping penetration room contains automatically actuated containment isolation valves for the steam generator blowdown system and the hydrogen line in the chemical and volume control system. Access to this room is from the turbine building. The access door and drain lines provided in this room drain from the auxiliary building to the turbine building. Maximum postulated flood level for this room is less than 36 inches. The containment isolation valves in the area are located above this maximum flood level.

- **Radiologically Controlled Area**

There are no safe shutdown components located on level 3.

Potential sources of flooding for this area include the normal residual heat removal system, the component cooling water system, ~~and~~ the effluent monitor tanks ***and the fire protection system, including an automatic suppression system in the rail car bay***. Flow from a component rupture ***or from fire fighting activities*** on level 3 drains directly to level 1.



Changes to SSAR 3.4.1.2.2.3:

Annex Building

- **Nonradiologically Controlled Areas**

The primary sources of flooding in the nonradiologically controlled areas of the annex building are the component cooling water, chilled water and fire protection systems. Water from postulated breaks above elevation 100'-0" flows primarily through floor drains to the annex building sump that discharges to the turbine building drain tank. Alternate paths include flows to the turbine building via flow under access doors at elevations 135'-3" and 117'-6" and flows down to elevation 100'-0" via stairwells and elevator shaft. Water accumulation at elevation 100'-0" is minimized by floor drains to the annex building sump and by flow under the access doors to the turbine building or the doors leading directly to the yard area. The floors of the annex building are sloped away from the access doors to the nuclear island in the vicinity of the access doors to prevent migration of flood water to the nonradiologically controlled areas of the nuclear island.

There is no safety-related equipment in the nonradiologically controlled area portion of the annex building. The main ac power system components with regulatory treatment of nonsafety-related systems important missions are located on elevation 117'-6" in the electrical switchgear rooms, which are separated from potential flood sources. Water from manual fire fighting operations is collected by floor drains discharging to the annex building sump or down a hatch or stairwell to elevation 100'-0". The non-Class 1E dc and UPS system (EDS) equipment with regulatory treatment of nonsafety-related systems important missions is located on elevation 100'-0" in separate battery rooms. Water in one of these rooms due to manual fire fighting in the room is collected by floor drains to the annex building sump or flows to the turbine building under doors or to the yard area through doors. This is not expected to affect functionality of equipment in the adjacent rooms.

- **Radiologically Controlled Areas**

There is no safety-related equipment in the radiologically controlled area portion of the annex building. The primary sources of flooding in the radiologically controlled areas of the annex building are the component cooling water, chilled water and fire protection systems, ***including an automatic sappression system that protects the containment access corridor***. Water from postulated breaks above elevation 100'-0" drains through floor drains to ***the radioactive waste drain system sump in the radiologically controlled area of*** ~~a waste holdup tank of the liquid radwaste system in~~ the auxiliary building or drains to elevation 100'-0" via stairwells and equipment handling hatches or under access doors to the radiologically controlled area portion of the auxiliary building. Accumulated water at elevation 100'-0" is minimized by floor drains discharging to ~~a waste holdup tank~~***the radioactive waste drain system sump*** or chemical waste tank in the auxiliary building. The contribution of water to the flooding of the radiologically controlled area portion of the auxiliary building is bounded by flooding events which could occur in the auxiliary building.

FSER Open Item 9.5.1-4 (OITS 6621)

The applicant has not fully evaluated the smoke control capability of the normal ventilation system against the criteria specified in NFPA 92A, "Recommended Practice for Smoke Control Systems," concerning stair tower pressurization in the auxiliary building. Specifically the applicant has not specified the minimum design pressure difference across the doors in the stair towers to be maintained in accordance with the guidance specified in NFPA 92A. Therefore, the applicants resolution of this issue is unacceptable.

Response:

A. Westinghouse and the NRC met on 02/24/98 and Westinghouse agreed to add stairwell pressurization to maintain the difference across the doors in the auxiliary building stairwells S01 and S02 in accordance with the guidance of NFPA 92A.

B. This RAI response also addresses an NRC question regarding the use of backdraft dampers to slow down smoke migration from one fire zone to another in each divisional non-1E equipment room and its corresponding I&C room. (Re: FSER Open Item 9.5.1-3)

   The backdraft dampers are equivalent to smoke dampers for the intended function. Attachment B contains pertinent information showing that the backdraft dampers meeting ASME N509 Leakage Class II requirements provide leakage rates similar to that of smoke dampers. A SSAR change will be made to clearly state the backdraft dampers will meet N509 Leakage Class II requirements. (See item B in "SSAR Revision" below.)

   An added benefit of using backdraft dampers in lieu of smoke dampers for this application is that the HVAC system can be operated in all modes without the need to reenter the fire area to reset the dampers.

SSAR Revision:

A. See Attachment A for revisions to SSAR 9A and 9.4.1 for stairwell pressurization.

B. Revisions to SSAR to address the use of backdraft dampers for divisional non-1E equipment rooms and corresponding I&C rooms:

   Change SSAR section 9.4.1.2.2 as noted on the following page.



Change to SSAR section 9.4.1.2.2:

Shutoff ~~and~~ Balancing *and Backdraft* Dampers

Multiblade, two-position remotely operated shutoff dampers are parallel-blade type. Multiblade, balancing dampers are opposed-blade type. ***Backdraft dampers are of the counterbalanced type and are provided to delay smoke migration through ductwork in case of fire. The backdraft dampers meet the Leakage Class II requirements of ASME N509 (Reference 2).*** 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.

ATTACHMENT A

9A.3.1.2.8.2 Fire Area 1201 AF 01

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

Room No.	
S02	Stairwell

The stairwell serving the northwest portion of the auxiliary building is enclosed by fire barrier walls having a minimum rating of 2 hours. The structural walls are concrete and the nonstructural walls are made of gypsum boards. There are no safe shutdown components and no radioactive systems in this fire area. The quantity of combustible materials in the stairwell is negligible and no fire is postulated in this fire area. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

This fire area is served by the division B & D Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Air at a positive pressure is supplied near the top of the stairwell and exfiltrates through an opening near the bottom of the stairwell. ~~For fires outside the stairwell in fire areas where the supply and exhaust penetrations are located, combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to prevent the spread of fire and smoke into the stairwell. For fires not affecting the ventilation system, the system continues to provide ventilation to the stairwell, and may be manually realigned to the once-through ventilation mode to further minimize the potential for migration of smoke and hot gases.~~ ***During a fire, the pressure difference across the doors in the stairwell is maintained in accordance with the guidance of NFPA 92A (Reference 4), using a dedicated stairwell pressurization fan.***



9A.3.1.2.8.3 Fire Area 1202 AF 01

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

Room No.	
S01	Stairwell

The stairwell serving the northeast portion of the auxiliary building is enclosed by fire barrier walls having a minimum rating of 2 hours. The structural walls are concrete and the nonstructural walls are made of gypsum board. There are no safety-related components and no radioactive systems in this fire area. The quantity of combustible materials in the stairwell is negligible and no fire is postulated in this fire area. NFPA Class I hose connections are provided in the stairwell for use in fighting fires in other fire areas.

This fire area is served by the division A & C Class 1E electrical room HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS). Air at a positive pressure is supplied near the top of the stairwell and exfiltrates through an opening near the bottom of the stairwell. ~~For fires outside the stairwell in fire areas where the supply and exhaust penetrations are located, combination fire-smoke dampers close automatically in response to a smoke detector signal or high temperature to prevent the spread of fire and smoke into the stairwell. For fires not affecting the ventilation system, the system continues to provide ventilation to the stairwell, and may be manually realigned to the once-through ventilation mode to further minimize the potential for migration of smoke and hot gases.~~ ***During a fire, the pressure difference across the doors in the stairwell is maintained in accordance with the guidance of NFPA 92A (Reference 4), using a dedicated stairwell pressurization fan.***

9A.4 References

1. NUREG-0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Section 9.5.1, "Fire Protection Program," Revision 3, July 1981, including Branch Technical Position (BTP) CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 2, July 1981.

2. Fire Protection Handbook, Edited by A. E. Cote, National Fire Protection Association, 16th edition.

3. NRC Generic Letter 81-12, February 20, 1981.

4. NFPA 92A-1993, "Recommended Practice for Smoke Control Systems"



9.4.1.2.1.2 Class 1E Electrical Room HVAC Subsystem

The Class 1E electrical room HVAC subsystem serves the Class 1E electrical rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, spare Class 1E battery room, remote shutdown area and reactor coolant pump trip switchgear rooms. The A and C electrical divisions, spare battery room, and reactor coolant pump trip switchgear rooms are served by one ventilation subsystem; the B and D electrical divisions and remote shutdown workstation area are served by a second ventilation subsystem.

Each subsystem consists of two 100 percent capacity supply air handling units, return/exhaust air fans, associated dampers, controls and instrumentation, and common ductwork. The supply air handling units and return/exhaust air fans are connected to a common ductwork which distributes air to the Class 1E electrical rooms. The outside supply air intake enclosure for the A and C subsystem is common to the main control room/technical support center intake located on the roof of the auxiliary building at elevation 153′-0″. The outside supply air intake for the B and D subsystem is located separate from the main control room/technical support center air intake enclosure on the auxiliary building roof at elevation 153′-0″. The exhaust ducts from the battery rooms are connected to the turbine building vent to remove hydrogen gas generated by the batteries.

The HVAC equipment which serves the A and C electrical divisions is located in the nuclear island nonradioactive ventilation system main control room/A and C equipment room at elevation 135′-3″ in the auxiliary building. The HVAC equipment which serves the B and D division of Class 1E electrical equipment is located in the upper and lower nuclear island nonradioactive ventilation system B and D equipment rooms at elevation 117′-0″ and at elevation 135′-3″.

The supply air handling unit cooling coils are provided with chilled water from the air-cooled chillers in the central chilled water system. The two air handling units for each set of electrical divisions are provided with chilled water from redundant air-cooled chillers. Refer to subsection 9.2.7 for the chilled water system description.

Each subsystem for the Class 1E battery rooms is provided with two 100 percent capacity exhaust fans.

The Class 1E electrical room HVAC subsystem is designed so that smoke, hot gases, and fire suppressant does not migrate from one fire area to another to the extent that they could adversely affect safe shutdown capabilities, including operator actions. Separate ventilation subsystems are provided to serve the electrical division A and C equipment rooms and the electrical division B and D equipment rooms. The use of separate HVAC distribution subsystems for the redundant trains of electrical equipment prevents smoke and hot gases from migrating from one distribution division to the other through the ventilation system ducts. In addition, combination fire/smoke dampers are provided for Class 1E equipment rooms, including the remote shutdown workstation room, to isolate each fire area and block the migration of smoke and hot gases to or from adjacent fire areas in accordance with NFPA 90A requirements. These combination fire/smoke dampers close in response to smoke detector signals or in response to the heat from a fire. ***During a fire, the pressure difference across the doors in the stairwells S01 and S02 is maintained in accordance with***

the guidance of NFPA 92A (Reference 32) by dedicated stairwell pressurization fans. See Appendix 9A for identification of fire areas.



9.4.1.2.3.2 Class 1E Electrical Room HVAC Subsystem

The Class 1E electrical room HVAC equipment that serves electrical division A and C equipment is described in this section. The operation of the Class 1E electrical room HVAC equipment that serves electrical division B and D is similar.

Normal Plant Operation

During normal plant operation, one of the redundant supply air handling units, return fans, and battery room exhaust fans operate continuously to provide room temperature control, to maintain the Class 1E electrical room emergency passive heat sink below its initial ambient air temperature, and to purge and prevent build-up of hydrogen gas concentration in the Class 1E Battery Rooms. The temperature of the air supplied by each air handling unit is controlled by temperature sensors located in the return air duct to maintain the room air temperature within the normal design range by modulating electric heating or chilled water cooling.

During normal plant operation, the exhaust airflow from the Class 1E battery rooms is vented directly to the turbine building vent to limit the concentration of hydrogen gas in the rooms to less than 2 percent by volume in accordance with the guidelines of Regulatory Guide 1.128.

The outside makeup air to the supply air handling units is provided through an outside air intake duct. The outside airflow rate is manually balanced during system startup to provide adequate makeup air for the battery room exhaust fans.

The standby supply air handling unit and the corresponding return/exhaust fans are started automatically if one of the following conditions occurs:

- Airflow rate of the operating fan is above or below predetermined set points
- Return air temperature is above or below predetermined setpoints.
- Loss of electrical and/or control power to the operating unit.

Abnormal Plant Operation

The operation ~~of the~~ Class 1E electrical room HVAC subsystem is not affected by the detection of airborne radioactivity in the main control room supply air duct of the main control room/technical support center HVAC subsystem. During a design basis accident (DBA), if the plant ac electrical system is unavailable, the Class 1E electrical room passive heat sink provides area temperature control. Refer to Section 6.4 for further details.

If a high concentration of smoke is detected in the outside air intake and an alarm is initiated in the main control room, the Class 1E electrical HVAC subsystem(s) can be manually aligned to the recirculation mode by closing the outside air intake damper to the air handling unit mixing plenum. This allows 100 percent

room air to return to the supply air subsystem air handling unit. The subsystem continues to provide cooling, ventilation, and temperature control to maintain the areas served by the subsystem(s) within their design temperatures and pressures.

In the event of a fire in a Class 1E electrical room, in response to heat from the fire or upon receipt of a smoke signal from an area smoke detector, the combination fire/smoke dampers close automatically to isolate the fire area. The affected subsystem continues to provide ventilation/cooling to the remaining areas and maintains the remaining areas at a slightly positive pressure. Either or both subsystems can be manually realigned to the once-through ventilation mode to supply 100 percent outside air to the unaffected areas. Realignment to the once-through ventilation mode minimizes the potential for migration of smoke and hot gases from a non-Class 1E electrical room or a Class 1E electrical room of one division into the Class 1E electrical room of another division. Smoke and hot gases can be removed from the affected areas by reopening the closed combination fire/smoke dampers from outside of the affected fire area during the once-through ventilation mode. In the once-through ventilation mode, the outside air intake damper to the air handling unit mixing plenum opens and the return air damper to the air handling unit closes to allow 100 percent outside air to the supply air handling unit. The subsystem exhaust air isolation damper also opens to exhaust room air directly to the turbine building vent. ***During a fire, the pressure difference across the doors in stairwells S01 and S02 is maintained in accordance with the guidance of NFPA 92A (Reference 32) by dedicated stairwell pressurization fans.***

The power supplies to the Class 1E electrical room HVAC subsystem are provided by the plant ac electrical system and the onsite standby diesel generators. In the event of a loss of the plant ac electrical system, the Class 1E electrical room HVAC subsystem is automatically transferred to the onsite standby diesel generators.

When complete ac power is lost, division B and C instrumentation and control room temperature is maintained by operating their respective ancillary fans to supply outside air to the I&C rooms. It is expected that outside air will be supplied within 72 hours following a radiological release. The outside air pathway to the ancillary fans is through the nonradioactive ventilation system outside air intake opening located on the roof, the mechanical room at floor elevation 135'-3", stairway No. 1 doors at elevation 135'-3" and 82'-6", the access corridor at floor elevation 82'6", and the divisional battery rooms. The warm air is vented to the annex building through the clean access corridor at elevation 100'-0". The outside air supply provides cooling and maintains room temperature below the qualification temperature of the I&C equipment. The ancillary fans and flow path are located within the auxiliary building which is a Seismic Category I structure.

Power supply to the ancillary fans is from the respective division B or C regulating transformers which receive power from the ancillary diesel generators. For post-72-hours power supply discussion see subsection 8.3.1.1.1.

9.4.13 References

1. "Functional Criteria For Emergency Response Facilities," USNRC NUREG 0696.
2. "Nuclear Power Plant Air-Cleaning Units and Components," ASME N509-1989.
3. "Testing of Nuclear Air-Cleaning Systems," ASME N510-1989.
4. "Laboratory Method of Testing Fans for Rating Purposes," ANSI/AMCA 210-85.
5. "Certified Ratings Program Air Performance," ANSI/AMCA 211-85.
6. "Reverberant Room Method of Testing Fans For Rating Purposes," ANSI/AMCA 300-85.
7. "Methods of Testing Air-Cleaning Devices Used in General Ventilation for Removing Particulate Matter," ASHRAE 52-76.
8. "Test Performance of Air-Filter Units," UL-900, 1986.
9. "High-Efficiency, Particular, Air-Filter Units," UL-586, 1985.
10. "Electric Central Air Heating Equipment," UL-1096, 1986.
11. "Methods of Testing for Rating Forced Circulation Air Cooling and Air Heating Coils," ASHRAE 33-78.
12. "Forced-Circulation Air Cooling and Air Heating Coils," ANSI/ARI 410-91.
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.

30. "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Regulatory Guide (RG) 1.140-1979, Revision 1.

31. "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants," Regulatory Guide 1.128, Revision 1, October 1978."

32. NFPA 92A-1993, "Recommended Practice for Smoke Control Systems"

AP600

ATTACHMENT B

Leakage Characteristic of VBS Backdraft Dampers and Combination Fire-Smoke Dampers

1. Backdraft Dampers

VBS backdraft dampers are designed to meet the Leakage Class II requirements of ASME N509 (Reference VBS SSD, VBS-M3-001, Rev. 3), which are:

Damper Blade Length or Diameter, inch	Permissible leakage (scfm/sq ft of damper face area) at 1 inch w.g. differential pressure for Leakage Class II
12	15
24	10
36	8
48	8

VBS backdraft dampers in the Class 1E electrical rooms are in the 12" range.

2. Combination Fire-Smoke Dampers

VBS combination fire-smoke dampers are designed to meet the requirements of UL-555S (Reference VBS SSD, VBS-M3-001, Rev. 3). No leakage classification has been established for these dampers, and there are no recommended classifications in NFPA 90A or NFPA 92A. UL leak classifications are:

Classification	Permissible leakage (cfm/sq ft of damper face area) at 1 inch w.g. differential pressure
I	4
II	10
III	40
IV	60

3. Conclusion

The permissible leakage of the existing VBS backdraft dampers (meeting ASME N509 Leakage Class II criteria) is well within the range of allowable leakage for smoke dampers.

FSER Open Item 9.5.1-5 (OITS 6622)

New Fuel Storage Area
The new fuel storage pit is provided with automatic fire detection, hose stations and portable extinguishers; automatic suppression has not been provided. Floor drains are provided to prevent the accumulation of water that could result in an inadvertent criticality. The new fuel storage pit is located in the same fire area (1200 AF 02) as the rail car bay/filter storage area. The rail car bay/filter storage area has not been provided with automatic suppression. The applicant has not included the potential for transient vehicles in the fire protection summary Table 9A-3 of the SSAR. The potential for transient vehicles in the rail car bay coupled with the other combustibles located in this fire area represents a significant exposure hazard to the new fuel storage area and is, therefore, unacceptable.

Spent Fuel Pool Area
The fuel handling area is provided with automatic fire detection, hose stations and portable extinguishers, automatic suppression has not been provided. The fuel handling area is located in the same fire area (1200 AF 02) as the rail car bay/filter storage area. The rail car bay/filter storage area has not been provided with automatic suppression.

Materials Containing Radioactivity
Materials that collect and contain radioactivity such as spent resins, charcoal filters, and HEPA filters are stored in closed metal containers. However, materials containing radioactivity are located in fire area 1200 AF 02, which contains numerous ignition sources and combustibles. Therefore, the radioactive materials located in this fire area are not adequately protected from an exposure fire. Therefore, the fire protection provided for radioactive materials is unacceptable.

Response:

Westinghouse agrees to add automatic fire suppression in the truck bay areas to close this issue.

SSAR Revision:

See attachment to Response to FSER Open Item 9.5.1-1.

Enclosure 2 to Westinghouse
Letter DCP/NRC1286

March 6, 1998

The reactor coolant loop is surrounded by structural walls of the containment internal structures. These structural walls are a minimum of 2-feet - 6-inches thick and enclose the reactor vessel, steam generators, reactor coolant pumps, and the pressurizer.

The containment vessel is designed and constructed in accordance with the ASME Code, Section III, Subsection NE, Metal Containment, including Addenda through 1989, as described in subsection 3.8.2.

Structural steel non-pressure retaining parts such as ladders, walkways, and handrails are designed to the requirements for steel structures defined in subsection 3.8.4.

The design features provide adequate containment sump levels following a design basis event as described in subsection 3.4.

Containment and subcompartment atmospheres are maintained during normal operation within prescribed pressure, temperature, and humidity limits by means of the containment air recirculation system (VCS), and the central chilled water system (VWS). The recirculation system cooling coils are provided with chilled water for temperature control. The filtration supply and exhaust subsystem can be utilized periodically to purge the containment air for pressure control. Periodic inspection and maintenance verify functional capability.

6.2.1.1.3 Design Evaluation

The Westinghouse-GOTHIC (WGOTHIC) computer code (Reference 20 and 32) is a computer program for modeling multiphase flow in a containment transient analysis. It solves the conservation equations in integral form for mass, energy, and momentum for multicomponent flow. The momentum conservation equations are written separately for each phase in the flow field (drops, liquid pools, and atmosphere vapor). The following terms are included in the momentum equation: storage, convection, surface stress, body force, boundary source, phase interface source, and equipment source.

To model the passive cooling features of the AP600, several assumptions are made in creating the plant decks. The external cooling water does not completely wet the containment shell, therefore, both wet and dry sections of the shell are modeled in the WGOTHIC analyses. The analyses use conservative coverage fractions to determine evaporative cooling.

Heat conduction from the dry to wet section is considered in the analysis. The combination of passive containment cooling system coverage area and heat conduction from the dry to wet sections is explained in Chapter 7 of Reference 20. The analyses conservatively assume that the external cooling water is not initiated until 337 seconds into the transient, allowing time to initiate the signal and to fill the headers and weirs and to develop the flow down the containment side walls. The effects of water flowing down the shell from gravitational forces are explicitly considered in the analysis.

The containment initial conditions of pressure, temperature, and humidity are provided in Table 6.2.1.1-2.

For the LOCA events, two double-ended guillotine reactor coolant system pipe breaks are analyzed. The breaks are postulated to occur in either a hot or a cold leg of the reactor coolant system. The hot leg break results in the highest blowdown peak pressure. The cold leg break results in the higher post-blowdown peak pressure. The cold leg break analysis includes the long term contribution to containment pressure from the sources of stored energy, such as the steam generators. The LOCA mass and energy releases described in subsection 6.2.1.3 are used for these calculations.

For the MSLB event, a representative pipe break spectrum is analyzed. Various break sizes and power levels are analyzed with the WGOTHIC code. The MSLB mass and energy releases described in subsection 6.2.1.4 are used for these calculations.

The results of the LOCA and MSLB postulated accidents are provided in Table 6.2.1.1-3. A comparison of the containment integrity analyses results to General Design Criterion 38 and the Acceptance Criteria presented in Reference 25 are also provided in Table 6.2.1.1-3. ~~The calculated containment pressure is reduced to and maintained at less than one-half of design pressure at 24 hours.~~ *

The containment pressure response for the peak pressure steam line break case is provided in Figure 6.2.1.1-1. The temperature response for this case is provided in Figure 6.2.1.1-2. Figures 6.2.1.1-3 and 6.2.1.1-4 provide the containment pressure and temperature response for the peak temperature steam line break case.

Reference 20, Section 4

The passive internal containment heat sink data used in the WGOTHIC analyses is presented in ~~Tables 6.2.1.1-4 through 6.2.1.1-7~~. The containment pressure and temperature responses to a double-ended cold leg guillotine are presented in Figures 6.2.1.1-5 and 6.2.1.1-6 for the 24 hour portion of the transient and Figures 6.2.1.1-7 and 6.2.1.1-8 for the 72 hour transient. The containment pressure and temperature response to a double-ended hot leg guillotine break are presented in Figures 6.2.1.1-9 and 6.2.1.1-10. Data for both metallic and concrete heat sinks are presented. The physical properties of the materials corresponding to the heat sink information is presented in Table 6.2.1.1-8. *

The instrumentation provided inside containment to monitor and record the containment pressure and temperature is found in Section 7.5.

6.2.1.1.4 External Pressure Analysis

Certain design basis events and credible inadvertent systems actuation have the potential to result in containment external pressure loads. Evaluations of these events show that a loss of all ac power sources during extreme cold ambient conditions has the potential for creating the worst-case external pressure load on the containment vessel. This event leads to a reduction in the internal containment heat loads from the reactor coolant system and other active components, thus resulting in a temperature reduction within the containment and an

for a given time period is determined from the difference between the energy required to raise the temperature of the incoming flow to saturation and the sum of the decay heat, core stored energy, reactor coolant system metal energy and SG mass and metal energy release rates. The energy release rate for the available break flow is determined from a comparison of the total energy available release rate and the energy release rate assuming that the break flow is 100-percent saturated steam. Saturated steam releases maximize the calculated containment pressurization.

6.2.1.3.2.4 Single Failure Analysis

The assumptions for the containment mass and energy release analysis are intended to maximize the calculated release. A single failure could reduce the flow rate of water to the RCS, but would not disable ~~any~~ the passive core cooling ~~system~~ function. For example, if one of the two parallel valves from the CMT were to fail to open, the injection flow rate would be reduced and, as a result, the break mass release rate would decrease. Therefore, to maximize the releases, the AP600 mass and energy release calculations do not assume a single failure. The effects of a single failure are taken into account in the containment analysis of subsection 6.2.1.1 conservatively.

6.2.1.3.2.5 Metal-Water Reaction

Consistent with 10 CFR 50, Appendix K criteria, the energy release associated with the zirconium-water exothermic reaction has been considered. The LOCA peak cladding temperature analysis, presented in Chapter 15, that demonstrates compliance with the Appendix K criteria demonstrates that no appreciable level of zirconium oxidation occurs. This level of reaction has been bounded in the containment mass and energy release analysis by incorporating the heat of reaction from 1 percent of the zirconium surrounding the fuel. This exceeds the level predicted by the LOCA analysis and results in additional conservatism in the mass and energy release calculations.

6.2.1.3.2.6 Energy Inventories

Inventories of the amount of mass and energy released to containment during a postulated LOCA are provided in summary Tables 6.2.1.3-2 through 6.2.1.3-7.

6.2.1.3.2.7 Additional Information Required for Confirmatory Analysis

System parameters and hydraulic characteristics needed to perform confirmatory analysis are provided in Table 6.2.1.3-8 and Figures 6.2.1.3-1 through 6.2.1.3-4.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary-System Pipe Rupture Inside Containment

Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a

layout downstream of the isolation valves determine the volume in the feedwater lines that cannot be isolated from the steam generators. As the steam generator pressure decreases, some of the fluid in this volume will flash into the steam generator, providing additional secondary fluid that may exit out the rupture.

The feedwater addition that occurs prior to closing of the feedwater line isolation valves influences the steam generator blowdown in several ways. First, the rapid addition increases the amount of entrained water in large-break cases by lowering the bulk quality of the steam generator inventory. Second, because the water entering the steam generator is subcooled, it lowers the steam pressure, thereby reducing the flow rate out the break. Finally, the increased flow rate causes an increase in the heat transfer rate from the primary-to-secondary system, resulting in greater energy being released out the break. Since these are competing effects on the total mass and energy release, no worst case feedwater transient can be defined for all plant conditions. In the results presented, the worst effects of each variable have been used. For example, moisture entrainment for each break is calculated assuming conservatively small feedwater additions so that the entrained water is minimized or zero. Determination of total steam generator inventory is based on conservatively large feedwater additions, as explained in subsection 6.2.1.4.3.2.

The unisolated feedwater line volumes between the steam generator and the isolation valves serve as a source for additional high-energy fluid to be discharged through the pipe break. This volume is accounted for in the mass and energy release data presented in subsection 6.2.1.4.3.2.

6.2.1.4.1.3 Startup Feedwater System Design

Within the first minute following a steam line break, the startup feedwater system may be initiated on any one of several protection system signals. The addition of startup feedwater to the steam generators increases the secondary mass available for release to the containment, as well as the heat transferred to the secondary fluid. The effects on the steam generator mass are maximized in the calculation described in subsection 6.2.1.4.3.2 by assuming full startup feedwater flow to the faulted steam generator starting at time zero from the safeguard system(s) signal or low steam generator level reactor trip and continuing until automatically terminated.

6.2.1.4.1.4 Postulated Break Type, Size and Location

Postulated Break Type

Two types of postulated pipe ruptures are considered in evaluating steam line breaks.

First is a split rupture in which a hole opens at some point on the side of the steam pipe ~~or steam header~~ but does not result in a complete severance of the pipe. A single, distinct break area is fed uniformly by both steam generators until steam line isolation occurs. The blowdown flow rates from the individual steam generators are interdependent, since fluid coupling exists between the steam lines. Because flow limiting orifices are provided in each

steam generator, the largest split rupture can have an effective area prior to isolation, that is no greater than the throat area of the flow restrictor times the number of steam generators. Following isolation, the effective break area for the steam generator with the broken line can be no greater than the flow restrictor throat area.

The second break type is the double-ended guillotine rupture in which the steam pipe is completely severed and the ends of the break displace from each other. Guillotine ruptures are characterized by two distinct break locations, each of equal area, but being fed by different steam generators. The largest guillotine rupture can have an effective area per steam generator no greater than the throat area of one steam line flow restrictor.

Postulated Break Size

Break area is also important when evaluating steam line breaks. It controls the rate of releases to the containment, and influences the steam pressure decay and the amount of entrained water in the blowdown flow. The data presented in this section include releases for four breaks at each of four initial power levels. Included are three double-ended ruptures and one split rupture, as follows:

- A full double-ended pipe rupture downstream of the steam line flow restrictor. For this case, the actual break area equals the cross-sectional area of the steam line, but the blowdown from the steam generator with the broken line is controlled by the flow restrictor throat area (1.388 square feet). The reverse flow from the intact steam generator is controlled by the smaller of the pipe cross section, the steam stop valve seat area, or the total flow restrictor throat area in the intact steam generator. The reverse flow has been conservatively assumed to be controlled by the flow restrictor in the intact loop steam generator.

- An intermediate size double-ended rupture having an area of 0.4 square feet.

- A small double-ended rupture having an area of 0.1 square feet.

- A split rupture representing the largest break which can ~~neither~~ generate a steam line isolation signal from the primary protection equipment. Steam and feedwater line isolation signals are generated by high containment pressure signals for this type of break.

Table 6.2.1.4-1 lists the spectrum of secondary system pipe ruptures analyzed.

Postulated Break Location

Break location affects steam line blowdown due to the pressure losses which occur in the length of piping between the steam generator and the break. The effect of the pressure loss is to reduce the effective break area seen by the steam generator. Although this reduces the rate of blowdown, it would not significantly change the total release of energy to the

6.2.1.4.1.7 Steam Generator Reverse Heat Transfer and Reactor Coolant System Metal Heat Capacity

Once steam line isolation is complete, the steam generator in the intact steam loop becomes a source of energy that can be transferred to the steam generator with the broken line. This energy transfer occurs through the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes drops below the temperature of the secondary fluid in the intact unit, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steam line.

Similarly, the heat stored in the metal of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps is transferred to the primary coolant as the plant cooldown progresses. This energy also is available to be transferred to the steam generator with the broken line.

The effects of both the reactor coolant system metal and the reverse steam generator heat transfer are included in the results presented.

6.2.1.4.2 Description of Blowdown Model

The steamline blowdown is calculated with the AP600 version of LOFTRAN (Ref. 31). This is a version of LOFTRAN (Ref. 6) which has been modified to include simulation of the AP600 passive residual heat removal heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. Documentation of the code changes for the passive models ~~has been submitted to the NRC (Ref. 31)~~ is provided in Reference 31. The methodology for the steamline break analysis is based on Ref. 5.

6.2.1.4.3 Containment Response Analysis

The WGOTHIC Computer Code (Reference 1) is used to determine the containment responses following the steam line break. The containment response analysis is described in subsection 6.2.1.1.

6.2.1.4.3.1 Initial Conditions

The initial containment conditions are discussed in subsection 6.2.1.1.3.

6.2.1.4.3.2 Mass and Energy Release Data

Using References 5 and 6 as a basis, mass and energy release data are developed to determine the containment pressure-temperature response for the spectrum of breaks analyzed. Tables 6.2.1.4-2 and 6.2.1.4-3 provide the mass and energy release data for the cases that produce the highest containment pressure and temperature in the containment response analysis. Table 6.2.1.4-4 provides plant data used for the cases used in the mass and energy releases determination.

30. Nuclear Energy Institute Report, NEI 94-01, "Industry Guidelines for Implementing Performance Based Option of 10 CFR 50, Appendix J," Revision 0.

31. Carlin, E. L. and U. Bachrach, "LOFTRAN and LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1 (Proprietary), June 1997.

32. Letter, B.A. McIntyre to T.R. Quay, DCP/NRC1247, dated February 20, 1998, "SSAR 6.2 Markup and Evaluation Model Changes" (Attachment 2) *

Table 6.2.1.1-1

SUMMARY OF CALCULATED PRESSURES AND TEMPERATURES

Break	Peak Pressure (psig)	Available[1] Margin (psi)	Peak Temperature (°F)
Double-ended hot leg guillotine	~~38.9~~ 38.7	~~6.1~~ 6.3	~~390.9~~ 389.5
Double-ended cold leg guillotine	~~44.0~~ 44.5	~~1.0~~ 0.5	~~280.3~~ 280.3
~~1.38~~ F ft², full DER, 102% power, MSIV failure	~~43.6~~ 44.0	~~1.4~~ 1.0	~~356.1~~ 363.9
~~1.388~~ F ft², full DER, 30% power, MSIV failure	~~44.8~~ 44.7	~~0.2~~ 0.3	~~353.6~~ 361.3

1. Design Pressure is 45 psig

DELETE Table 6.2.1.1-4 through 6.2.1.1-7

Table ~~6.2.1.1-4~~ (Sheet 1 of 6)

METAL HEAT SINKS

Metallic Description	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
Reactor Cavity				
Miscellaneous Equipment and Structural Steel	211.49	0.083	SS	None
	62.06	0.002	CS	CZ
	2119.7	0.00756	CS	CZ+E
	104.0	0.00870	CS	G
	275.5	0.012	CS	G
	986.7	0.1667	CS	CZ+E
Accumulator Cavity Southeast				
Miscellaneous Equipment and Structural Steel	767.7	0.166	CS	CZ
	2304.2	0.00756	CS	CZ+E
	104.54	0.0124	CS	G
Accumulator Cavity (Northeast)				
Miscellaneous Equipment and Structural Steel	767.7	0.166	CS	CZ
	3360.4	0.00756	CS	CZ+E
	104.54	0.01240	CS	G
Steam Generator Room East				
Miscellaneous Equipment and Structural Steel	2240.10	0.02430	CS	CZ+E
	3388.77	0.00756	CS	CZ+E
	104.54	0.01200	CS	G
	181.39	0.10400	CS	CZ+E
	281.03	0.40500	CS	CZ+E
Steam Generator Room West				
Miscellaneous Equipment and Structural Steel	2240.10	0.02430	CS	CZ+E
	3388.77	0.00756	CS	CZ+E
	104.54	0.01200	CS	G
	181.39	0.10400	CS	CZ+E
	281.03	0.40500	CS	CZ+E

Table 6.2.1.1-4 (Sheet 2 of 6)

METAL HEAT SINKS

Metallic Description	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
CVS & CMT Room				
Miscellaneous Equipment and Structural Steel	79.72	0.60000	SS	None
	480.86	0.73700	SS	None
	114.67	0.75000	SS	None
	1449.32	0.03620	CS	CZ+E
	1252.95	0.41700	CS	CZ+E
	1262.14	0.08330	CS	CZ+E
	520.82	0.25000	CS	CZ+E
	1646.66	0.00756	CS	CZ+E
	505.50	0.00870	CS	G
	1848.80	0.44500	CS	CZ+E
	11063.75	0.13542	CS	CZ+E
	3285.94	0.03250	CS	CZ+E
	7563.38	0.03860	CS	CZ+E
	436.92	0.01632	CS	CZ+E
	321.80	0.17500	CS	CZ+E
	24952.44	0.01100	CS	CZ+E
	320.15	0.01200	CS	G
	43.04	0.01386	CS	G
Refueling Room				
Miscellaneous Equipment and Structural Steel	933.8	0.04200	SS	None
	79.48	0.36660	SS	None
	2016.00	0.05830	SS	None
	279.83	0.00756	CS	CZ+E
IRWST Room				
Miscellaneous Equipment and Structural Steel	7050.96	0.05000	SS	None

Table 6.2.1.1-4 (Sheet 3 of 6)

METAL HEAT SINKS

Metallic Description	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
Upper East SG Compartment				
Miscellaneous Equipment and Structural Steel	77.39	0.37700	CS	CZ+E
	103.67	0.37500	CS	CZ+E
	5612.9	0.00756	CS	CZ+E
Upper West SG Compartment				
Miscellaneous Equipment and Structural Steel	103.67	0.37500	CS	CZ+E
	5612.9	0.00756	CS	CZ+E
	77.39	0.37700	CS	CZ+E
South Inner Half Annulus				
Miscellaneous Equipment and Structural Steel	491.60	0.02100	SS	None
	742.20	0.05520	CS	CZ+E
	457.70	0.00756	CS	CZ+E
South Inner Quarter Annulus				
Miscellaneous Equipment and Structural Steel	819.80	0.05520	CS	CZ+E
North Inner Half Annulus				
Miscellaneous Equipment and Structural Steel	607.26	0.01200	CS	G
	147.60	0.00756	CS	CZ+E
North Inner Half Annulus				
Miscellaneous Equipment and Structural Steel	3987.82	0.00756	CS	CZ+E
	399.95	0.01200	CS	G
North Mid Quarter Annulus				
Miscellaneous Equipment and Structural Steel	540.71	0.01632	CS	CZ+E
	356.68	0.06700	CS	CZ+E
	888.90	0.00756	CS	CZ+E
	241.18	0.01947	CS	CZ+E
	788.80	0.00870	CS	G
	494.90	0.13000	CS	CZ+E

Table 6.2.1.1-4 (Sheet 4 of 6)

METAL HEAT SINKS

Metallic Description	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
West Mid Quarter Annulus				
Miscellaneous Equipment and Structural Steel	491.60	0.02100	CS	None
	3442.0	0.00756	CS	CZ+E
	136.80	0.01200	CS	G
	204.75	0.04180	CS	CZ+E
	700.28	0.00400	CS	CZ+E
	356.68	0.06700	CS	CZ
	265.90	0.00400	CS	CZ+E
	788.80	0.00870	CS	G
	494.90	0.13000	CS	CZ+E
South Mid Quarter Annulus				
Miscellaneous Equipment and Structural Steel	356.68	0.06700	CS	CZ+E
	3783.0	0.00756	CS	CZ+E
	788.80	0.00870	CS	G
	494.90	0.13000	CS	CZ+E
	204.75	0.04180	CS	CZ+E
	700.28	0.00400	CS	CZ+E
	356.68	0.06700	CS	CZ+E
East Mid Quarter Annulus				
Miscellaneous Equipment and Structural Steel	265.90	0.00400	CS	CZ+E
	788.80	0.00870	CS	G
	494.90	0.13000	CS	CZ+E
North Outer Quarter Annulus				
Miscellaneous Equipment and Structural Steel	1032.4	0.00756	CS	CZ+E
	402.10	0.07000	CS	CZ+E
	788.80	0.00870	CS	CZ+E
	1101.60	0.13000	CS	CZ+E

Table 6.2.1.1-4 (Sheet 5 of 6)

METAL HEAT SINKS

Metallic Description	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
West Outer Quarter Annulus				
Miscellaneous Equipment and Structural Steel	1648.85	0.00756	CS	CZ+E
	402.10	0.07000	CS	CZ+E
	788.80	0.00870	CS	G
	1101.60	0.13000	CS	CZ+E
South Outer Quarter Annulus				
Miscellaneous Equipment and Structural Steel	1099.2	0.00756	CS	CZ+E
	402.10	0.07000	CS	CZ+E
	788.80	0.00870	CS	G
	1101.60	0.13000	CS	CZ+E
East Outer Quarter Annulus				
Miscellaneous Equipment and Structural Steel	615.2	0.22000	CS	CZ+E
	402.10	0.07000	CS	CZ+E
	824.50	0.00756	CS	CZ+E
	788.80	0.00870	CS	G
	1101.60	0.13000	CS	CZ+E
East Quarter Inner Dome				
Miscellaneous Equipment and Structural Steel	1996.00	0.14500	CS	CZ+E
North Quarter Inner Dome				
Miscellaneous Equipment and Structural Steel	1996.00	0.14500	CS	CZ+E

Table 6.2.1.1-4 (Sheet 6 of 6)

METAL HEAT SINKS

Metallic Description	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
West Quarter Inner Dome				
Miscellaneous Equipment and Structural Steel	1996.00	0.14500	CS	CZ+E
East Quarter Inner Dome				
Miscellaneous Equipment and Structural Steel	1996.00	0.14500	CS	CZ+E
Lower Chimney				
Miscellaneous Equipment and Structural Steel	1454.14	0.01990	CS	CZ+E
	277.51	0.02250	CS	CZ+E
	110.24	0.01200	CS	G
	142.11	0.00756	CS	CZ+E
	78.93	0.00756	CS	CZ+E

Notes:

1. Two types of materials will be used for metallic structures:
   A. Carbon Steel (CS)
   B. Stainless Steel (SS)
2. Three paint coatings are used on metallic structures:
   A. Epoxy (E)
   B. Carbo Zinc (CZ)
   D. Hot dipped galvanizing (G)

Table 6.2.1.1-5 (Sheet 1 of 3)

CONCRETE HEAT SINKS

Surface	Exposed Area (ft^2)	Thickness (ft)	Liner Plate		Paint	
			Interior	Exterior	Interior	Exterior
Reactor Cavity						
Structural walls, floors and ceilings	219.50	3.260	CS	CS	CZ+E	CZ+E
	693.00	8.890	CS	CS	CZ+E	CZ+E
	1044.28	4.000	CS	CS	CZ+E	CZ+E
	976.7	6.460	CS	CS	CZ+E	CZ+E
	214.50	2.500	CS	CS	CZ+E	CZ+E
	2429.00	12.520	CS	---	CZ+E	---
Accumulator Cavity South-East Room						
Structural walls, floors and ceilings	245.46	4.000	CS	SS	CZ+E	None
	718.50	2.000	CS	None	CZ	E
	2118.55	12.520	CS	---	CZ+E	---
Accumulator Cavity North-East Room						
Structural walls, floors and ceilings	375.10	4.500	CS	None	CZ+E	E
	993.00	2.000	CS	None	CZ+E	E
	2816.80	12.520	CS	---	CZ+E	---
Steam Generator Room East						
Structural walls, floors and ceilings	2970.37	2.500	CS	CS	CZ+E	CZ+E
	758.16	4.000	CS	SS	CZ+E	None
	941.44	12.520	CS	---	E	---
Steam Generator Room West						
Structural walls, floors and ceilings	2333.7	2.500	CS	CS	CZ+E	CZ+E
	758.16	4.000	CS	SS	CZ+E	None
	1986.6	12.520	CS	---	CZ+E	---

Table 6.2.1.1-5 (Sheet 2 of 3)

CONCRETE HEAT SINKS

Surface	Exposed Area (ft^2)	Thickness (ft)	Liner Plate		Paint	
			Interior	Exterior	Interior	Exterior
CMT and CVS Room						
Structural walls, floors and ceilings	2550.78	2.500	CS	None	CZ	E
	1630.00	4.000	CS	SS	CZ+E	None
	5109.33	1.125	CS	None	CZ	E
	346.46	2.000	CS	CS	CZ	CZ
	5161.52	12.520	None	---	E	---
Refueling Room						
Structural walls, floors and ceilings	776.37	1.420	SS	---	None	---
	1705.99	4.000	SS	SS	None	None
	1539.00	12.520	SS	---	None	---
IRWST Room						
Structural walls, floors and ceilings	3146.36	2.000	SS	None	None	E
	3585.20	12.520	SS	---	None	---
Upper East SG Compartment						
Structural walls, floors and ceilings	1241.22	2.500	CS	CS	CZ+E	CZ+E
Upper West SG Compartment						
Structural walls, floors and ceilings	1241.22	2.500	CS	CS	CZ+E	CZ+E
South Inner Half Annulus						
Structural walls, floors and ceilings	491.66	2.000	SS	SS	None	None
Inner Half Annulus						
Structural walls, floors and ceilings	1281.38	2.500	CS	CS	CZ+E	CZ+E

Table 6.2.1.1-5 (Sheet 3 of 3)

CONCRETE HEAT SINKS

Surface	Exposed Area (ft^2)	Thickness (ft)	Liner Plate		Paint	
			Interior	Exterior	Interior	Exterior
North Inner Half Annulus						
Structural walls, ceilings, and floors	1005.00	2.500	CS	CS	CZ+E	CZ+E
West Mid Quarter Annulus						
Structural walls, ceilings, and floors	491.66	2.000	SS	SS	None	None
Air Baffle & Shield Building Lower Gap						
Structural walls, ceilings, and floors	53065.02	3.000	None	None	None	None
Lower Chimney						
Structural walls, ceilings, and floors	13357.18	0.500	None	None	None	None
	13975.6	2.000	None	None	None	None
	1272.00	1.000	None	None	None	None
Top Chimney						
Structural walls, ceilings, and floors	2161.40	2.000	None	None	None	None

Notes:

1. Steel liner plates are either:
   A. Carbon Steel (CS)
   B. Stainless Steel (SS)
2. Paint on liner plates and concrete surfaces is either:
   A. Epoxy (E)
   B. Carbo Zinc (CZ)

   When Carbo Zinc and Epoxy are used together, the Carbo Zinc is applied as a primer, then the Epoxy is applied as a topcoat.

Table 6.2.1.1-6

CONTAINMENT SHELL AND BAFFLE METAL HEAT SINKS

	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
Stack #1				
Shell	11266.6	0.13641	CS	CZ
Baffle	10653.7	0.00839	CS	CZ
Stack #2				
Shell	11266.6	0.13641	CS	CZ
Baffle	10653.7	0.00839	CS	CZ
Stack #3				
Shell	11266.6	0.13641	CS	CZ
Baffle	10653.7	0.00839	CS	CZ
Stack #4				
Shell	11266.6	0.13641	CS	CZ
Baffle	10653.7	0.00839	CS	CZ
Stack #5				
Shell	2128.9	0.13641	CS	CZ
Baffle	1183.9	0.00839	CS	CZ
Stack #6				
Shell	2128.9	0.13641	CS	CZ
Baffle	1183.9	0.00839	CS	CZ
Stack #7				
Shell	2128.9	0.13641	CS	CZ
Baffle	1183.9	0.00839	CS	CZ
Stack #8				
Shell	2128.9	0.13641	CS	CZ
Baffle	1183.9	0.00839	CS	CZ

Table 6.2.1.1-7

SHIELD BUILDING CONCRETE HEAT SINKS

	Exposed Area (ft^2)	Thickness (ft)	Material	Paint
Stack #1				
Shield Bldg.	12057.5	3.0	Concrete	None
Stack #2				
Shield Bldg.	12057.5	3.0	Concrete	None
Stack #3				
Shield Bldg.	12057.5	3.0	Concrete	None
Stack #4				
Shield Bldg.	12057.5	3.0	Concrete	None
Stack #5				
Shield Bldg.	2497.5	3.0	Concrete	None
Stack #6				
Shield Bldg.	2497.5	3.0	Concrete	None
Stack #7				
Shield Bldg.	2497.5	3.0	Concrete	None
Stack #8				
Shield Bldg.	2497.5	3.0	Concrete	None

Table 6.2.1.1-8

PHYSICAL PROPERTIES OF PASSIVE HEAT SINKS

Material	Density (lbm/ft^3)	Thermal Conductivity (Btu/hr-ft-°F)	Specific Heat (Btu/lbm-°F)	Dry Emis.	Wet Emis.
Epoxy	105	0.1875	0.35	0.81	0.95
Carbon Steel	490.7	23.6	0.107	0.81	0.95
Concrete	140.	0.83	0.19	0.81	0.95
Stainless Steel	501.	9.4	0.12	0.81	0.95
Carbo Zinc	207.5	1.21	0.15	0.81	0.95
Oxidized Carbo Zinc	207.5	0.302	0.15	0.81	0.95
Carbo Zinc-PCS	207.5	0.302	0.15	1e-10	1e-10
Inside Surface					
Air @ 0°F	0.0864	0.0131	0.240	1e-10	1e-10
Air @ 250°F	0.056	0.0192	0.242	1e-10	1e-10
Air @ 500°F	0.0414	0.0246	0.248	1e-10	1e-10

Table 6.2.1.3-9 (Sheet 4 of 5)

LONG-TERM DECLG BREAK
POST-BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
524.64	58.96	380.28	120.27	1167.40
544.64	60.92	374.34	117.19	1167.40
564.64	62.77	368.65	114.23	1167.40
584.64	64.52	363.17	111.37	1167.40
604.64	66.17	357.91	108.61	1167.40
654.64	69.55	345.88	102.43	1167.40
704.64	72.42	334.73	96.78	1167.40
754.64	74.81	324.37	91.60	1167.40
804.64	76.79	314.65	86.84	1167.40
854.64	78.24	305.69	82.60	1167.40
904.64	79.36	297.25	78.70	1167.40
954.64	80.20	289.27	75.09	1167.40
1004.64	80.74	281.76	71.76	1167.40
1504.75	223.22	149.04	47.12	1167.40
2004.75	211.76	127.84	37.58	1167.40
3504.75	156.70	104.74	27.09	1167.40
4004.75	157.03	100.94	25.08	1167.40
6004.84	153.75	96.34	21.54	1167.40
7504.95	134.76	96.64	20.33	1167.40
8004.95	132.14	96.53	19.75	1167.40
10005.00	128.76	96.18	18.53	1167.40
15005.00	130.22	95.25	16.61	1167.40
20005.80	114.62	95.62	15.38	1167.40
26007.30	~~94.79~~ 35.36	~~96.77~~ 111.52	14.64	1167.40
30007.90	~~0.85~~ 0.0	~~1028.58~~ 1168.30	~~14.15~~ 16.32	~~1167.40~~ 1168.30
36008.10	0.00	1168.30	15.38	1168.30
40000.00	0.00	1168.30	14.76	1168.30
60000.00	0.00	1168.30	13.20	1168.30

Table 6.2.1.3-9 (Sheet 5 of 5)

LONG-TERM DECLG BREAK
POST-BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
80000.00	0.00	1168.30	12.19	1168.30
100000.00	0.00	1168.30	11.42	1168.30
150000.00	0.00	1168.30	10.10	1168.30
200000.00	0.00	1168.30	9.20	1168.30
400000.00	0.00	1168.30	7.15	1168.30
600000.00	0.00	1168.30	6.07	1168.30
800000.00	0.00	1168.30	5.38	1168.30
1000000.00	0.00	1168.30	4.89	1168.30
1500000.00	0.00	1168.30	4.11	1168.30
2000000.00	0.00	1168.30	3.62	1168.30
4000000.00	0.00	1168.30	2.57	1168.30

Table 6.2.1.4-2 (Sheet 1 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Use initial steam mass with blowdown

Initial steam generator mass (lbm)	~~167750~~ 164530
Mass added by feedwater flashing (lbm)	~~9493~~ 10390
Mass added ~~by unisolatable steam~~ (lbm)	~~9637~~ 9970
Initial steam pressure (psia)	~~975.6~~ 976.5
Feedwater line isolation at (sec)	~~7.99~~ 7.92
Steam line isolation at (sec)	~~7.99~~ 7.92

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	0.00
0.1	12438.1	1193.58
0.2	12412.4	1193.65
0.3	12392.9	1193.71
0.4	12373.5	1193.76
0.5	12354.8	1193.81
0.6	12336.5	1193.86
0.7	12318.5	1193.91
0.8	12300.9	1193.95
0.9	12283.7	1194.00
1.0	12266.8	1194.04
1.1	12250.1	1194.08
1.2	5397.5	1193.94
1.3	5353.1	1194.22
1.7	5213.8	1195.06
2.1	5084.4	1195.81
2.5	4964.0	1196.49
2.9	4850.3	1197.11
3.3	4743.1	1197.67
3.7	4642.6	1198.19
4.1	4546.9	1198.67

Replace all the data in the table with Insert 30

TIME (sec)	MASS (lbm/sec)	ENTHALPY (BTU/lbm)
0.0	12575	1193
0.1	12575	1193
0.2	12547	1193
0.3	12525	1193
0.4	12503	1193
0.5	12483	1193
0.6	12462	1193
0.7	12442	1193
0.8	12423	1193
0.9	12404	1193
1.0	12385	1193
1.1	5667	1194
1.2	5621	1194
1.3	5584	1194
1.4	5547	1195
1.5	5511	1195
1.6	5475	1195
1.7	5440	1195
1.8	5405	1195
1.9	5371	1196
2.0	5337	1196
2.1	5305	1196
2.2	5272	1196
2.3	5241	1196
2.4	5210	1196
2.5	5179	1197
2.6	5149	1197
2.7	5119	1197
2.8	5089	1197
2.9	5060	1197
3.0	5032	1197
3.1	5004	1198
3.2	4975	1198
3.3	4948	1198
3.4	4921	1198
3.5	4894	1198
3.6	4868	1198
3.7	4843	1198
3.8	4818	1198
3.9	4792	1198
4.0	4767	1199
4.1	4743	1199
4.2	4719	1199
4.3	4696	1199
4.4	4673	1199
4.5	4650	1199
4.6	4628	1199
4.7	4606	1199
4.8	4584	1199
4.9	4562	1199

5.0	4542	1200
5.1	4520	1200
5.2	4500	1200
5.3	4480	1200
5.4	4459	1200
5.5	4439	1200
5.6	4421	1200
5.7	4401	1200
5.8	4382	1200
5.9	4363	1200
6.0	4344	1200
6.1	4326	1200
6.2	4308	1200
6.3	4291	1201
6.4	4273	1201
6.5	4256	1201
6.6	4238	1201
6.7	4222	1201
6.8	4205	1201
6.9	4189	1201
7.0	4172	1201
7.1	4156	1201
7.2	4140	1201
7.3	4125	1201
7.4	4109	1201
7.5	4094	1201
7.6	4079	1201
7.7	4065	1201
7.8	4049	1201
7.9	4036	1201
8.0	4021	1201
8.1	4007	1202
8.2	3997	1202
8.3	3984	1202
8.4	3973	1202
8.5	3962	1202
8.6	3950	1202
8.7	3940	1202
8.8	3929	1202
8.9	3918	1202
9.0	3907	1202
9.1	1872	1203
9.2	1868	1203
9.3	1864	1203
9.4	1860	1203
9.5	1856	1203
9.6	1852	1203
9.7	1848	1203
9.8	1844	1203
9.9	1840	1203
10	1836	1203
11	1796	1203
12	1757	1203
13	1718	1204

14	1679	1204
15	1639	1204
16	1601	1204
17	1561	1204
18	1517	1204
19	1470	1204
20	1422	1204
21	1385	1204
22	1340	1204
23	1295	1204
24	1252	1204
25	1211	1204
26	1171	1204
27	1133	1204
28	1097	1204
29	1062	1204
30	1029	1204
31	998	1204
32	968	1203
33	939	1203
34	912	1203
35	888	1203
36	866	1203
37	845	1202
38	824	1202
39	805	1202
40	786	1202
41	768	1201
42	751	1201
43	735	1201
44	719	1201
45	704	1201
46	689	1200
47	676	1200
48	662	1200
49	650	1200
50	637	1199
55	584	1198
60	541	1197
65	508	1196
70	481	1195
75	459	1194
80	442	1194
85	428	1193
90	417	1193
95	408	1192
100	399	1192
105	392	1192
110	385	1191
115	378	1191
120	372	1191
125	365	1190
130	358	1190
135	351	1190

Insert 3D (3.4.5)

140	344	1189
145	337	1189
150	330	1189
155	324	1188
160	318	1188
165	312	1188
170	306	1187
175	301	1187
180	297	1187
185	292	1186
190	288	1186
195	285	1186
200	281	1186
205	278	1185
210	275	1185
215	272	1185
220	270	1185
225	267	1185
230	265	1185
235	263	1184
240	261	1184
245	259	1184
250	256	1184
255	254	1184
260	252	1184
265	250	1183
270	248	1183
275	246	1183
280	244	1183
285	242	1183
290	240	1183
295	237	1182
300	235	1182
306	233	1182
312	230	1182
318	227	1181
324	225	1181
330	222	1181
336	220	1181
342	218	1181
348	215	1180
354	213	1180
360	211	1180
366	208	1180
372	206	1180
378	204	1179
384	202	1179
390	200	1179
396	198	1179
402	196	1179
408	194	1178
414	192	1178
420	190	1178
426	188	1178

432	186	1178
438	184	1177
444	183	1177
450	181	1177
456	179	1177
462	177	1177
468	175	1176
474	173	1176
480	168	1176
486	163	1175
492	157	1174
498	152	1174
504	145	1173
510	138	1172
516	130	1170
522	118	1168
528	102	1166
534	84	1162
540	63	1156
546	9	1151
552	0	1151

Table 6.2.1.4-2 (Sheet 2 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
4.5	4458.9	1199.10
4.9	4375.6	1199.49
5.3	4296.8	1199.86
5.7	4222.1	1200.20
6.1	4151.3	1200.52
6.5	4084.3	1200.81
6.9	4020.7	1201.08
7.3	3960.5	1201.32
7.7	3903.3	1201.55
8.1	3849.1	1201.76
8.5	3806.8	1201.93
8.9	3765.0	1202.09
9.3	1815.1	1202.58
9.7	1799.8	1202.68
10.1	1784.7	1702.78
10.5	1769.7	1202.87
10.9	1754.8	1202.96
11.3	1739.9	1203.05
11.7	1725.1	1203.14
12.1	1710.2	1203.22
12.5	1695.4	1203.31
12.9	1680.5	1203.39
13.3	1665.5	1203.46
13.7	1650.6	1203.54
14.1	1635.7	1203.61
14.5	1620.7	1203.69

Table 6.2.1.4-2 (Sheet 3 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
14.9	1605.8	1203.75
15.3	1590.9	1203.82
15.7	1576.1	1203.88
16.1	1561.3	1203.94
16.5	1546.5	1204.00
16.9	1531.1	1204.05
17.3	1515.1	1204.11
17.7	1498.4	1204.16
18.1	1481.3	1204.21
18.5	1463.8	1204.26
18.9	1446.0	1204.29
19.3	1428.0	1204.33
19.7	1409.9	1204.36
20.5	1382.9	1204.40
22.5	1295.8	1204.47
24.5	1213.5	1204.45
26.5	1137.6	1204.34
28.5	1067.9	1204.16
30.5	1004.2	1203.87
32.5	946.2	1203.57
34.5	893.5	1203.20
36.5	845.6	1202.79
38.5	802.6	1202.35
40.5	763.6	1201.90
42.5	728.0	1201.42
44.5	695.5	1200.92

Table 6.2.1.4-2 (Sheet 4 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
46.5	665.7	1200.41
48.5	642.8	1199.99
50.5	620.9	1199.55
52.5	600.6	1199.12
54.5	581.8	1198.69
56.5	564.6	1198.28
58.5	548.7	1197.88
60.5	534.0	1197.49
62.5	520.6	1197.12
64.5	508.3	1196.77
66.5	497.1	1196.44
68.5	486.9	1196.13
70.5	477.4	1195.83
72.5	468.7	1195.54
74.5	460.8	1195.27
76.5	453.5	1195.01
78.5	446.8	1194.77
80.5	440.7	1194.54
82.5	435.1	1194.32
84.5	429.8	1194.11
86.5	425.0	1193.92
88.5	420.5	1193.73
90.5	416.3	1193.56
92.5	412.4	1193.39
94.5	408.7	1193.23
96.5	405.2	1193.08

Table 6.2.1.4-2 (Sheet 5 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
98.5	401.9	1192.94
101.0	398.3	1192.77
105.0	392.3	1192.50
109.0	386.5	1192.23
113.0	380.9	1191.98
117.0	375.4	1191.71
121.0	369.9	1191.45
125.0	364.3	1191.18
129.0	358.8	1190.90
133.0	353.2	1190.62
137.0	347.6	1190.34
141.0	342.2	1190.05
145.0	336.8	1189.76
149.0	331.4	1189.47
153.0	326.2	1189.19
157.0	321.2	1188.91
161.0	316.3	1188.63
165.0	311.6	1188.36
~~169.0~~	307.1	1188.10
173.0	302.8	1187.85
177.0	298.7	1187.61
181.0	294.9	1187.38
185.0	291.2	1187.16
189.0	288.0	1186.96
193.0	284.9	1186.77
197.0	282.1	1186.58

Table 6.2.1.4-2 (Sheet 6 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
201.0	279.4	1186.41
205.0	276.9	1186.24
209.0	274.5	1186.08
213.0	272.3	1185.93
217.0	270.1	1185.78
221.0	268.1	1185.64
225.0	266.1	1185.50
229.0	264.2	1185.36
233.0	262.4	1185.23
237.0	260.6	1185.10
241.0	258.8	1184.97
245.0	257.1	1184.84
249.0	255.4	1184.71
253.0	253.7	1184.59
257.0	252.0	1184.45
261.0	250.3	1184.32
265.0	248.6	1184.19
269.0	246.9	1184.06
273.0	245.2	1183.92
277.0	243.4	1183.78
281.0	241.7	1183.64
285.0	239.9	1183.50
289.0	238.2	1183.35
293.0	236.4	1183.21
297.0	234.6	1183.06
301.5	232.8	1182.90

Table 6.2.1.4-2 (Sheet 7 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
307.5	230.2	1182.68
313.5	227.5	1182.45
319.5	224.9	1182.22
325.5	222.4	1181.99
331.5	219.9	1181.77
337.5	217.4	1181.54
343.5	215.0	1181.32
349.5	212.6	1181.10
355.5	210.3	1180.88
361.5	208.0	1180.67
367.5	205.8	1180.46
373.5	203.6	1180.24
379.5	201.5	1180.04
385.5	199.4	1179.83
391.5	197.3	1179.62
397.5	195.4	1179.43
403.5	193.5	1179.23
409.5	191.6	1179.04
415.5	189.7	1178.85
421.5	187.8	1178.65
427.5	185.9	1178.46
433.5	184.0	1178.26
439.5	182.2	1178.06
445.5	180.3	1177.85
451.5	178.4	1177.65
457.5	176.5	1177.44

Table 6.2.1.4-2 (Sheet 8 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
463.5	174.6	1177.23
469.5	172.6	1177.02
475.5	170.9	1176.83
481.5	169.0	1176.62
487.5	166.9	1176.39
493.5	165.6	1176.23
499.5	164.8	1176.14
505.5	163.3	1175.98
511.5	161.1	1175.72
517.5	158.6	1175.41
523.5	154.7	1174.95
529.5	149.5	1174.28
535.5	144.1	1173.58
541.5	138.6	1172.82
547.5	133.0	1172.01
553.5	128.4	1171.14
559.5	121.6	1169.92
565.5	113.7	1168.52
571.5	103.7	1166.73
577.5	91.6	1164.41
583.5	77.5	1161.22
589.5	60.5	1156.41
595.5	35.2	1151.42
601.5	0.0	1150.91
607.5	0.0	1150.91
613.5	0.0	1150.91

Table 6.2.1.4-3 (Sheet 1 of 6)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

From initial Steamline break blowdown

Initial steam generator mass (lbm)	:	~~133650~~ 137052
Mass added ~~by unisolatable steam~~ (lbm)	:	~~8357~~ 7810
Mass added by feedwater flashing (lbm)	:	~~8650~~ 9504
Initial steam pressure (psia)	:	~~827.2~~ 843.2
Feedwater line isolation at (sec)	:	~~8.52~~ 8.56
Steam line isolation at (sec)	:	~~8.52~~ 8.56

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	0.00
0.1	10767.6	1198.33
0.2	10754.4	1198.36
0.3	10745.0	1198.38
0.4	10735.7	1198.40
0.5	10726.6	1198.43
0.6	10717.7	1198.45
0.7	10709.0	1198.47
0.8	10700.5	1198.49
0.9	10692.1	1198.50
1.0	10683.9	1198.52
1.1	10675.8	1198.54
1.2	4721.4	1197.81
1.3	4698.2	1197.93
~~1.7~~	4627.7	1198.29
2.1	4561.2	1198.62
2.5	4498.1	1198.94
2.9	4439.3	1199.22
3.3	4384.3	1199.48
3.7	4332.8	1199.72
4.1	4284.4	1199.95
4.5	4238.9	1200.16

Replace all the data in the table with Insert 1.2

TIME (sec)	MASS (lbm/sec)	ENTHALPY (BTU/lbm)
0.0	10886	1197
0.1	10886	1197
0.2	10871	1197
0.3	10860	1197
0.4	10849	1197
0.5	10839	1197
0.6	10829	1197
0.7	10819	1197
0.8	10809	1197
0.9	10800	1197
1.0	4962	1197
1.1	4938	1197
1.2	4919	1198
1.3	4900	1198
1.4	4882	1198
1.5	4863	1198
1.6	4846	1198
1.7	4828	1198
1.8	4811	1198
1.9	4794	1198
2.0	4777	1198
2.1	4760	1198
2.2	4743	1198
2.3	4727	1199
2.4	4711	1199
2.5	4695	1199
2.6	4680	1199
2.7	4664	1199
2.8	4649	1199
2.9	4634	1199
3.0	4619	1199
3.1	4605	1199
3.2	4591	1199
3.3	4577	1199
3.4	4563	1199
3.5	4550	1199
3.6	4536	1199
3.7	4522	1199
3.8	4510	1200
3.9	4497	1200
4.0	4485	1200
4.1	4472	1200
4.2	4459	1200
4.3	4448	1200
4.4	4436	1200
4.5	4424	1200
4.6	4413	1200
4.7	4403	1200
4.8	4392	1200
4.9	4382	1200

5.0	4372	1200
5.1	4361	1200
5.2	4351	1200
5.3	4341	1200
5.4	4332	1200
5.5	4322	1200
5.6	4312	1200
5.7	4303	1200
5.8	4294	1200
5.9	4285	1201
6.0	4275	1201
6.1	4267	1201
6.2	4258	1201
6.3	4249	1201
6.4	4241	1201
6.5	4233	1201
6.6	4224	1201
6.7	4216	1201
6.8	4208	1201
6.9	4200	1201
7.0	4192	1201
7.1	4184	1201
7.2	4177	1201
7.3	4170	1201
7.4	4162	1201
7.5	4154	1201
7.6	4147	1201
7.7	4140	1201
7.8	4133	1201
7.9	4126	1201
8.0	4119	1201
8.1	4112	1201
8.2	4105	1201
8.3	4099	1201
8.4	4092	1201
8.5	4085	1201
8.6	4078	1201
8.7	4072	1201
8.8	4067	1201
8.9	4061	1201
9.0	4055	1201
9.1	4050	1201
9.2	4044	1201
9.3	4038	1202
9.4	4033	1202
9.5	4026	1202
9.6	1975	1202
9.7	1972	1202
9.8	1970	1202
9.9	1967	1202
10	1965	1202
11	1936	1202
12	1901	1203
13	1861	1203

14	1818	1203
15	1772	1203
16	1726	1204
17	1680	1204
18	1632	1204
19	1579	1204
20	1525	1204
21	1482	1204
22	1429	1204
23	1377	1204
24	1327	1204
25	1279	1204
26	1234	1204
27	1190	1204
28	1149	1204
29	1111	1204
30	1074	1204
31	1040	1204
32	1009	1204
33	979	1203
34	950	1203
35	924	1203
36	899	1203
37	875	1203
38	852	1202
39	833	1202
40	815	1202
41	797	1202
42	781	1202
43	764	1201
44	749	1201
45	734	1201
46	720	1201
47	707	1201
48	694	1200
49	682	1200
50	670	1200
55	621	1199
60	584	1198
65	555	1197
70	534	1197
75	517	1196
80	504	1196
85	493	1196
90	483	1195
95	473	1195
100	464	1195
105	455	1194
110	445	1194
115	435	1193
120	425	1193
125	416	1193
130	406	1192
135	397	1192

140	388	1191
145	380	1191
150	372	1191
155	365	1190
160	358	1190
165	352	1190
170	346	1189
175	341	1189
180	336	1189
185	331	1189
190	327	1188
195	323	1188
200	320	1188
205	316	1188
210	313	1188
215	310	1187
220	307	1187
225	304	1187
230	301	1187
235	299	1187
240	296	1187
245	293	1186
250	291	1186
255	288	1186
260	286	1186
265	283	1186
270	281	1186
275	278	1185
280	276	1185
285	273	1185
290	271	1185
295	268	1185
300	266	1185
306	256	1184
312	244	1183
318	230	1182
324	214	1180
330	194	1178
336	168	1176
342	135	1171
348	95	1164
354	3	1157
360	0	1158

Table 6.2.1.4-3 (Sheet 2 of 6)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
4.9	4197.8	1200.34
5.3	4160.5	1200.51
5.7	4125.2	1200.66
6.1	4091.8	1200.80
6.5	4060.3	1200.94
6.9	4030.5	1201.06
7.3	4002.3	1201.18
7.7	3975.6	1201.29
8.1	3950.2	1201.39
8.5	3925.7	1201.49
8.9	3903.9	1201.58
9.3	3883.2	1201.66
9.7	3861.9	1201.74
10.1	1901.4	1201.97
10.5	1890.9	1202.05
10.9	1879.5	1202.13
11.3	1867.2	1202.22
11.7	1854.0	1202.31
12.1	1839.9	1202.41
12.5	1825.0	1202.51
12.9	1809.3	1202.62
13.3	1793.0	1202.72
13.7	1776.2	1202.83
[illegible]	1758.9	1202.94
14.5	1741.4	1203.04
14.9	1723.6	1203

Table 6.2.1.4-3 (Sheet 3 of 6)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
15.3	1705.7	1203.25
15.7	1687.8	1203.35
16.1	1670.0	1203.44
16.5	1652.3	1203.53
16.9	1634.8	1203.62
17.3	1616.8	1203.70
17.7	1598.0	1203.79
18.1	1578.6	1203.87
18.5	1558.7	1203.95
18.9	1538.3	1204.03
19.3	1517.6	1204.10
19.7	1496.7	1204.17
20.5	1465.3	1204.25
22.5	1363.6	1204.42
24.5	1266.8	1204.48
26.5	1179.2	1204.41
28.5	1100.0	1204.25
30.5	1029.8	1204.01
32.5	968.1	1203.70
34.5	913.4	1203.35
36.5	864.5	1202.96
38.5	820.7	1202.54
40.5	781.3	1202.11
42.5	745.3	1201.66
44.5	712.9	1201.19
46.5	683.6	1200.72

Table 6.2.1.4-3 (Sheet 4 of 6)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
48.5	659.1	1200.30
50.5	639.7	1199.93
52.5	621.7	1199.57
54.5	605.3	1199.22
56.5	590.6	1198.89
58.5	577.3	1198.59
60.5	565.4	1198.30
62.5	554.8	1198.03
64.5	545.2	1197.79
66.5	536.7	1197.56
68.5	529.1	1197.36
70.5	522.2	1197.17
72.5	516.0	1196.99
74.5	510.4	1196.83
76.5	505.3	1196.69
78.5	500.6	1196.55
80.5	496.3	1196.42
82.5	492.2	1196.29
84.5	488.3	1196.17
86.5	484.5	1196.06
88.5	480.8	1195.94
90.5	477.2	1195.82
92.5	473.6	1195.71
94.5	470.1	1195.59
96.5	466.5	1195.47
98.5	462.9	1195.34

Table 6.2.1.4-3 (Sheet 5 of 6)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
101.0	458.8	1195.20
105.0	451.6	1194.94
109.0	444.0	1194.66
113.0	436.3	1194.36
117.0	428.4	1194.06
121.0	420.6	1193.74
125.0	412.9	1193.42
129.0	405.3	1193.09
133.0	397.9	1192.76
137.0	390.9	1192.43
141.0	384.1	1192.12
145.0	377.6	1191.82
149.0	371.5	1191.53
153.0	365.6	1191.24
157.0	360.2	1190.97
161.0	355.0	1190.72
165.0	350.2	1190.47
169.0	345.6	1190.23
173.0	341.5	1190.01
177.0	337.5	1189.80
181.0	333.7	1189.60
185.0	330.2	1189.41
189.0	326.9	1189.23
193.0	323.8	1189.05
197.0	320.8	1188.89
202.0	317.7	1188.71

Table 6.2.1.4-3 (Sheet 6 of 6)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
210.0	312.6	1188.42
218.0	307.8	1188.15
226.0	303.3	1187.88
234.0	299.0	1187.62
242.0	294.7	1187.37
250.0	290.5	1187.12
258.0	286.3	1186.86
266.0	282.2	1186.59
274.0	278.0	1186.32
282.0	273.8	1186.03
290.0	269.6	1185.74
298.0	265.4	1185.45
306.0	255.6	1184.73
314.0	238.8	1183.41
322.0	218.0	1181.60
330.0	189.5	1178.83
338.0	149.0	1174.23
346.0	96.7	1165.42
354.0	0.1	1158.98
362.0	0.0	1159.02
370.0	0.0	1159.02

Replace with attached Figure

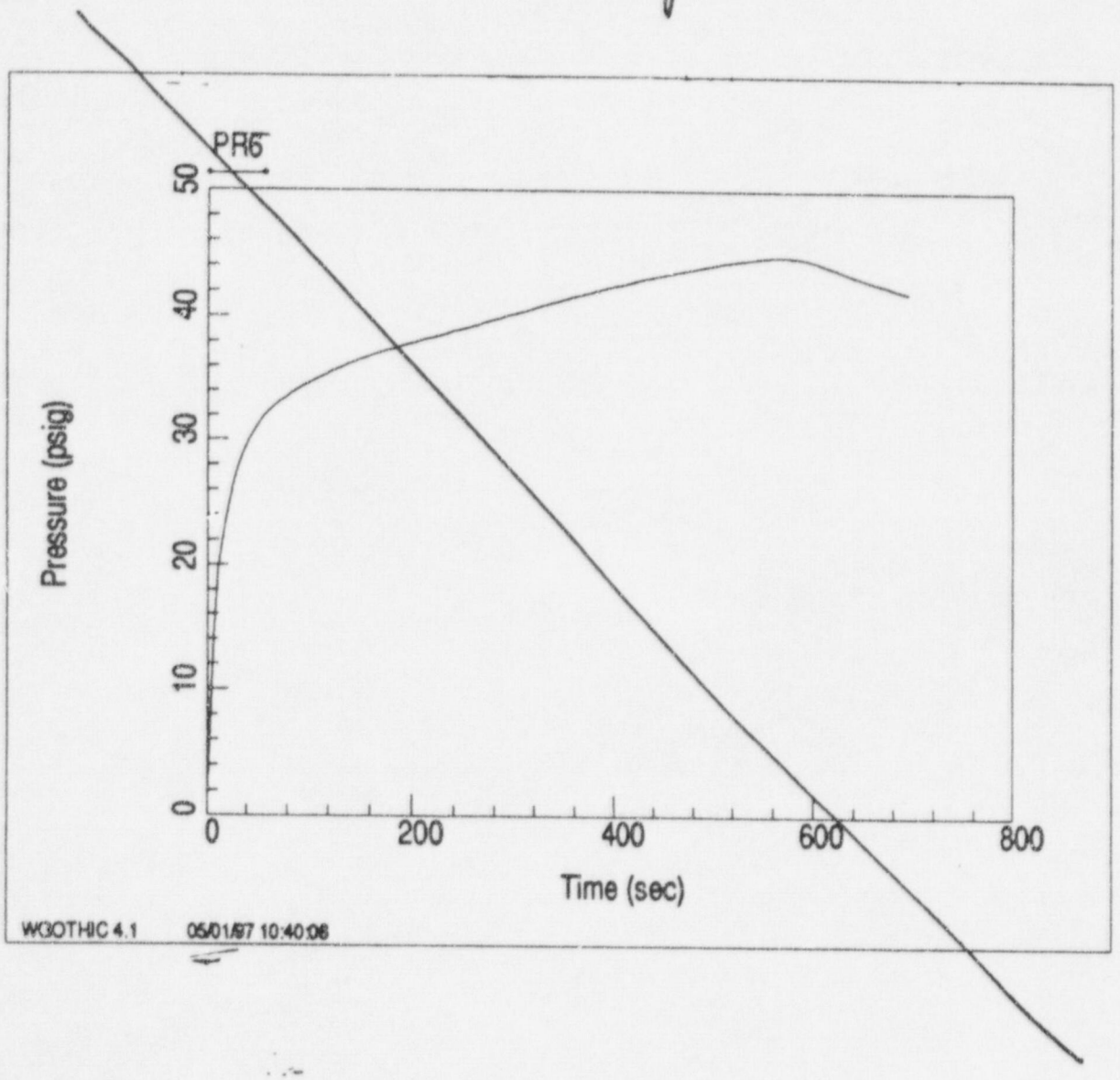
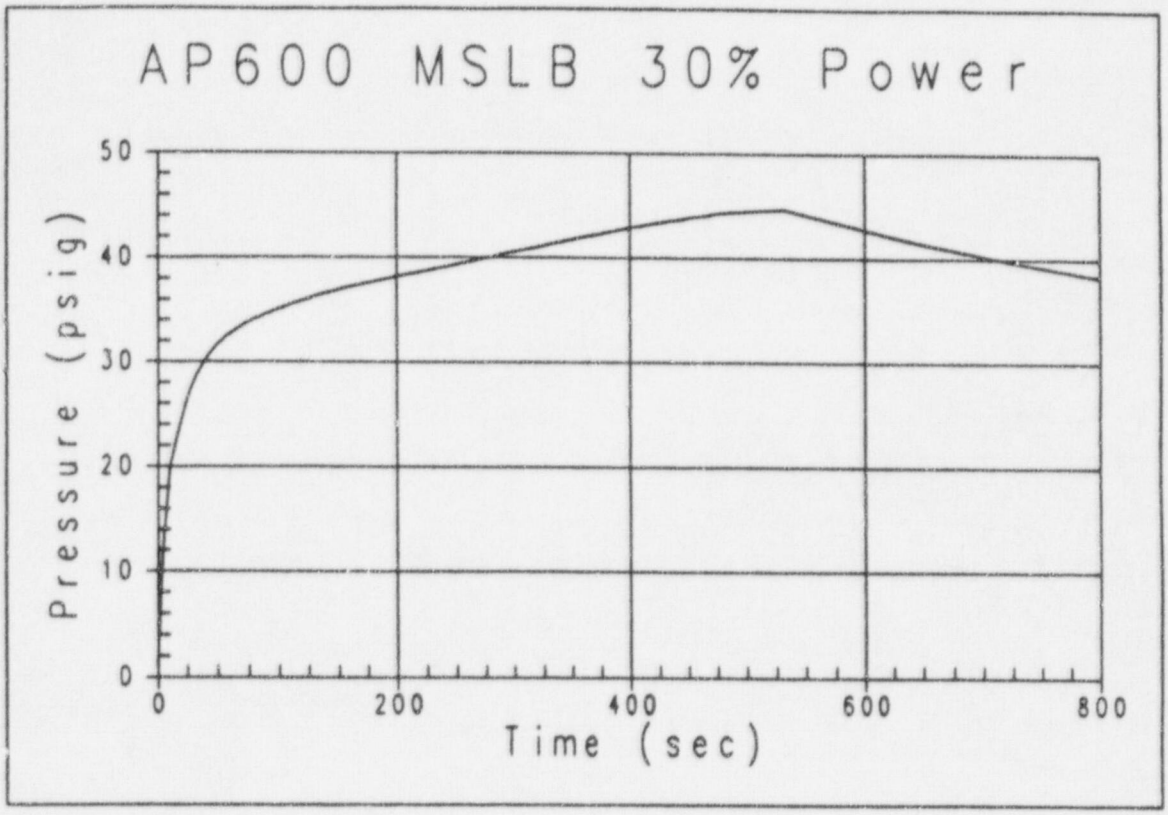


Figure 6.2.1.1-1

MSLB Full DER 30% Power Case

NEW FIGURE 6.2.1.1-1



Replace with attached figure

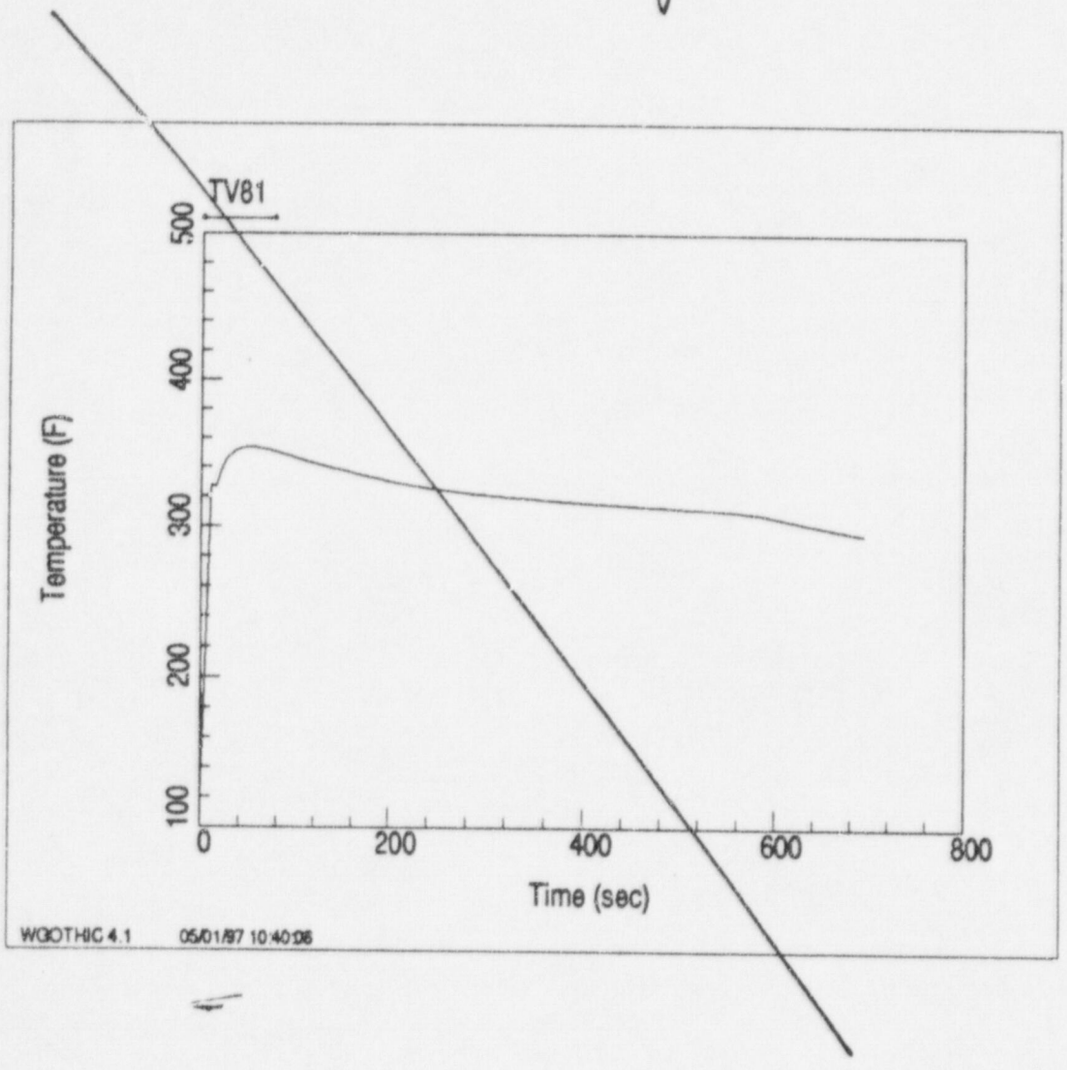
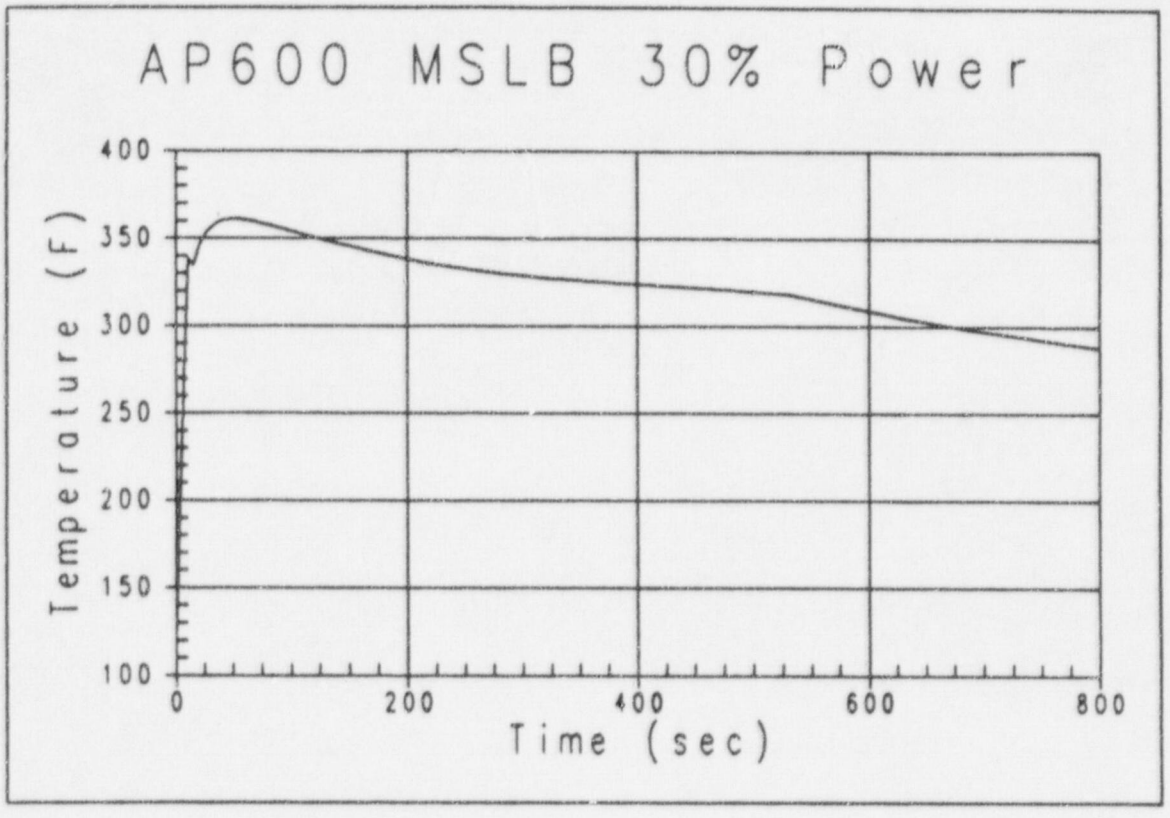


Figure 6.2.1.1-2

MSLB full DER 30% Power Case

NEW FIGURE 6.2.1.1-2



Replace with attached figure

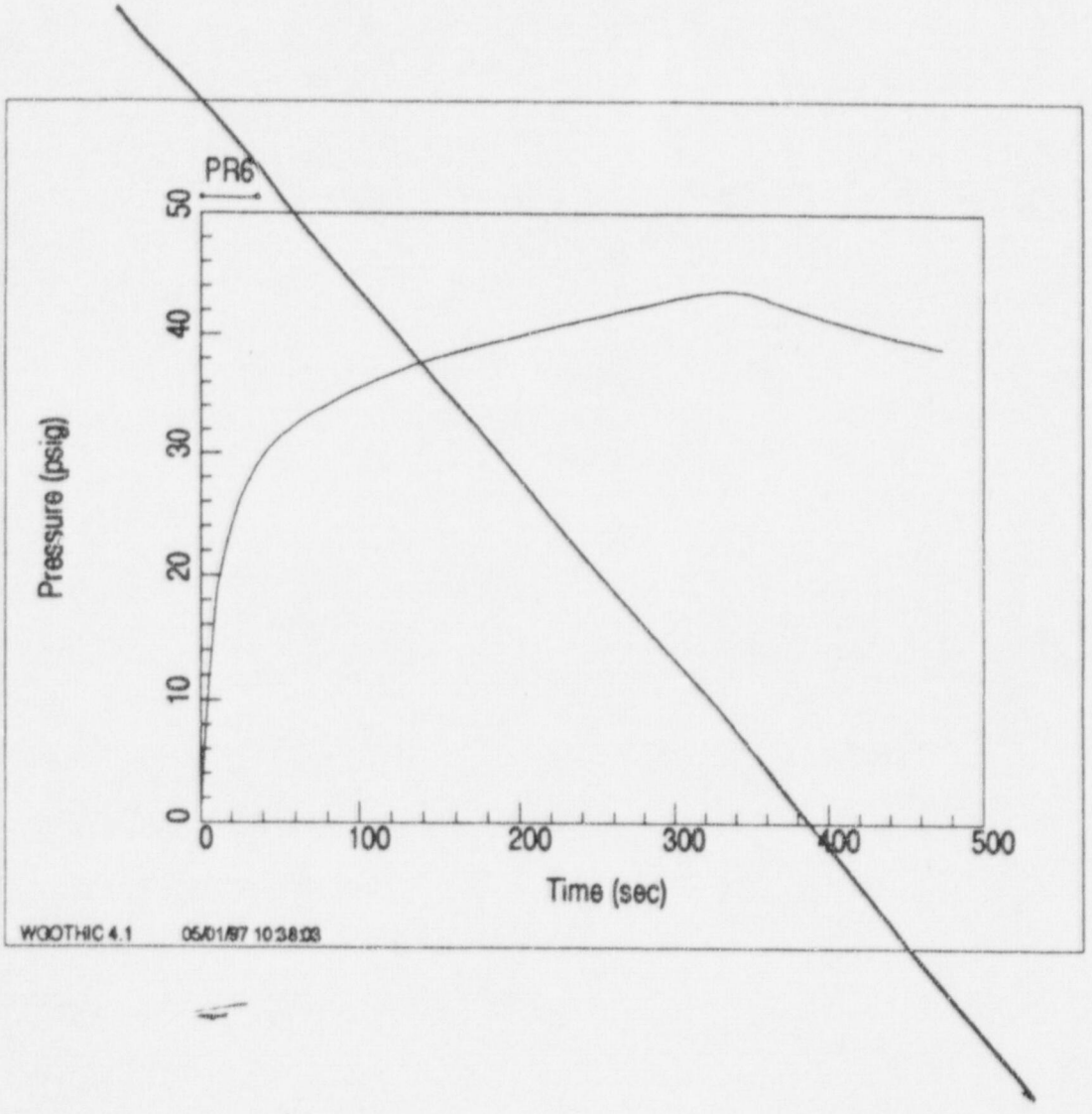
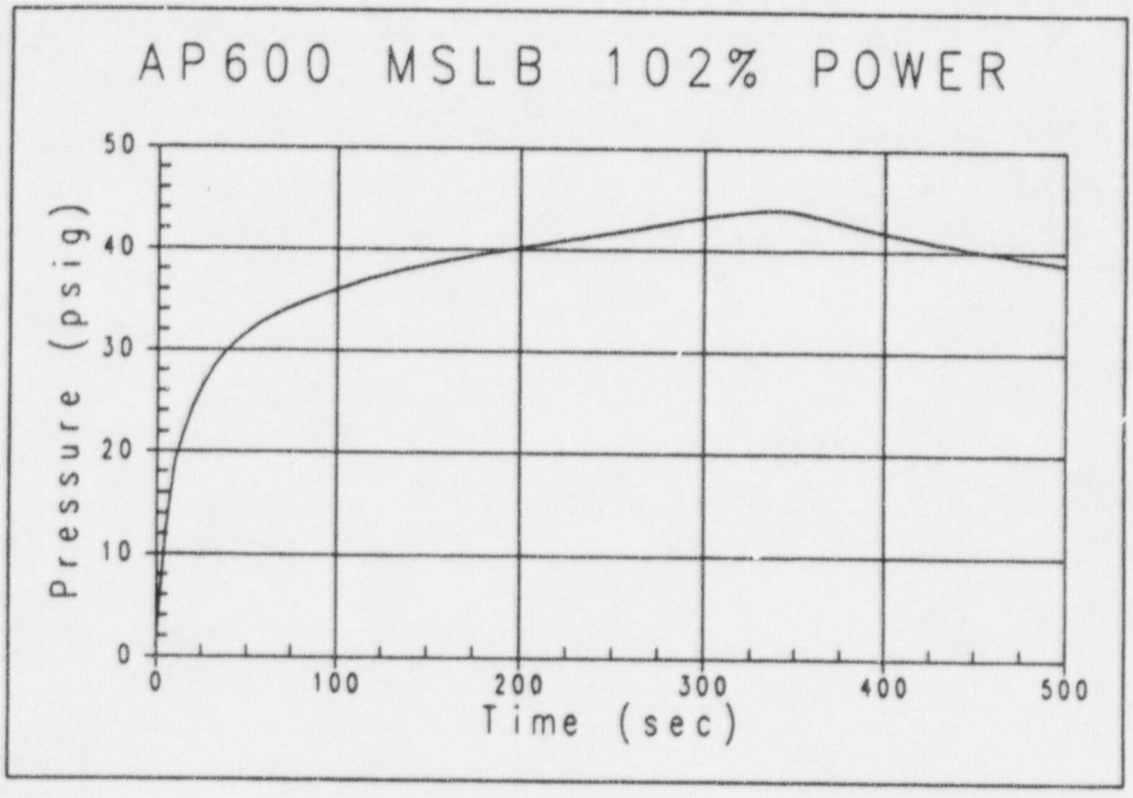


Figure 6.2.1.1-3

MSLB Full DER 102% Power Case

NEW FIGURE 6.2.1.1-3



Replace with attached figure

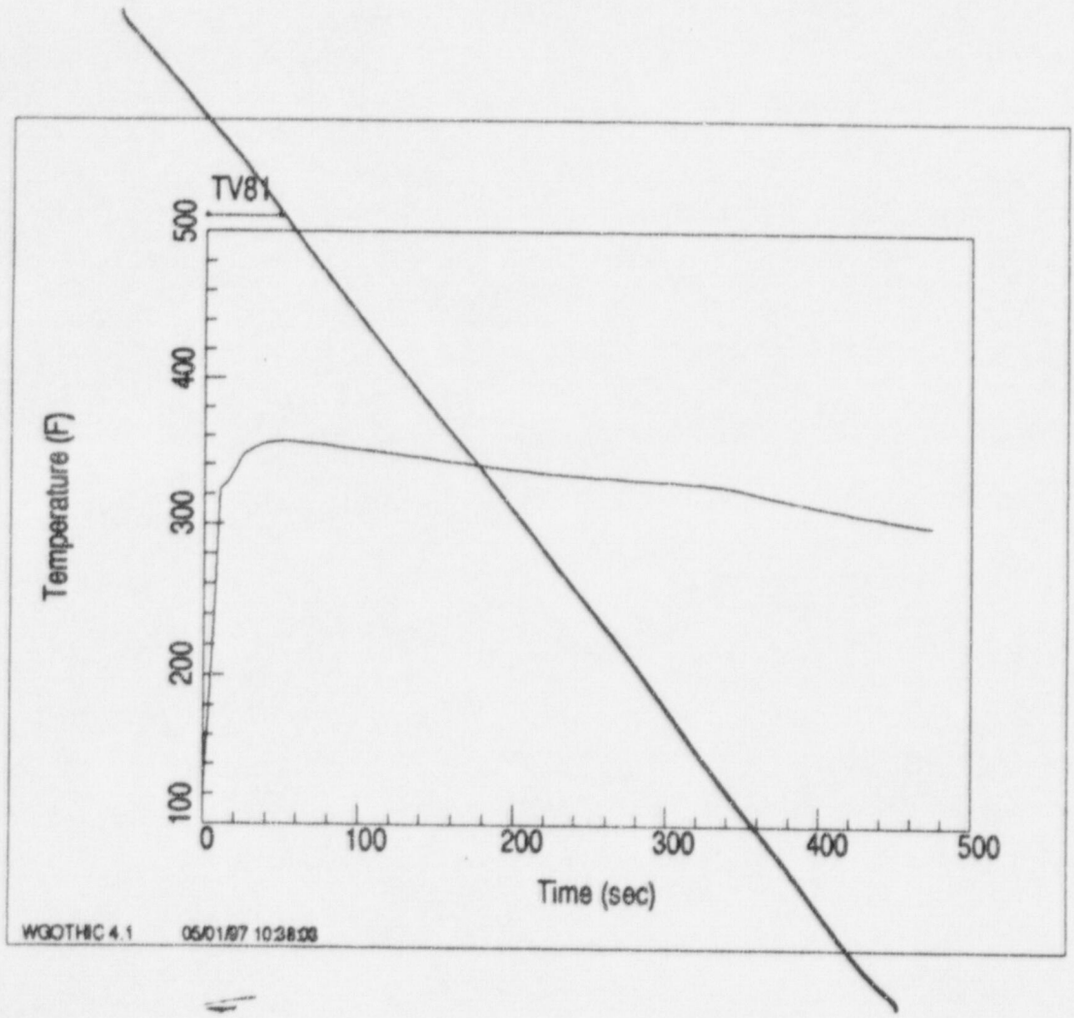
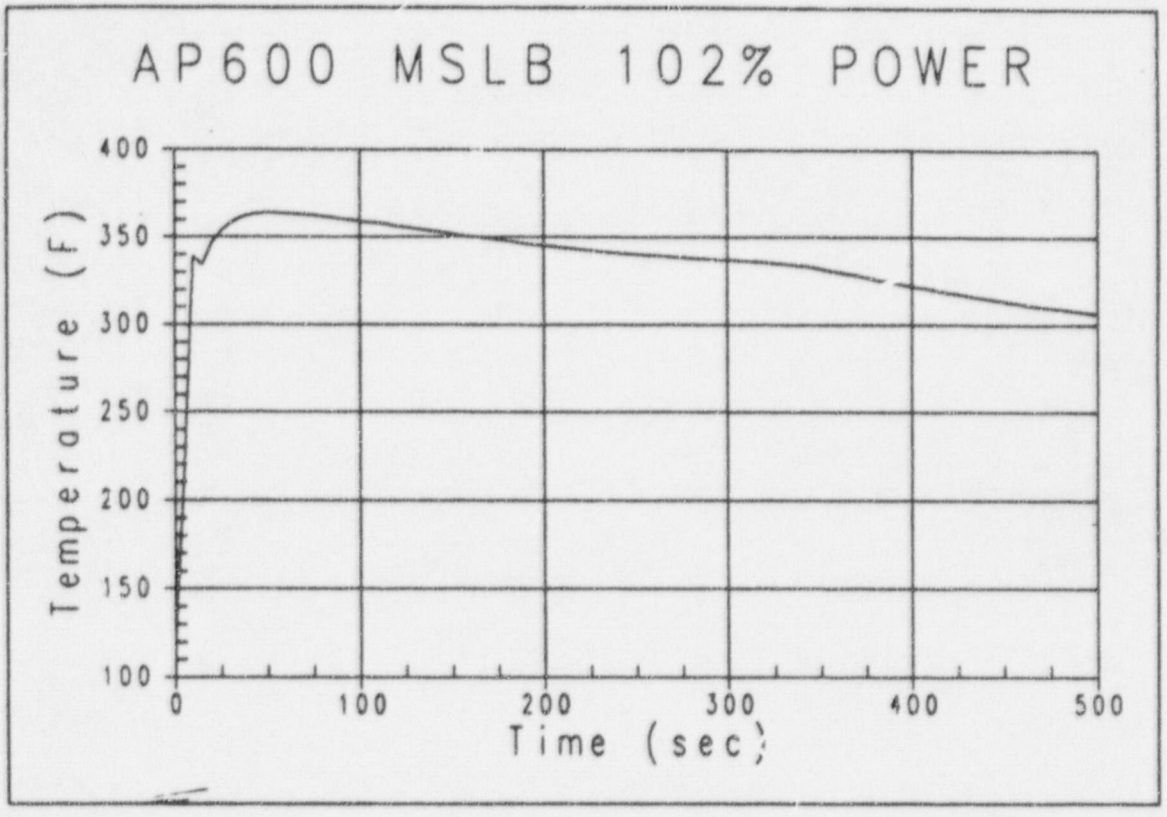


Figure 6.2.1.1-4

MSLB Full DER 102% Power Case

NEW FIGURE 6.2.1.1-4



Replace

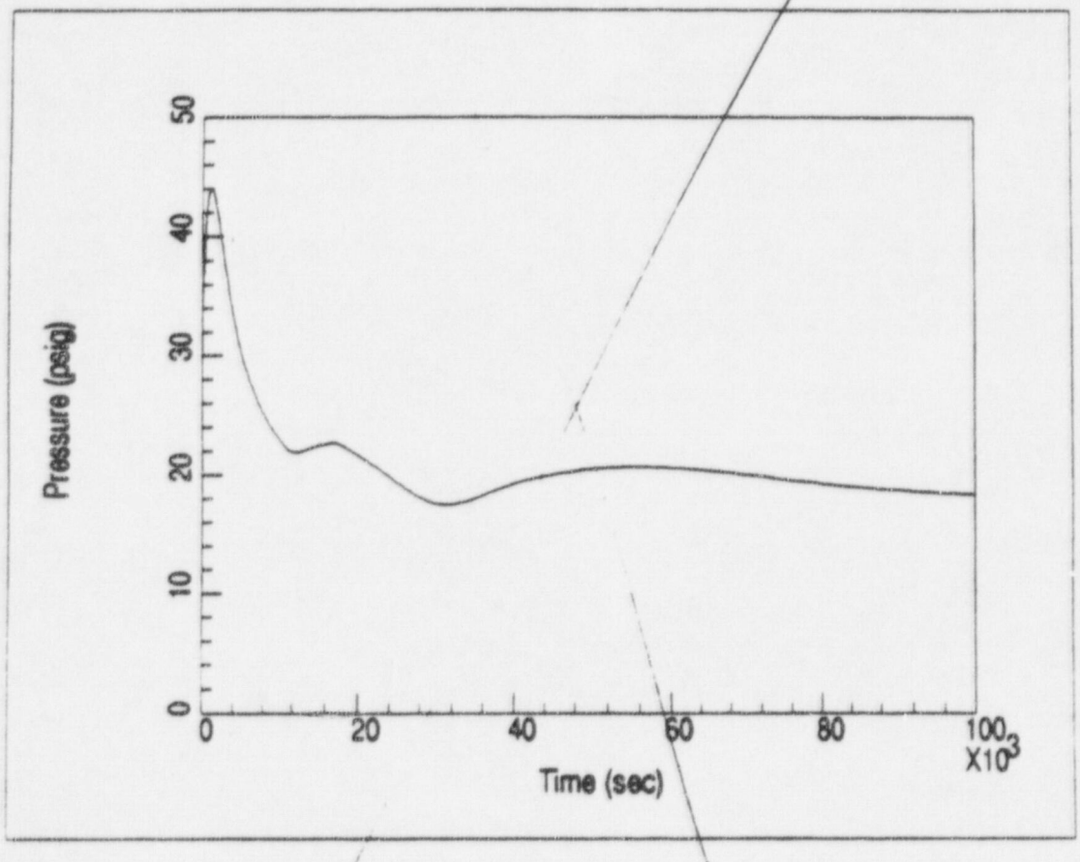
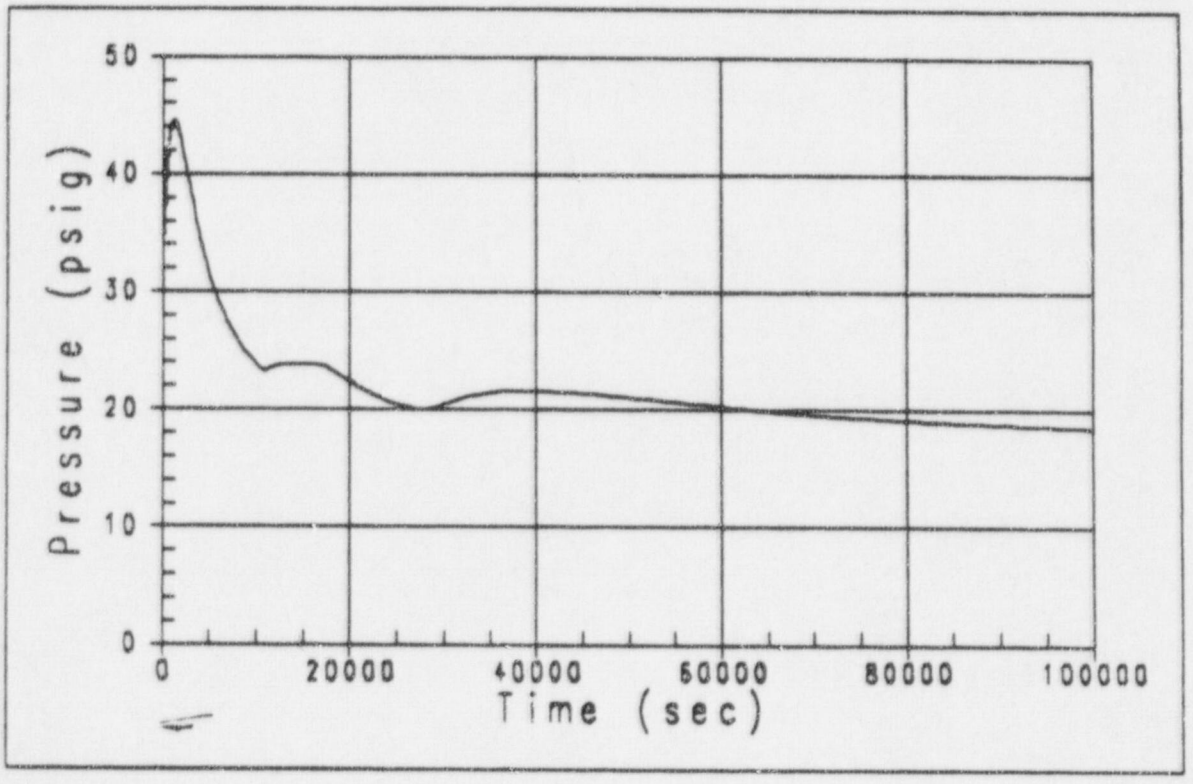


Figure 6.2.1.1-5

DECLG LOCA Case for 24 Hours



New Fig. 6.2.1.1-5

DECLG LOCA case for 24 Hours

2/13/98

Replace

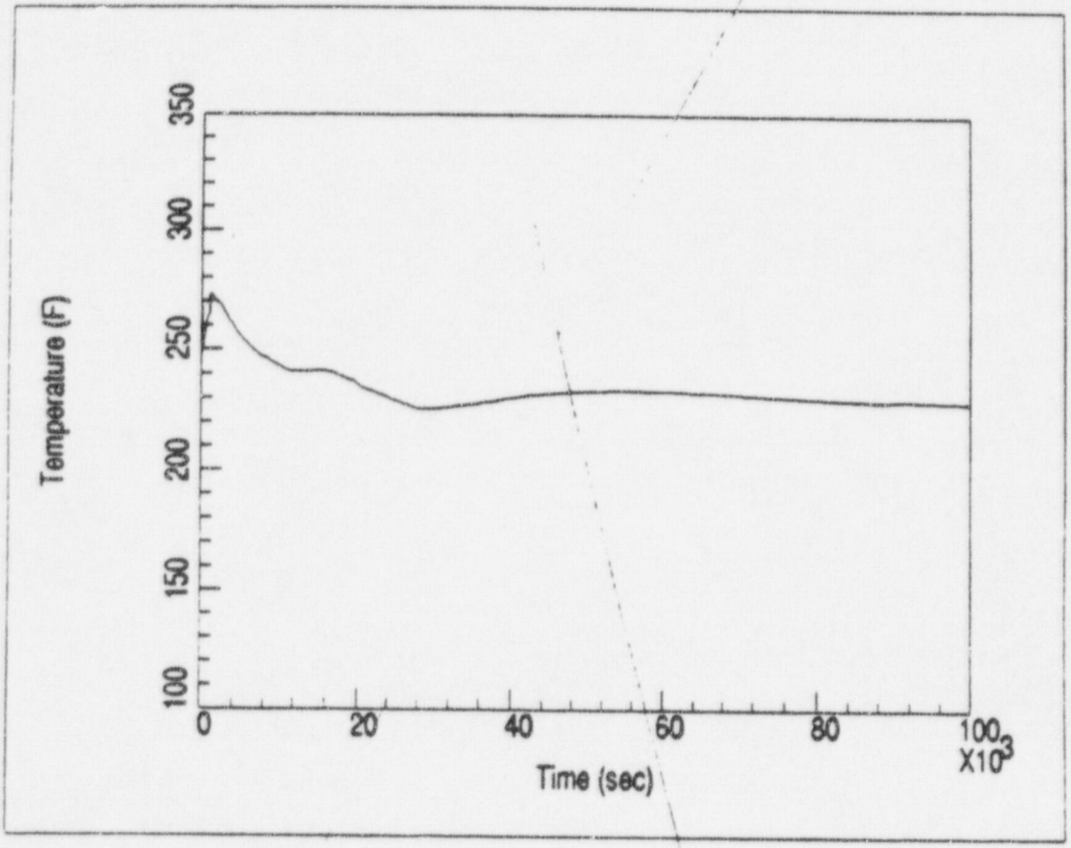
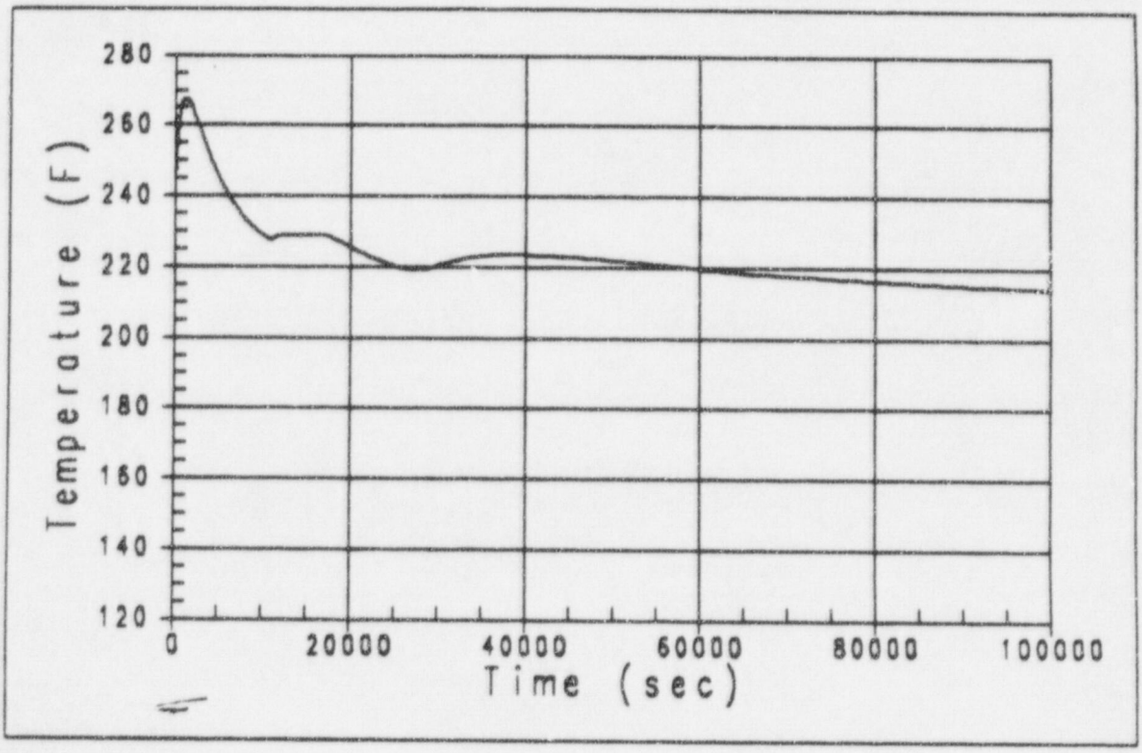


Figure 6.2.1.1-6

DECLG LOCA Case for 24 Hours



New Fig 6.2.1.1-6

DECLG LOCA Case for 24 Hours

2/13/98

Replace

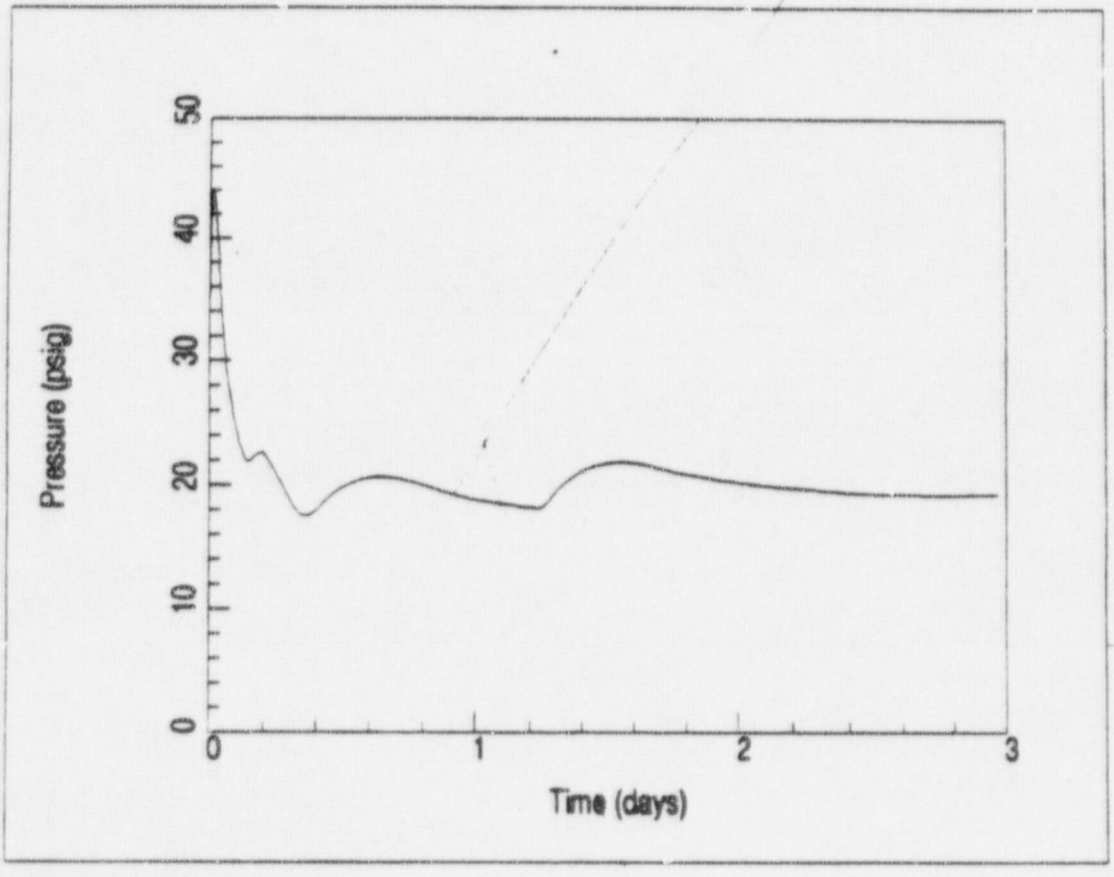
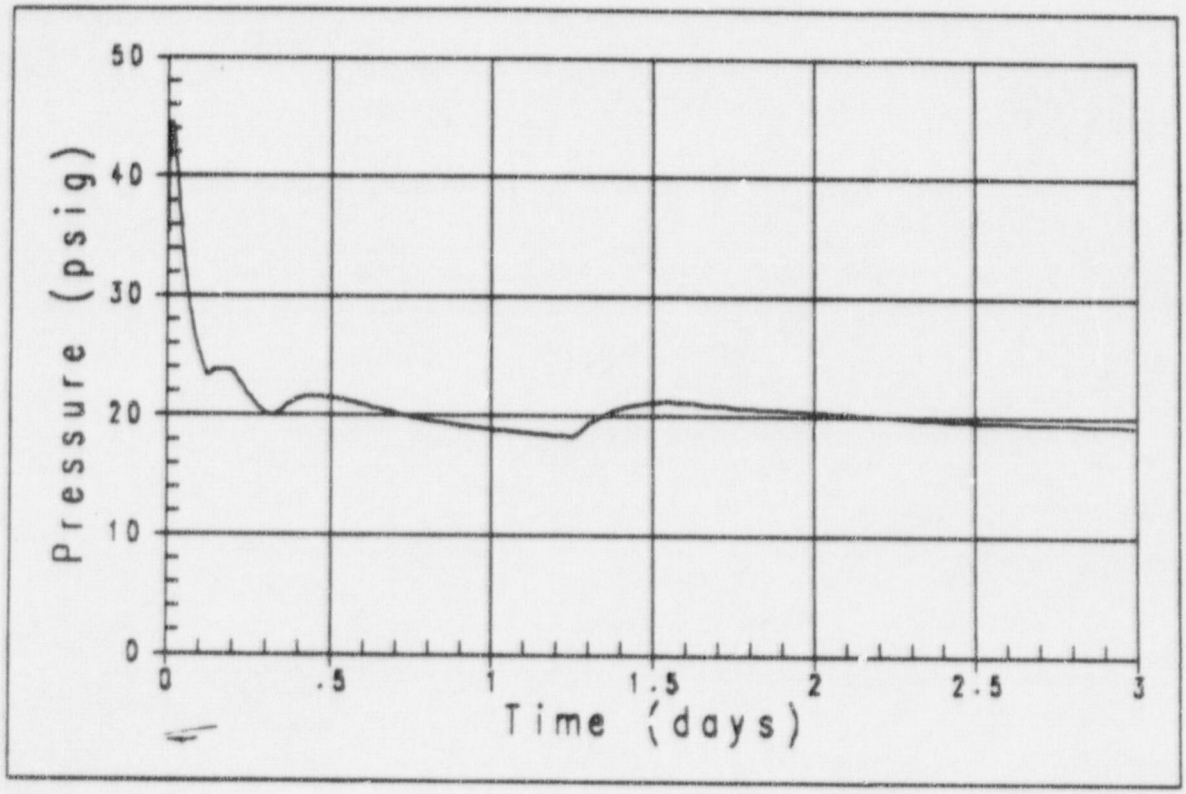


Figure 6.2.1.1-7

DECLG LOCA Case for 3 Days



New Fig 6.2.1.1-7

DECLG LOCA Case for 3 Days

2/15/98

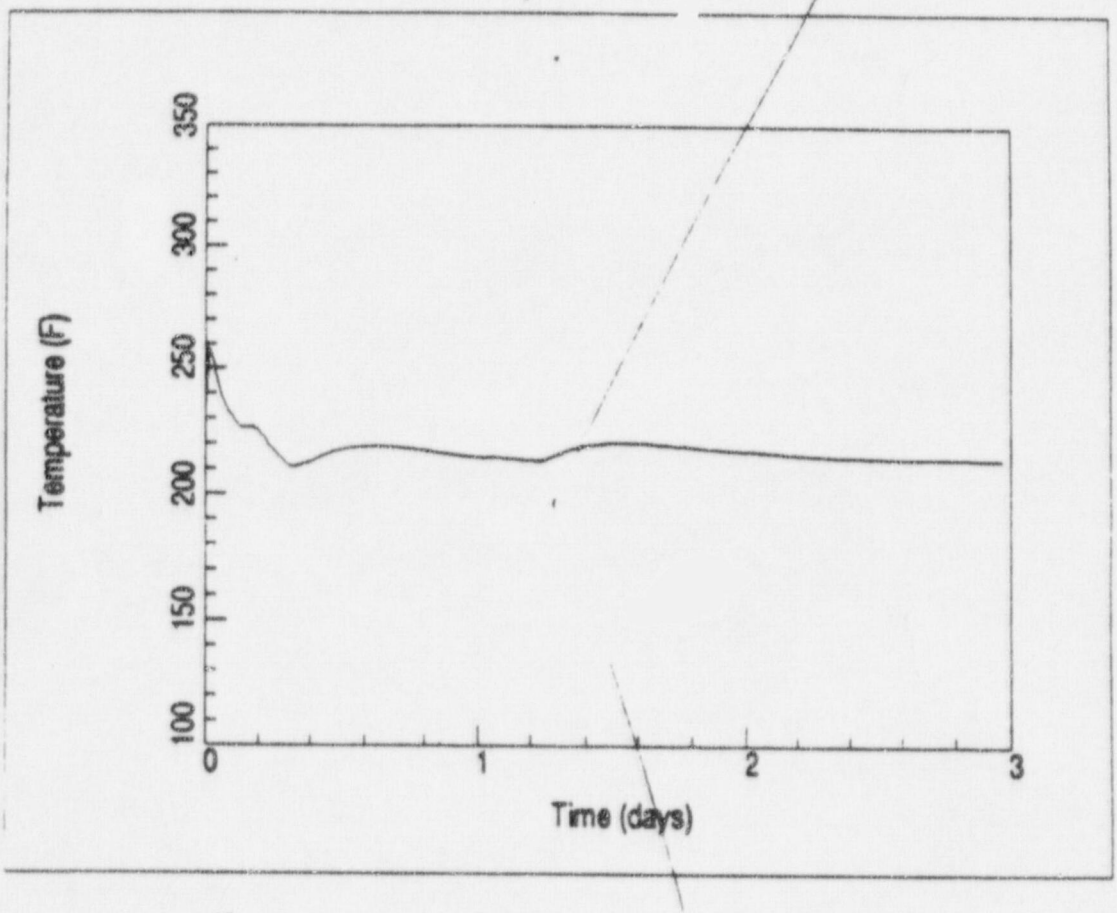
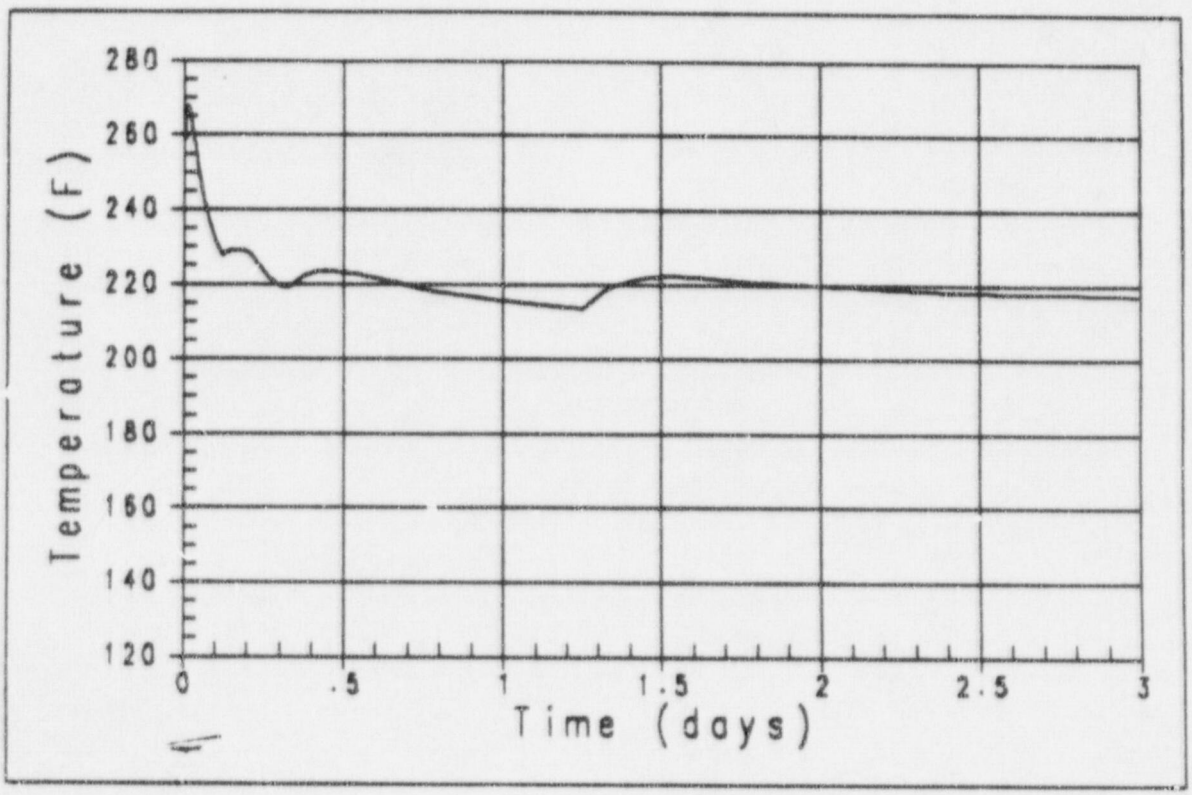


Figure 6.2.1.1-8

DECLG LOCA Case for 3 Days



New Fig 6.2.1.1-8

DECLG LOCA Case for 3 days

2/13/98

Replace

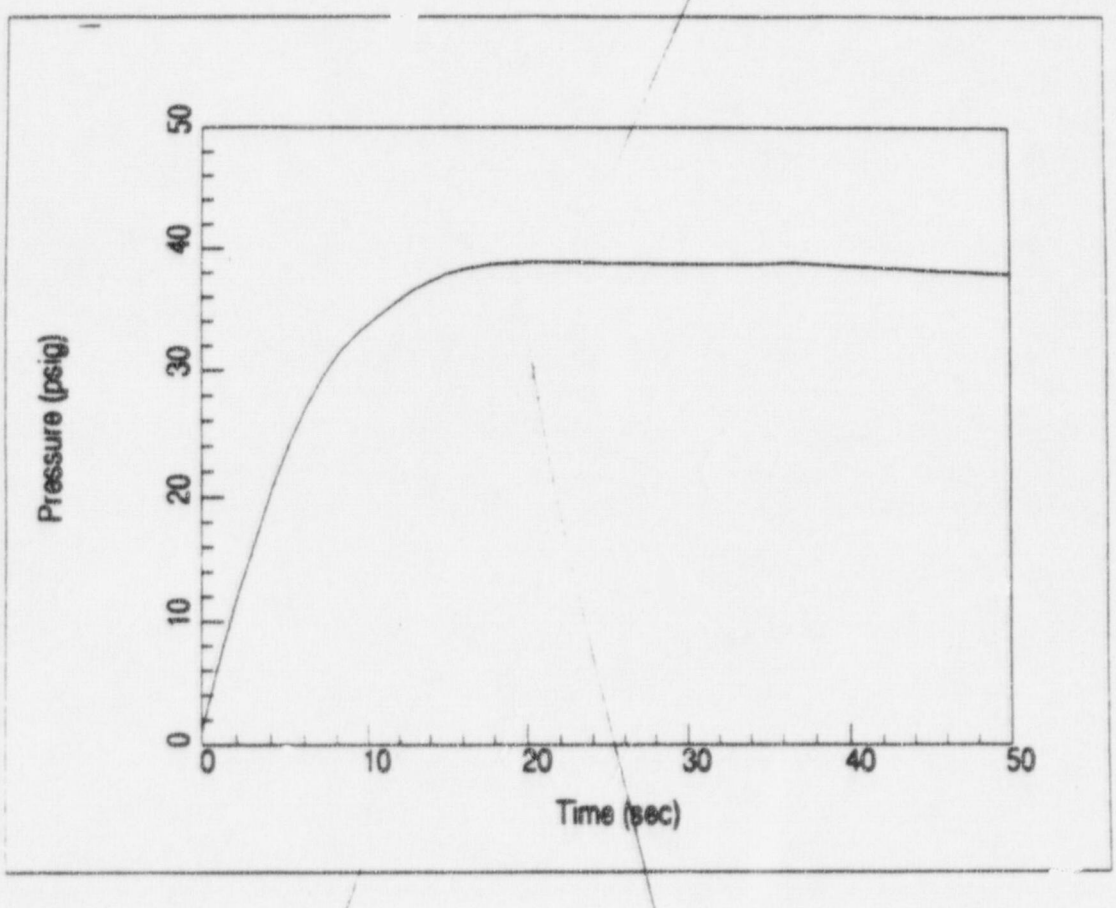
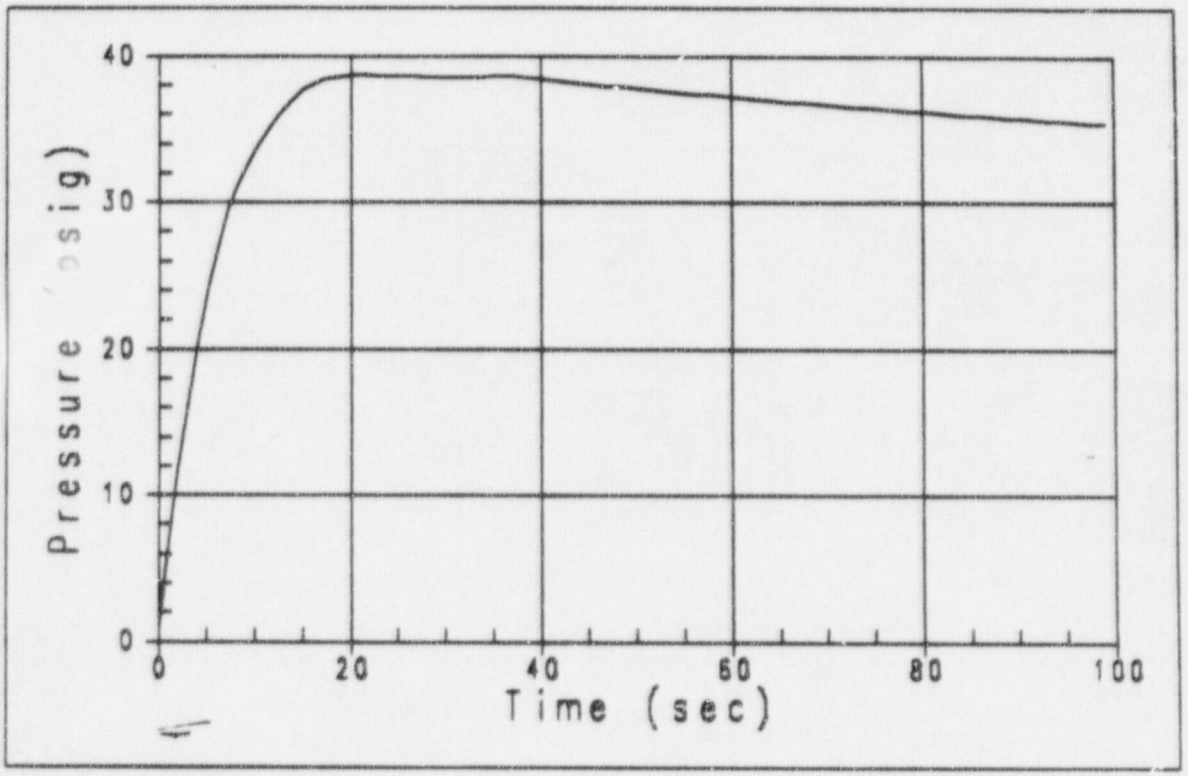


Figure 6.2.1.1-9

DEHLG LOCA Case



New Fig 6.2.1.1-9
DEHLG LOCA Case

2/13/98

Replace

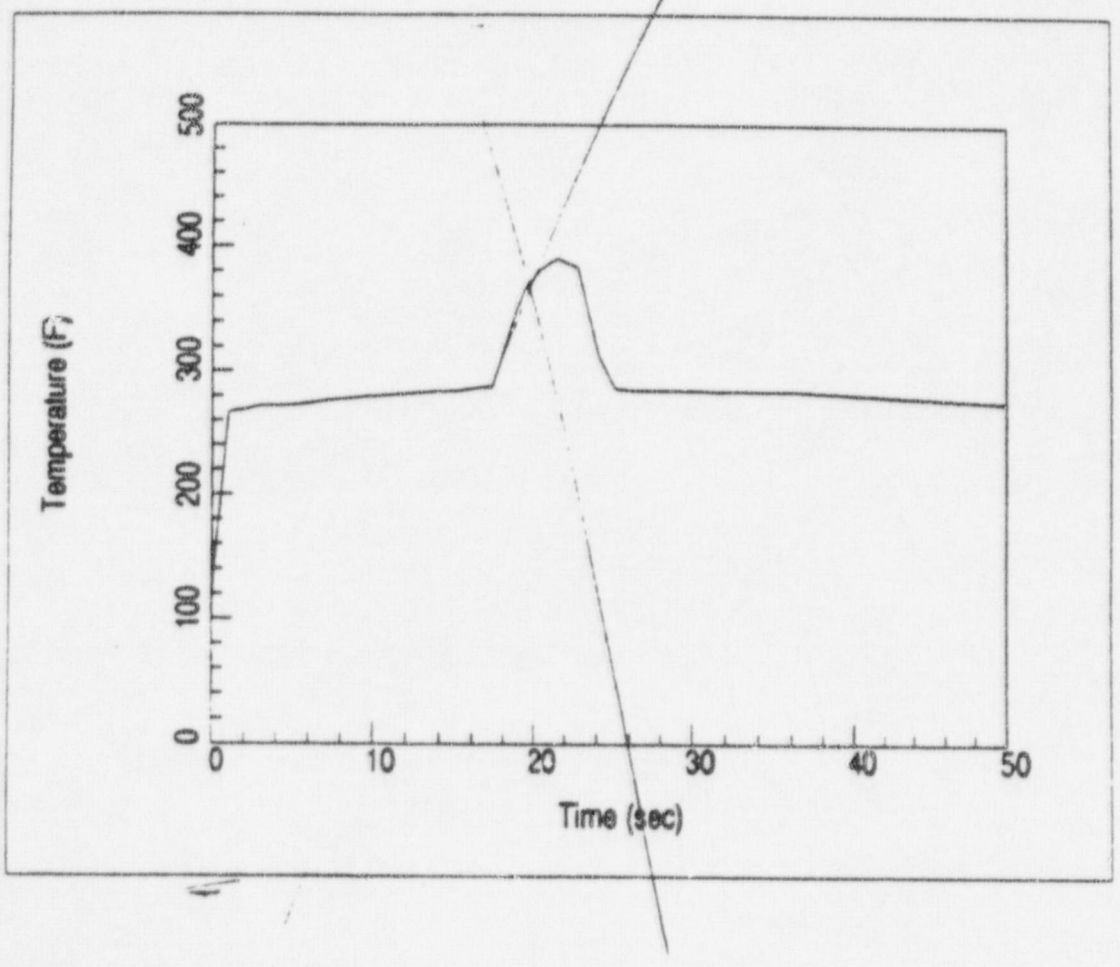
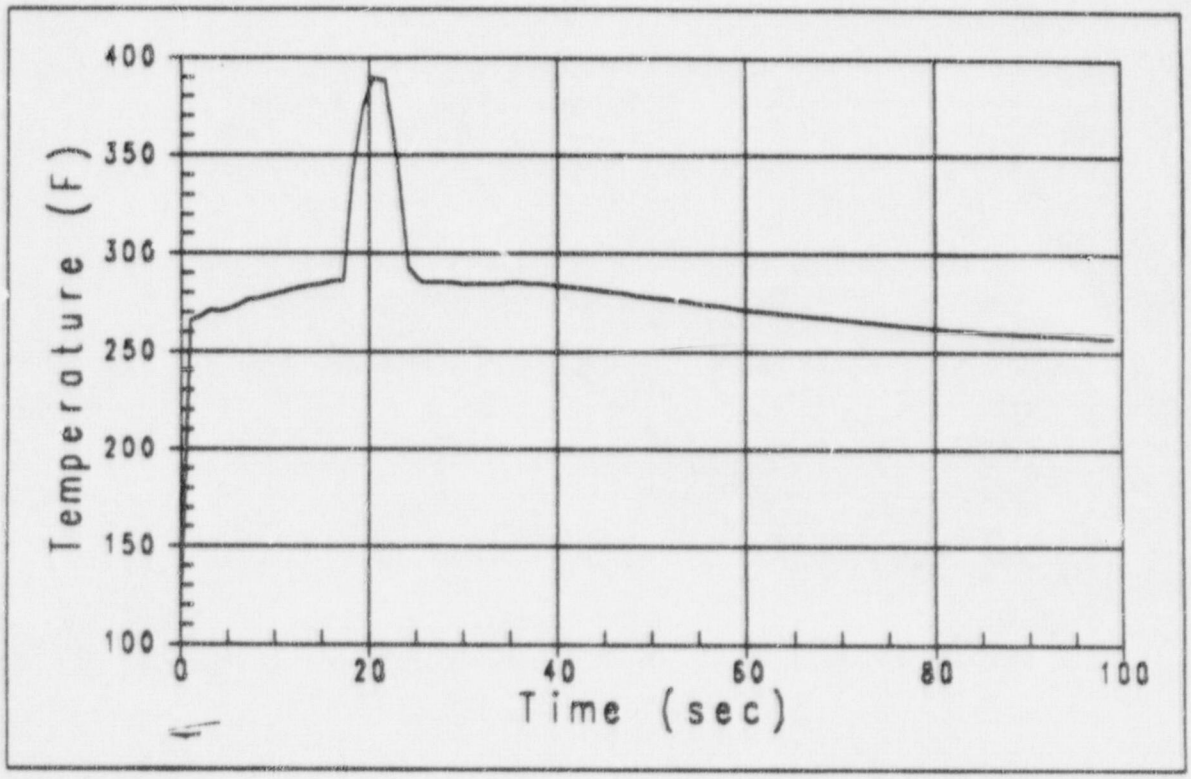


Figure 6.2.1.1-10

DEHLG LOCA Case



Next Fig. 6.2.1.1-10

DEHLG LOCA Case

2/13/90

Replace

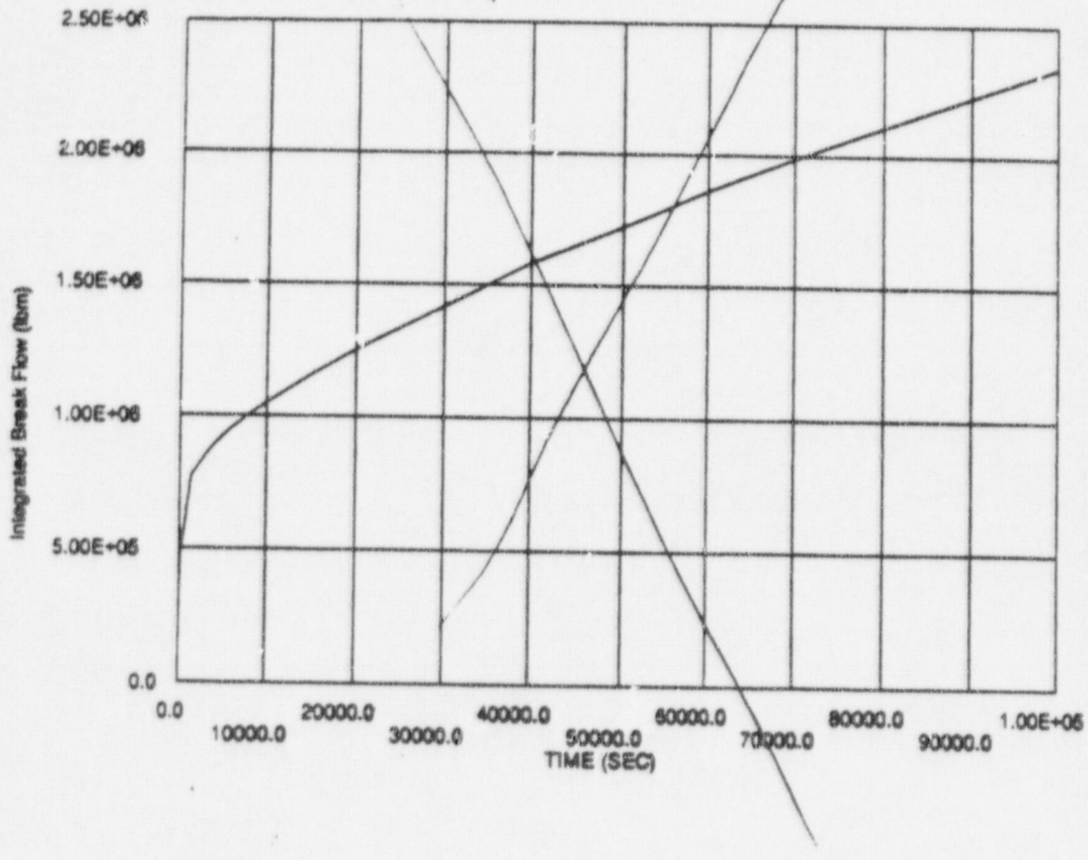
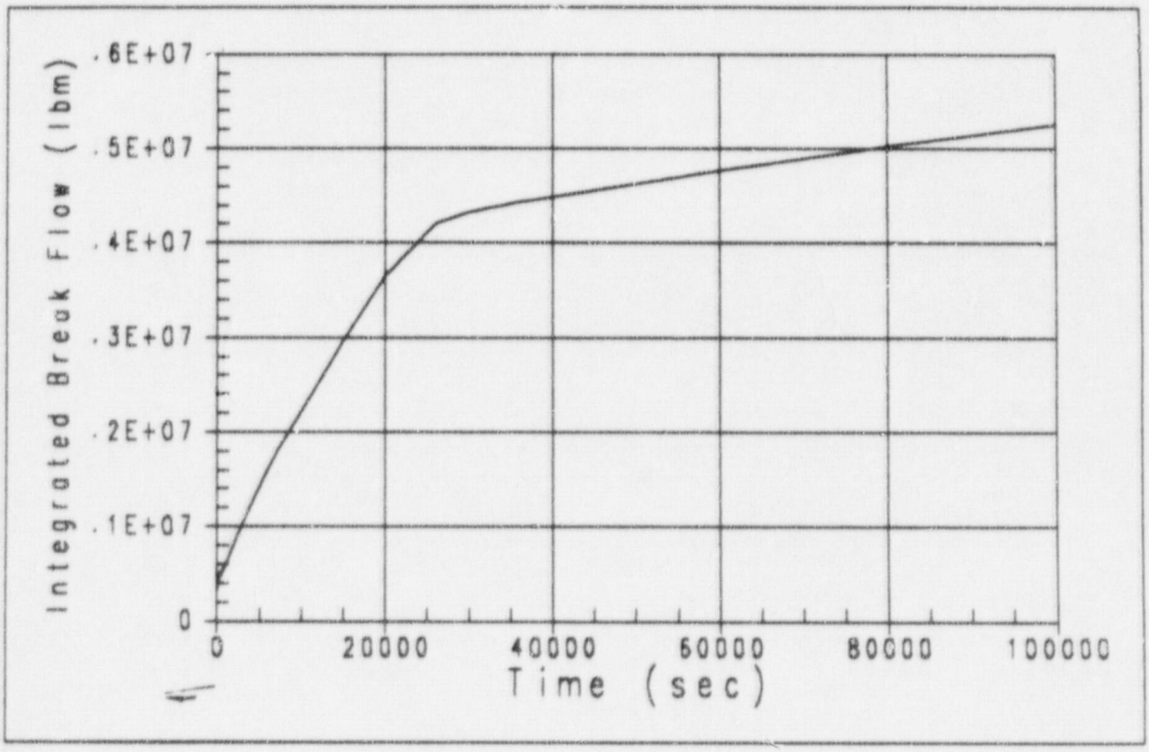


Figure 6.2.1.3-1

Integrated Mass Released for DECLG Break



New figure 6.2.1.3-1

Replace

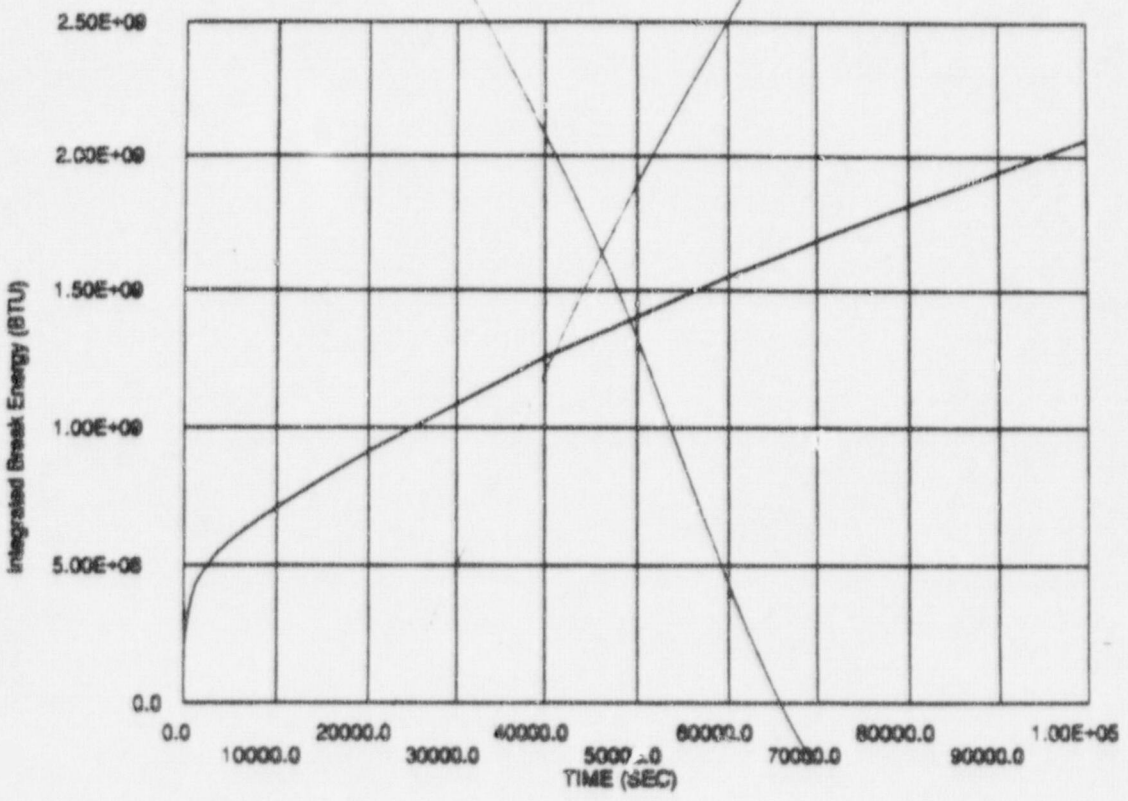
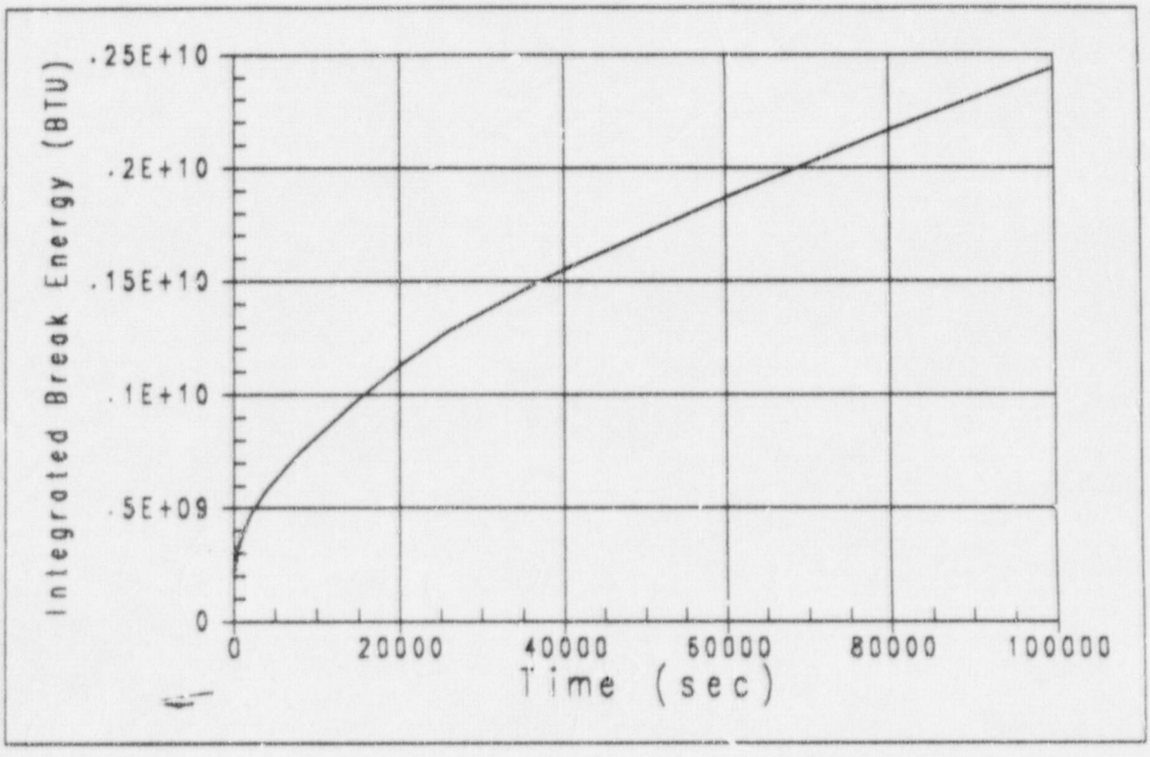


Figure 6.2.1.3-2

Integrated Energy Released for DECLG Break


Integrated Break Energy (BTU)
.25E+10
.2E+10
.15E+10
.1E+10
.5E+09
0
0
20000
40000
60000
80000
100000
Time (sec)

6.2.1.3-2

Enclosure 3 to Westinghouse
Letter DCP/NRC1286

March 6, 1998

Open Item 640.110 (OITS #5864)

REACTOR COOLANT SYSTEM

In Table 2.1.2-4, item #8(c), ADS piping flow resistance values given in the acceptance criterion cannot be found in the SSAR.

Response:

The ADS piping resistance is used in the Chapter 15 accident analyses. The piping resistance is not included explicitly in the SSAR, however, SSAR subsection 5.4.6.3.1 provides a basis for these piping resistances. *This subsection is revised to add a discussion for the ADS stage 4 paths and to make it consistent with SSAR subsection 6.3.6.1.1. Note that this subsection includes a statement that the calculated flow resistances are bounded by the resistances used in the Chapter 15 safety analysis.*

SSAR Revision:

Revision to 5.4.6.3.1 is attached.

ITAAC Revision:

None.



Revision to SSAR Subsection 5.4.6.3.1

5.4.6.3 Inspection and Testing Requirements

The requirements for tests and inspections for reactor coolant system valves is found in subsection 5.4.8.4. In addition, tests for the automatic depressurization system valves and piping are conducted during preoperational testing of the passive core cooling system, as discussed in Sections 6.3 and 14.2.

5.4.6.3.1 Flow Testing

Initial verification of the resistance of the automatic depressurization system piping and valves is performed during the plant initial test program. A low pressure flow test and associated analysis is conducted to determine the total piping flow resistance of each automatic depressurization system valve group connected to the pressurizer (i.e. stages 1-3) from the pressurizer through the outlet of the downstream valve. The reactor coolant system shall be at cold conditions with the pressurizer full of water. The normal residual heat removal pumps will be used to provide injection flow into the reactor coolant system, discharging through the ADS valves.

Inspections and associated analysis of the piping flow paths from the discharge of the automatic depressurization system valve groups connected to the pressurizer (i.e. stages 1-3) to the spargers *are* conducted to verify the line routings are consistent with the line routings used for design flow resistance calculations. The calculated piping flow resistances from the pressurizer through the sparger, with valves of each group open *are bounded by the resistances used in the Chapter 15 safety analysis.*

Inspection of the piping flow paths from each hot leg through the automatic depressurization stage 4 valves is conducted. The calculated flow resistances with valves in each group open are bounded by the resistances used in the Chapter 15 safety analysis.

Open Item 640.117 (OITS #5871)

PASSIVE CORE COOLING SYSTEM

The orifices R01B and R02B are used to establish the required flow rates assumed in the CMT tanks and the accumulators design. They should be shown in the Figure and their sizes should be verified by ITAAC.

Response:

The *resistance of the* orifices in the accumulators and core makeup tanks *lines* are covered by the existing ITAAC's that verify the flow resistances of these lines. The ITAAC's cover the lines from the discharge of the CMTs and the accumulators to the connection to the reactor vessel, which includes the lines containing these orifices. There is no requirement for how much resistance is in the orifice as compared to the other components in the line.

The PXS figure (sheet 2) shown in the ITAAC will be revised to show these orifices.

SSAR Revisions:

None

ITAAC Revisions:

The PXS figure contained in the ITAAC (figure 2.2.3-1, sheet 2) is revised to show these orifices.

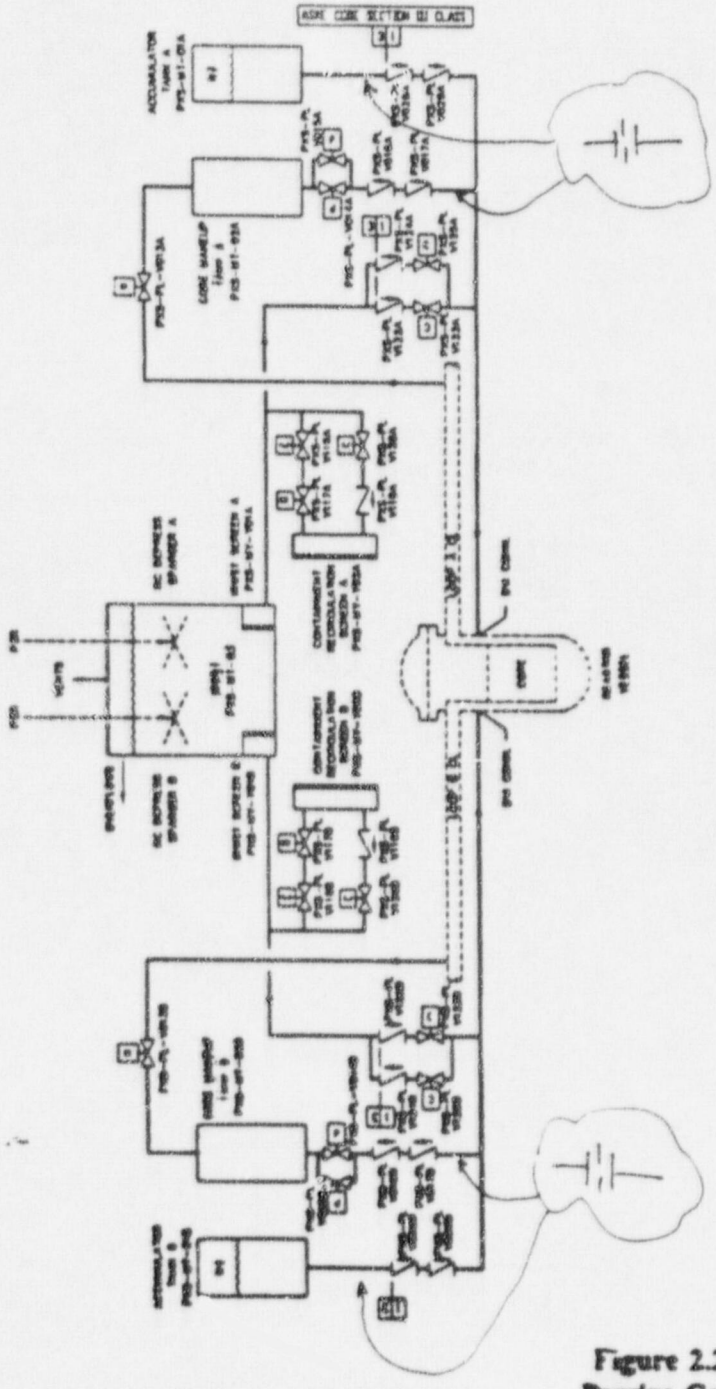


Figure 2.2.3-1 (Sheet 2 of 2)
Passive Core Cooling System

Open Item 640.121 (OITS #5875)

PASSIVE CORE COOLING SYSTEM

Add IRWST injection line number PXS-L123A and B to Table 2.2.3-2 (p. 2.2.3-9).

Response:

These IRWST injection line numbers *will be added* to Table 2.2.3-2.

SSAR Revisions:

None

ITAAC Revisions:

Revised Table 2.2.3-2 to add these lines.

Certified Design Material

Table 2.2.3-2

Line Name	Line Number	ASME Code Section III	Leak Before Break	Functional Capability Required
PRHR HX inlet line from hot leg and outlet line to steam generator channel head	RCS-L134, PXS-L102, PXS-L103, PXS-L104A, PXS-L104B, PXS-L105, PXS-L107, RCS-L114, RCS-L113	Yes	Yes	Yes
CMT A inlet line from cold leg C and outlet line to reactor vessel direct vessel injection (DVI) nozzle A	RCS-L118A, PXS-L007A, PXS-L015A, PXS-L016A, PXS-L017A, PXS-L018A, PXS-L019A, PXS-L020A, PXS-L021A, PXS-L070A	Yes	Yes	Yes
CMT B inlet line from cold leg D and outlet line to reactor vessel DVI nozzle B	RCS-L118B, PXS-L007B, PXS-L015B, PXS-L016B, PXS-L017B, PXS-L018B, PXS-L019B, PXS-L020B, PXS-L021B, PXS-L070B	Yes	Yes	Yes
Accumulator A discharge line to DVI line A	PXS-L025A	Yes	Yes	Yes
	PXS-L027A, PXS-L029A	Yes	No	Yes
Accumulator B discharge line to DVI line B	PXS-L025B	Yes	Yes	Yes
	PXS-L027B, PXS-L029B	Yes	No	Yes
IRWST injection line A to DVI line A	PXS-L123A, PXS-L127A	Yes	Yes	Yes
	PXS-L124A, PXS-L117A, PXS-L116A, PXS-L112A	Yes	No	Yes
IRWST injection line B to DVI line B	PXS-L124B, PXS-L117B, PXS-L116B, PXS-L114B, PXS-L112B	Yes	No	Yes
	PXS-L123B, PXS-L127B	Yes	Yes	Yes
Containment recirculation line A	PXS-L114A, PXS-L131A, PXS-L132A	Yes	No	Yes
Containment recirculation line B	PXS-L114B, PXS-L131B, PXS-L132B	Yes	No	Yes

L123A

L123B

AP600

Open Item 640.123 (OITS #5877)

PASSIVE CORE COOLING SYSTEM

Accumulator level sensors PXS-021, 022, 023, and 024 (Table 2.2.3-3, Page 2.2.3-11) are not shown in SSAR Figure 6.3-1,2. Because all of the information provided in Tier 1 should also be in Tier 2, revise the SSAR Figure to show the accumulator level sensors.

Response:

These level sensors are contained in the SSAR. They are discussed in notes on the SSAR PXS P&ID figure. They are also discussed in SSAR subsection 6.3.7.4 and listed in the post accident monitoring table 7.5-1. The information provided in these SSAR sections is similar to that provided in the ITAAC. The P&ID depiction of instruments provides additional information (sensor type) that is not necessary in the ITAACs. It is not necessary to show these instruments on the PXS P&ID figure in the SSAR. The P&ID depiction of instruments provides a level of detail (sensor type, connection to system) that is not necessary in the ITAACs. *The accumulator level instrument number will be added to the note on the PXS P&ID SSAR figure 6.3-2 (sheet 1).*

SSAR Revisions:

e note on SSAR figure 6.3-2 (sheet 1).

ITAAC Revisions:

None



Revision to SSAR Figure 6.3-2, Sheet 1

Note 6:

Current - Two redundant accumulator water level instruments are provided covering span from full to empty for main control indication and alarm.

Revised - Two redundant accumulator water level instruments are provided *for each accumulator (Accum A - L021, L023, Accum B - L022, L024)* covering span from full to empty for main control *room* indication and alarm.

Open Item 640.124 (OITS #5878)

PASSIVE CORE COOLING SYSTEM

The calculated flow resistance values given for CMTs, accumulators and the IRWST in the acceptance criteria (Table 2.2.3.4, Page 2.2.3-15, items #8.c and #9) can not be found in the SSAR. Identify the section in SSAR where these values exist or include these values in the SSAR.

Response:

The piping resistances of the injection lines from the CMT's, accumulators and the IRWST are used in the Chapter 15 accident analyses. The piping resistances are not included explicitly in the SSAR, however, SSAR subsection 6.3.1.1 provides a basis for these flow resistances. *This subsection is revised to make it consistent with SSAR subsection 5.4.6.3. Note that this subsection includes a statement that the calculated flow resistances are bounded by the resistances used in the Chapter 15 safety analysis.*

SSAR Revisions:

Revise SSAR subsection 6.3.6.1.1

ITAAC Revisions:

None



Revised SSAR Subsection 6.3.6

6.3.6 Inspection and Testing Requirements

6.3.6.1 Preoperational Inspection and Testing

Preoperational inspections and tests of the passive core cooling system are performed to verify the operability of the system prior to loading fuel. This testing includes valve inspection and testing, flow testing, and verification of heat removal capability.

Preoperational testing of the passive core cooling system is completed in conjunction with testing of the reactor coolant system following flushing and hydrostatic testing, with the system cold and the reactor vessel head removed. The passive core cooling system is aligned for normal power operation. This testing provides the following information:

- Satisfactory safeguards actuation signal generation and transmission
- Valve operating times
- Injection starting times
- Injection delivery rates

The preoperational testing program includes testing of the following passive core cooling system components:

- Core makeup tanks
- Accumulators
- In-containment refueling water storage tank
- Containment recirculation
- Passive residual heat removal heat exchanger

Conformance with the recommendations of Regulatory Guide 1.79 is described in subsection 1.9.1. Preoperational testing of the passive core cooling system is conducted in accordance with the requirements presented in subsection 14.2.9.1.3.

6.3.6.1.1 Flow ~~Testing~~

Initial verification of the resistance of the passive core cooling injection lines is performed by conducting a series of flow tests for the core makeup tanks, accumulators, in-containment refueling water storage tank, and containment recirculation piping. *The calculated flow resistances are bounded by the resistances used in the Chapter 15 safety analysis.*

Open Item 640.164 (OITS #6192)

AP600 is a passive plant and natural circulation is a key feature of the plant. The piping lay-out of the passive systems is critical to the proper operation of the passive system. We understand the detailed piping lay-out of several systems are already finalized including the pipe stress analysis. Confirm that piping slope and elevation are specified in the ITAAC to verify that the passive systems will function as designed. Systems flow resistance and nozzle elevations given in the ITAAC are not sufficient to verify the system function.

Response:

The PXS ITAAC's (Table 2.2.3-4) cover important characteristics of the PXS performance including the arrangement of the lines (Item 1), the elevations of the injection tanks vs the RCS (Item 8.c.iii), and the lines resistances (item 8.c.i). It is not necessary or practical to ITAAC specific detailed pipe routings, such as the number / radius of elbows or specific slopes. Although the routing of these lines and their stress analysis have been completed, it is possible that the as-built configuration *may be* different than the current configuration. The following additional pipe routing ITAACs are provided to verify that the function of the PXS system is not affected by minor routing changes:

1. Slope the line from HL up to PRHR HX inlet line high point.
2. Slope the lines from CL up to CMT inlet line high point.
3. The maximum elevation of the *following* lines must be *limited*:
   - *PRHR HX outlet line to SG connection* - *less than top inside surface of PRHR HX lower channel head*
   - *CMT discharge line to DVI connection* - *less than bottom of CMT*
   - IRWST injection and containment recirculation lines - less than bottom of the IRWST
4. *The minimum elevation of the ADS stage four discharge* - *greater than plant elevation 110'*

The line slope requirements support maintaining the water in these lines hot during plant operation which provides timely initiation of natural circulation flow in these lines when the isolation valves are opened. *The maximum elevation of the PRHR HX provides for effective PRHR HX startup and operation over a range of RCS conditions including those occurring during LOCAs. The maximum elevation of the CMT discharge line allows the CMTs to completely drain.* The maximum elevation of the IRWST injection and containment recirculation lines prevents possible voiding of the water in the lines during long term recirculation operation.

SSAR Changes: PXS P&ID (figures 6.3-1 and -2) *and RCS P&ID (figure 5.1-19)* are revised to indicate these routing requirements. These requirements are contained in notes *on these sheets.*

ITAAC Changes: Add pipe routing ITA and AC in PXS ITAAC, Table 2.3-4, Item 8.c).

PXS SSAR P&ID, Figure 6.3-1 and 2

Sheet 1, Note 16

Current - Slope line continuously upward from the cold leg connection to high point at CMT inlet.

Revised - Route line from cold leg connection to high point without downward sloping sections.

Sheet 1, Note 15

Current - Slope DVI line upward from RV to highpoint before connecting to CMT, IRWST and accum. Locate vent at high point. Locate strap-on RTDs (005/006) on top of pipe close to vent.

Revised - Slope DVI line upward from RV to highpoint before connecting to CMT, IRWST and accum. Locate vent at high point. Locate strap-on RTDs (005/006) on top of pipe close to vent. Maximum elevation of DVI line between RV and CMT is connection to CMT.

Sheet 2, Note 3

Current - [note is blanked out in SSAR]

Revised - Route line from hot leg connection to high point without downward sloping sections.

Sheet 2, Note 30

New - Locate IRWST injection lines (from IRWST to RV) and containment recirculation lines (from containment to IRWST injection lines) at an elevation below the bottom of IRWST.

Sheet 2, Note 3

Current - Route pipe from hot leg continuously upward to high point pipe stub downstream of valve V101. From the high point, route pipe continuously downward through the PRHR HX to the elbow that connects to the SG.

Revised - Route pipe from hot leg continuously upward to high point pipe stub downstream of valve V101. Maximum elevation of the PRHR HX outlet line to the SG connection is less than the PRHR HX lower channel head top.

RCS SSAR P&ID, Figure 5.1-19

Sheet 2, Note 21

Current - The 4th stage ADS valves discharge to the steam generator compartments.

Revised - The 4th stage ADS valves discharge to the steam generator compartments at an elevation greater than 110'.

Table 2.2.3-4 (Cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	ii) A low-pressure test and analysis will be conducted for each CMT to determine piping flow resistance from the cold leg to the CMT. The test will be performed by filling the CMT via the cold leg balance line by operating the normal residual heat removal pumps.	ii) The flow resistance from the cold leg to the CMT is $\leq 7.69 \times 10^{-6}$ ft/gpm^2.
	iii) Inspections of the routing of the following pipe lines will be conducted: *– CMT inlet line, cold leg to high point* *– PRHR HX inlet line, hot leg to high point*	*iii) These lines have no downward sloping sections between the connection to the RCS and the high point of the line.*
	iv) Inspections of the elevation of the following pipe lines will be conducted: *– IRWST injection lines; IRWST connection to DVI nozzles* *– Containment recirculation lines; containment to IRWST lines* *- CMT discharge lines to DVI connection* *- PRHR HX outlet line to SG connection*	*iv) The maximum elevation of the top inside surface of these lines is less than the elevation of:* *- IRWST bottom inside surface* *- IRWST bottom inside surface* *- CMT bottom inside surface* *- PRHR HX lower channel head top inside surface*

Table 2.1.2-4 (Cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
8.c) The RCS provides automatic depressurization during design basis events.	iii) Inspections of each fourth-stage ADS valve will be conducted to determine the flow area through each valve.	iii) The flow area through each fourth-stage ADS valve is ≥ 38 in^2.
	iv) Type tests and analysis will be performed to determine the effective flow area through each stage 1,2,3 ADS valve.	iv) A report exists and concludes that the effective flow area through each stage 1 ADS valve ≥ 4.6 in^2 and each stage 2,3 ADS valve is ≥ 21 in^2.
	v) Inspections of the elevation of the ADS stage 4 valve discharge will be conducted.	*v) The minimum elevation of the bottom inside surface of the outlet of these valves is greater than plant elevation 110 feet.*



Open Item (OITS #6632)

Westinghouse needs to add the CMT inlet sparger to the PXS ITAAC, CDM 2.2.3.

Response:

The CMT inlet configuration will be added to the PXS ITAAC, 2.2.3. SSAR subsection 6.3.2.2.1 will be revised to provide Tier 2 information to support the ITAAC.

SSAR Change:

SSAR subsection 6.3.2.2.1 is revised per attachment.

ITAAC Change:

ITAAC Table 2.2.3-4 is revised per attachment.

Revised SSAR Subsection 6.3.2.2.1

6.3.2.2.1 Core Makeup Tanks

The two core makeup tanks are vertical, cylindrical tanks with hemispherical upper and lower heads. They are made of carbon steel, clad on the internal surfaces with stainless steel. The core makeup tanks are AP600 Equipment Class A and are designed to meet seismic Category I requirements. They are located inside containment on the 107-foot floor elevation. The core makeup tanks are located above the direct vessel injection line connections to the reactor vessel, which are located at an elevation near the bottom of the hot leg.

During normal operation the core makeup tanks are completely filled with borated water and are maintained at reactor coolant system pressure by the cold leg pressure balance line. The temperature of the borated water in the core makeup tanks is about the same as the containment ambient temperature since the tanks are not insulated or heated.

The inlet line from the cold leg is sized for loss of coolant accidents, where the cold legs become voided and higher core makeup tank injection flows are required. The discharge line from each core makeup tank contains a flow-tuning orifice that provides a mechanism for the field adjustment of the injection line resistance. The orifice is used to establish the required flow rates assumed in the core makeup tank design. The core makeup tanks provide injection for an extended time after core makeup tank actuation. The duration of injection will be much longer when the core makeup tanks operate in the water recirculation mode as compared to the steam condensation mode.

Connections are provided for remotely adjusting the boron concentration of the borated water in each core makeup tank during normal plant operation, as required. Makeup water for the core makeup tank is provided by the chemical and volume control system. Samples from the core makeup tanks are taken periodically to check boron concentration.

Each core makeup tank has an inlet diffuser which is designed to reduce steam velocities entering the core makeup tank; thereby minimizing potential water hammer and reducing the amount of mixing that occurs during initial core makeup tank operation. ***The inlet diffuser flow area is ≥ 165 in^2.***

The core makeup tanks are located inside the containment but outside the secondary shield wall. This facilitates maintenance and inspection.

Core makeup tank level and inlet and outlet line temperatures are monitored by indicators and alarms. The operator can take action as required to meet the technical specification requirements for core makeup tank operability.

Revised ITAAC Table 2.2.3-4

Table 2.2.3-4 (Cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
8.c) The PXS provides RCS makeup, boration, and safety injection during design basis events.	***x) Inspection of the as-built CMT inlet diffuser will be conducted.***	***x) The CMT inlet diffuser has a flow area ≥ 165 in^2.***

Open Item (OITS #6633)

Westinghouse needs to add the ADS sparger submergence level in the IRWST to the PXS ITAAC, CDM 2.2.3

Response:

The ADS sparger submergence level in the IRWST will be added to the RCS ITAAC, 2.1.2. The RCS ITAAC contains the existing ADS / sparger ITAACs. SSAR subsection 6.3.2.2.6 will be revised to provide Tier 2 information to support the ITAAC.

SSAR Change:

SSAR subsection 6.3.2.2.6 is revised per attachment.

ITAAC Change:

ITAAC Table 2.1.2-4 is revised per attachment.



Revised SSAR Subsection 6.3.2.2.6

6.3.2.2.6 Depressurization Spargers

Two reactor coolant depressurization spargers are provided. Each one is connected to an automatic depressurization system discharge header (shared by three automatic depressurization system stages) and submerged in the in-containment refueling water storage tank. Each sparger has four branch arms inclined downward. The ***connection of the sparger branch*** arms ***to the sparger hub*** are submerged below the in-containment refueling water storage tank ***overflow level by ≤11.5 feet***. The component data for the spargers is shown in Table 6.3-4. The spargers are AP600 Equipment Class C and are designed to meet seismic Category I requirements.

The spargers perform a nonsafety-related function -- minimizing plant cleanup and recovery actions following automatic depressurization. They are designed to distribute steam into the in-containment refueling water storage tank, thereby promoting more effective steam condensation.

The first three stages of automatic depressurization system valves discharge through the spargers and are designed to pass sufficient depressurization venting flow, with an acceptable pressure drop, to support the depressurization system performance requirements. The installation of the spargers prevents undesirable and/or excessive dynamic loads on the in-containment refueling water storage tank and other structures.

Each sparger is sized to discharge at a flow rate that supports automatic depressurization system performance, which in turn, allows adequate passive core cooling system injection.

Revised ITAAC Table 2.1.2-4

Table 2.1.2-4
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.c) The RCS provides automatic depressurization during design basis events.	iii) Inspections of each fourth-stage ADS valve will be conducted to determine the flow area through each valve.	iii) The flow area through each fourth-stage ADS valve is ≥ 38 in^2.
	iv) Type tests and analysis will be performed to determine the effective flow area through each stage 1,2,3 ADS valve.	iv) A report exists and concludes that the effective flow area through each stage 1 ADS valve ≥ 4.6 in^2 and each stage 2,3 ADS valve is ≥ 21 in^2.
	v) Inspections of the elevation of the ADS stage 4 valve discharge will be conducted.	v) The minimum elevation of the bottom inside surface of the outlet of these valves is greater than plant elevation 110 feet.
	vi) Inspection of each ADS sparger will be conducted to determine the flow area through the sparger holes.	vi) The flow area through the holes in each ADS sparger is ≥ 274 in^2.
	vii) Inspection of the elevation of each ADS sparger will be conducted.	***vii) The centerline of the connection of the sparger arms to the sparger hub is ≤ 11.5 feet below the IRWST overflow level.***

Enclosure 4 to Westinghouse
Letter DCP/NRC1286

March 6, 1998



Open Item 640.56 (OITS #5238)

The following comments relate to Section 2.3.9, "Containment Hydrogen Control System," of the CDM which involve the hydrogen recombination subsystem and the hydrogen ignition subsystem. The hydrogen recombination subsystem provides hydrogen control during and the following a design basis LOCA while the hydrogen ignition subsystem provides hydrogen control during and following a degraded core or core melt scenario.

For the hydrogen recombination subsystem, the applicant has specified (a) Design Commitments, (b) Inspection, Tests, Analyses, and (c) Acceptance Criteria in Table 2.3.9-2 of the CDM document. The HRS is provided to meet the requirements of GDC 41, "Containment Control Systems in Light-Water-Cooled Power Reactors." Westinghouse references RG 1.7, "control of Combustible Gas Concentrations in Containment following A Loss-Of-Coolant Accident," in the SSAR as the methodology used for implementing these regulations. The ITAAC fail to address these requirements and Westinghouse's commitment to the methodology in RG 1.7.

a) Section 6.2.4.1.1, "Containment Mixing", of the SSAR is to provide an analysis which shows that excessive stratification of combustible gases will not occur within the containment or within a containment subcompartment. Verification of the analysis that supports the design commitment to provide a system and features to mix the combustible gases within containment has been omitted from the ITAAC.

Response:

The containment mixing commitment is incorporated in the ITAACs in a distributed form. The mixing function is driven by the passive containment cooling function. The application of water to the external surface maintains the containment shell at a cool temperature. The condensation of the steam on the interior of the containment shell creates a downward flowing layer at the wall to prevent stagnation in the dome. As the air flows downward along the wall, the wall layer also entrains surrounding mixture creating significant mixing forces. Additional details of containment mixing are discussed in subsection 6.2.4 of the SSAR. The inclusion of ITAACs for the Passive Containment Cooling System (PCS) components and operation and the configuration of the Containment System and the Nuclear Island buildings confirms the presence of systems to mix combustible gases within containment. *ITAACs are added to verify the ADS stage 4 elevation (see response to Open Item 640.164, revision 1) and that the ADS stage 4 discharge is not into the rooms that contain the CMTs.*

Table 2.1.2-4 (Cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
8.c) The RCS provides automatic depressurization during design basis events.	iii) Inspections of each fourth-stage ADS valve will be conducted to determine the flow area through each valve.	iii) The flow area through each fourth-stage ADS valve is ≥ 38 in^2.
	iv) Type tests and analysis will be performed to determine the effective flow area through each stage 1,2,3 ADS valve.	iv) A report exists and concludes that the effective flow area through each stage 1 ADS valve ≥ 4.6 in^2 and each stage 2,3 ADS valve is ≥ 21 in^2.
	v) Inspections of the elevation of the ADS stage 4 valve discharge will be conducted.	v) The minimum elevation of the bottom inside surface of the outlet of these valves is greater than plant elevation 110 feet.
	vi) Inspections of the ADS stage 4 valve discharge will be conducted.	*vi) The discharge of the ADS stage 4 valves is directed into the steam generator compartments.*

b) Design Commitment No. 3.a of Table 2.3.9-2 fails to verify conformance with design criteria, such as NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.F.1, Attachment 6, containment Hydrogen Monitor, and RG 1.97, for the containment hydrogen monitor.

Response:

Design commitments are established in Revision 3 of ITAAC 2.3.9 to provide for containment hydrogen monitoring consistent with the requirements of NUREG-0737 and Regulatory Guide 1.97 relative to equipment qualification, redundancy, power source design, quality assurance, and display. Other aspects of the requirements – such as channel availability, range, equipment identification, interfaces and servicing, testing, and calibration – are more appropriately addressed in the SSAR and Technical Specifications.

Table 2.3.9-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The seismic Category I equipment identified in Table 2.3.9-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.9-1 is located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) The seismic Category I equipment identified in Table 2.3.9-1 is located on the Nuclear Island. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
2.a) The equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on equipment located in a harsh environment.	A report exists and concludes that the equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
2.b) The Class 1E components identified in Table 2.3.9-1 are powered from their respective Class 1E division.	Testing will be performed by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.9-1 when the assigned Class 1E division is provided the test signal.
2.c) Separation is provided between VLS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See Certified Design Material, Section 3.3, Nuclear Island Buildings.	See Certified Design Material, Section 3.3, Nuclear Island Buildings.

Table 2.3.9-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
4.a) The VLS provides hydrogen monitors for indication of the containment hydrogen concentration.	Inspection for the existence of three Class 1E hydrogen monitors inside containment will be performed.	Three hydrogen monitors powered by a Class 1E power source are provided inside containment.
6. Safety-related displays identified in Table 2.3.9-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.9-1 can be retrieved in the MCR.

c) Design Commitment No. 3.b of Table 2.3.9-2 verifies the existence of a report that establishes the depletion rate for a single full-size PAR. There is no link between this acceptance criteria and the installed PARs.

Response:

The recombination rate specified in Design Commitment 4.b. of Revision 3 of the ITAACs specifies that "A report exists and concludes that the PAR depletion rate for each installed PAR is greater than or equal to 1 scfm of hydrogen at a prevailing concentration of 3 volume-percent...". The basis for this value is that under conservative design basis assumptions, the hydrogen production rate (Figure 6.2.4-3 of the SSAR) by the time containment concentration reaches a concentration of 3 percent (Figure 6.2.4-2) is less than 1 scfm.

d) Either the criteria used to locate the PARs inside containment or a description of their specific location inside containment should be provided in the ITAAC. Because the criteria used to locate the recombiners was subjective and based on engineering judgement, the staff recommends the use of a detailed description or figure to verify appropriate location of the PARs.

Response:

The specific and exact location of the passive autocatalytic recombiners (PARs) are not critical to proper operation of the hydrogen control function. Generally, it is desirable that the units be above the loop compartment (elevation 148') and several feet in from the side wall in order to be within a well mixed, nondown-draft region above the operating deck. The present design commitment verifies elevation location but does not confirm distance from the containment wall. This commitment will added to the ITAAC as indicated below.

Table 2.3.9-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
4.b) The VLS provides PAR devices for control of the containment hydrogen concentration during and following a design basis accident.	i) Inspection for the existence of two PAR devices inside containment will be performed.	i) Two PAR devices are provided inside containment within the upper compartment between elevations 150 and 175 ft and with PAR centerline greater than 10 ft from the containment shell.
	ii) Type tests, analyses, or a combination of type tests and analyses will be performed on the PARs.	ii) A report exists and concludes that the PAR depletion rate for each installed PAR is greater than or equal to 1 scfm of hydrogen at a prevailing concentration of 3 volume-percent for a test conducted at atmospheric pressure +2 psi and an ambient temperature of 120.

e) The ITAAC should verify the existence of a report that concludes that the installed PARs are qualified for a harsh environment and can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function. The report should also address the potential of the fission products that make up the post-accident radiation environment to be catalytic poisons.

Response:

Design commitment 2.a is made is to confirm that the PARs are qualified for operation during and following a harsh environment. Section 3.11 and subsection 6.2.4 of the SSAR specify environment for the equipment qualification. For the PARs, the environment includes potential catalyst poisons which could be present for the scenarios, including core damage events up through in-vessel releases as identified in these SSAR sections.

Table 2.3.9-3 Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
2.a) The equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on equipment located in a harsh environment.	A report exists and concludes that the equipment identified in Table 2.3.9-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

f) For the hydrogen ignition subsystem (HIS), the applicant has specified (a) Design commitments, (b) Inspection, Tests, Analyses, and (c) Acceptance Criteria in Table 2.3.9-2 of the CDM Document. The HIS is provided to safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal water reaction as required by 10 CFR 50.34(f)(2)(ix). The system also ensures that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent (by volume). This is accomplished by initiating a deflagration at the lower level of hydrogen flammability.

The ITAAC fail to verify several important design features of the HIS as described in Section 6.2.4.2.3 of the SSAR. The igniters have been divided into two power groups. Power to each group will be normally provided by offsite power, however should offsite power be unavailable, then each of the power groups is powered by one of the onsite non-essential diesels and finally should the diesels fail to provide power then approximately 4 hours of igniter operation is supported by the non-Class 1E batteries for each group. Assignment of igniters to each group is based on providing coverage for each compartment or area by at least one igniter from each group. The igniter assembly is designed to maintain the surface temperatures within a range of 1600 to 1700 degrees F. These design features are essential in establishing the HIS's ability to initiate a deflagration at the lower level of hydrogen flammability and should be verified by the ITAAC.

Response:

Table 2.3.9-2 of the ITAACs identifies the location (that is, compartment coverage of each igniter and the power group to which each igniter is assigned. Design commitment 3 confirms power group assignment, while design commitment 5 verifies both the igniter location and surface temperature of the igniters. Testing of the nonessential standby diesels is performed in ITAAC Section 2.6.4 "Onsite Standby Power System". The capacity of the of the non E batteries is verified in ITAAC Section 2.6.2 "Non-Class 1E dc and Uninterruptible Power Supply System."



Table 2.3.9-3 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
3. The components identified in Table 2.3.9-2 are powered from their respective non-Class 1E power group.	Testing will be performed by providing a simulated test signal in each non-Class 1E power group.	A simulated test signal exists at the equipment identified in Table 2.3.9-2 when the assigned non-Class 1E power group is provided the test signal.
5. The VLS provides the nonsafety-related function to control the containment hydrogen concentration for beyond design basis accidents.	i) Inspection for the number of igniters will be performed.	i) At least 64 hydrogen igniters are provided inside containment at the locations specified in Table 2.3.9-2.
	ii) Operability testing will be performed on the igniters.	ii) The surface temperature of the igniter exceeds 1700°F.

g) Either the criteria used to locate the igniters inside containment or a description of their specific location inside containment should be provided in the ITAAC. Because the criteria used to locate the recombiners was subjective and based on engineering judgement, the staff recommends the use of a detailed description or figure to verify appropriate location of the recombiners.

Response:

The location of the igniters is addressed in part f) above and the location of the PARs is addressed in d) above. ***The number of ignitors listed in ITAAC Table 2.3.9-3 is changed to 64.***

SSAR Revision:

None.

Open Item 640.177F (OITS #6610)

Your response to RAI 480.1079 was acceptable but you must include the following design commitments in the ITAAC for the reactor cavity:

a. No fibrous insulation shall be located within any spherical volume centered on the axis of a reactor coolant system pipe and with a radius of 12 times the inner diameter (ID) of that reactor coolant system pipe where any discharge from a break of that pipe would be intercepted by intervening components, supports and structures.

b. Where there are no intervening components, supports or structures, there shall be no fibrous insulation with in a cylindrical volume extending out 45 IDs from the break along an axis that is a continuation of the pipe axis and up to 5 IDs in the radial direction from the axis.

Response:

The response to Open Item 480.1079 includes an ITAAC that addresses the use of fibrous insulation inside containment. It specifies that no fibrous insulation may be on ASME class 1 lines or on the reactor vessel, steam generators, reactor coolant pumps or the pressurizer. This ITAAC should be sufficient for the reactor vessel cavity because it specifies that the reactor vessel insulation can not be fibrous insulation. There is no other insulation used in the reactor vessel cavity that would be affected by the suggested ITAAC.

SSAR Change:

None

ITAAC Change:

None

Open Item 640.181F (OITS #6614)

Your response to RAI 640.167F is unacceptable. The markup to Section 6.2.3.4.1 of the SSAR states that the containment isolation valves are verified to close within 20 seconds for the containment purge isolation valves and 60 seconds for all other containment isolation valves. This is inconsistent with Table 6.2.3-1 of the Revision 19 to the SSAR, which lists the containment closure times for the main steam line, main feedwater, and the containment air filter supply and exhaust as 5 seconds (refer also to SSAR chapters 9 and 15).

As stated previously, Westinghouse needs to implement a program that will verify than all tier 1 information was extracted from the AP600 SSAR (Tier 2) and that all Tier 1 information is consistent with the information in Tier 2. An example of the problem is Tier 1 design commitment 11b of Table 2.3.2-4, which is not consistent with the containment isolation valve table in the SSAR (Table 6.2.3-1). Also, it is not clear, based on item 8 of Table 2.2.1-3, where the containment isolation valves of the CNS are identified.

Response:

It was not intended that the statement in SSAR subsection 6.2.3.4.1 on containment isolation valve response times apply to the main steam or main feedwater isolation valves. This sentence has been revised to reference Table 6.2.3-1 for the response times (SSAR revision 20); Table 6.2.3-1 contains response times for all of the containment isolation valves, including a note that specifies that the response time for "standard" valves is $\leq$ 60 seconds.

The AP600 containment purge isolation valve response times were changed from 5 to 20 seconds in the response to Open Item 640.167F. This response included changes to the SSAR including the Technical Specifications and the ITAACs. However this response overlooked SSAR Table 6.2.3-1. Table 6.2.3-1 has been modified so that it shows 20 seconds for the containment purge isolation valves response time.

ITAAC Table 2.3.2-4, item 11.b (including ITAAC Table 2.3.2-1) is not inconsistent with SSAR Table 6.2.3-1 with respect to containment isolation valves. The valves shown in the ITAAC table are shown on the SSAR table. In addition the information for the containment isolation valves shown on the ITAAC table (2.3.2-1) is consistent with the SSAR table.

ITAAC figure 2.2.1-1 shows all of the containment mechanical penetrations. The penetrations included in the CNS are shown with solid lines; the penetrations included in other systems are shown with dashed lines. For example the CVS containment penetrations are shown with dashed lines in figure 2.2.1-1. The CVS ITAAC section (2.3.2) has ITAAC items in Table 2.3.2-4 which cover these lines and valves. Several items cover these lines and valves including 1, 2.a/b, 3.a/b, 4.a/b, 5, 6, 7.a/b/c, 10, 11.a/b, and 12.a/b. Item 8.a covers leak testing; it references the CNS ITAAC for leak testing of these containment isolation valves.



SSAR Change:
None

ITAAC Change:
None



Open Item (OITS #6627)

Westinghouse needs to add an ITAAC for the VFS line in the CNS for the minimum line diameter, e.g., 36 inches.

Response:

The minimum diameter of the containment purge discharge line (VFS-L204) that penetrates the containment will be added to the VFS ITAAC, 2.7.6.

SSAR Change:

None

ITAAC Change:

ITAAC Table 2.7.6-2 is revised per attachment.

Revised ITAAC Table 2.7.6-2

Table 2.7.6-2		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
2. The VFS provides the safety-related function of preserving containment integrity by isolation of the VFS lines penetrating containment.	See Certified Design Material, subsection 2.2.1, Containment System.	See Certified Design Material, subsection 2.2.1, Containment System.
3. The applicable portion of the VFS provides the intermittent flow of outdoor air to purge the containment atmosphere of airborne radioactivity during normal plant operation, and continuous flow during hot or cold plant shutdown conditions to provide an acceptable airborne radioactivity level prior to personnel access.	i) Testing will be performed to confirm that containment supply AHU fan A when operated with containment exhaust fan A provides a flow of outdoor air. ii) Testing will be performed to confirm that containment supply AHU fan B when operated with containment exhaust fan B provides a flow of outdoor air. *iii) Inspection will be conducted of the containment purge discharge line (VFS-L204) penetrating the containment.*	i) The flow rate measured at each fan is greater than or equal to 3,600 cfm. ii) The flow rate measured at each fan is greater than or equal to 3,600 cfm. *iii) The nominal line size is ≥ 36".*