NRC FORM 658
(9-1999)

U.S. NUCLEAR REGULATORY COMMISSION

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DATE OF MEETING
09/16/2003

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Docket Number(s)	Proj 689
Plant/Facility Name	
TAC Number(s) (if available)	
Reference Meeting Notice	9/3/03
Purpose of Meeting (copy from meeting notice)	To meet with Westinghouse to discuss construction scheduling. NRC held the meeting as part of a process to develop NRC ITAAC verification procedures

NAME OF PERSON WHO ISSUED MEETING NOTICE
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INCONSISNCIES NOTED WITH AP-600 DOCUMENTS

1. Tier 1 Figure 2.3.2-1, CVS, shows four penetrations, the lower three are listed in Table 2.3.2-1 as ASME but the top penetration is not listed. That penetration is also not listed in Tier 1 Table 2.2.1-1. This penetration is for spent resin flush and has a LC manual valve on the inside and outside with a relief valve inboard of the outside isolation valve. Shouldn't the penetration be listed in one of the above tables? (The same conditions exist in the AP-1000 Tier 1 text and figure.)

2. The CVS system description in Tier 2 Section 9.3.6.3 states "The [CVS] purification subsystem inside containment is defined as reactor coolant pressure boundary. This subsystem is nonsafety-related and is constructed using ... ASME Code Section III." However, the IRC portions shown on Tier 1 Figure 2.3.2-1 are designated "N". The Tier 2 P&ID (Figure 9.3.6-1) appears to show these inside lines as ??D (first 2 digits are difficult to read) where the third digit indicates that this is ANSI B31.1 piping. I checked the AP-1000 Tier 1 and Tier 2 text and figures and found the same situation; the lines are designated BBD on the Tier 2 Figure.

3. Also on Tier 1 Figure 2.3.2-1, the line connected to V084 (Aux. Spray) is designated "N" but it is connected to a line that is designated "3". P&ID 9.3.6-1 shows the line breaks as BBC indicating the lines are ASME Section III, Class C. I verified the BBC Code breaks on the AP-1000 P&ID but noted the same problem with Tier 1 Figure 2.3.2-1.)

4. Tier 1 Figure 2.3.7-1, SFP Cooling, specifies that a number of pipe segments are classified ASME Section III but there is no listing of ASME components in Table 2.3.7-1. Most of the other sections list the system valves that provide the Code break change. (The drawing in the Tier 2 document does not include line numbers or Code break flags.) Shouldn't the valves be included in the Table?

5. Tier 1 Table 2.3.7-1 lists Line L030 as the Reactor Cavity Drain line but it appears that the line should be labeled Refueling Cavity Drain Line. The same condition exists in the AP-1000 Tier 1 DCD which has two lines (L030 & L040) listed Reactor Cavity Drain lines. The piping shown on the P&ID for both plants are connected to the refueling cavity. A new Section 9.1.3.3.8 was added to the AP-1000 SFS System Description that discusses a line from the Reactor Cavity to the Steam Generator Compartment but this is apparently a different line not shown on the P&ID. A new line with a LO manual valve and two check valves as discussed in 9.1.3.3.8 is shown on AP-1000 Figure 9.1-6 connected to the refueling cavity but its opposite end appears to be blank flanged. (The Reactor Cavity Volume for these designs is somewhat different than earlier designs and is shown on Figures 19.39-5 &6.)

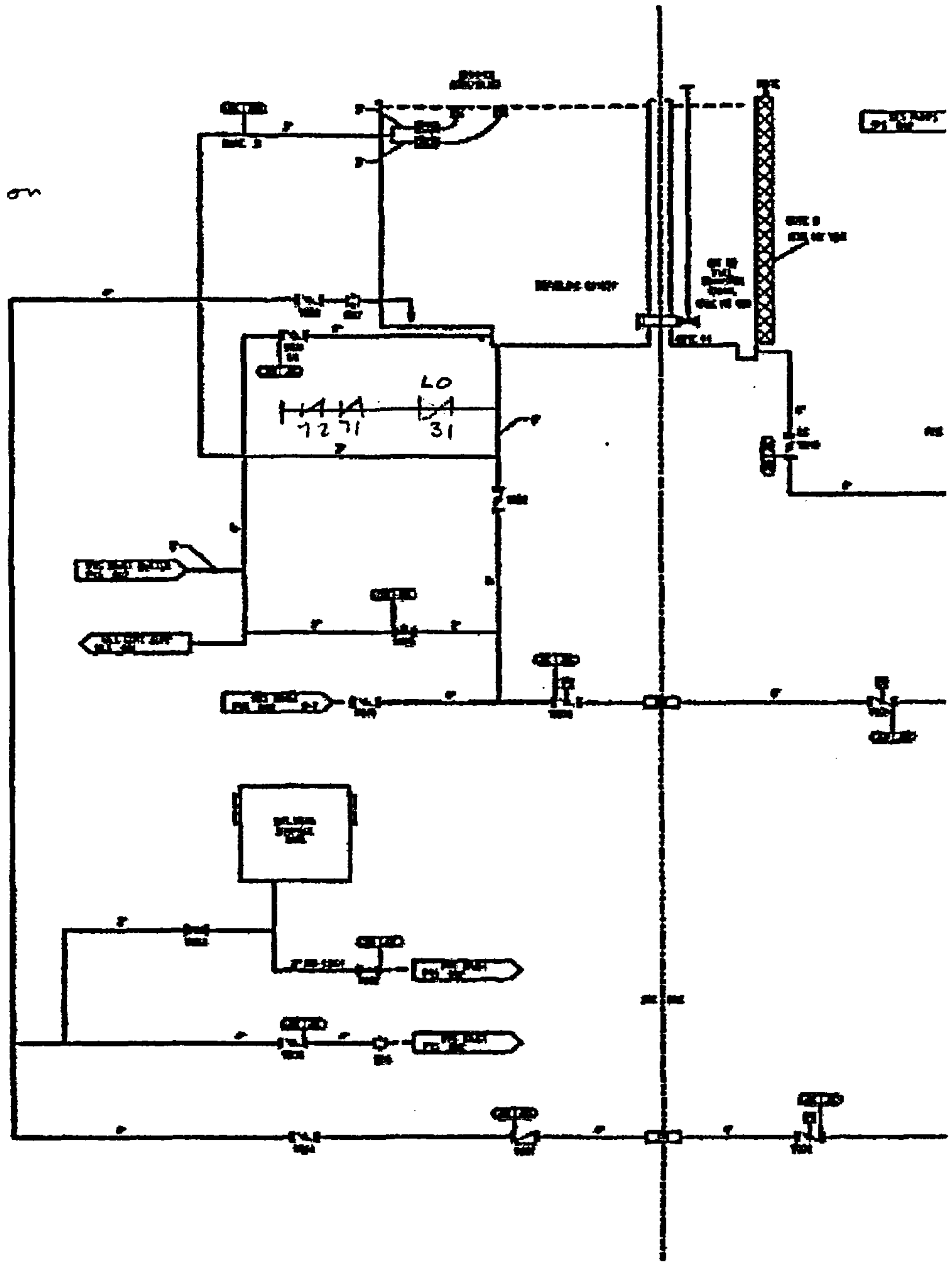
6. When reviewing Table 17.4-1 for Significant Risk SSCs, I noted that Nuclear Fuel was included under the heading for Onsite Power. This is the last entry in the Table, should it have a different heading? (The AP-1000 Table is the same.)

7. Chapter 19 includes sections for many of the Risk Significant Systems (but not all) and includes sections for the Non-Class 1E dc and UPS System (19.23) and the Compressed and Instrument Air System (19.25). These two systems are not, however, included in Table 17.4-1 that lists the Risk Significant SSCs. There are no details in the Chapter 19 Sections for any of these systems only a reference to see the system description section. The system description sections, however, do not discuss the PRA determinations. Also there is no listing in the PRA Based Insights Table 19.59-29. Should the Non-Class 1E dc and UPS System and the Compressed and Instrument Air System be included as Risk Significant; if not, why are sections designated for them in Chapter 19?

   The Non-class 1E dc system is discussed in AP-1000 Table 19-58-18, AP-1000 PRA Based Insights and the Compressed and Instrument Air System is listed in AP-1000 Table 17.4-1, Risk Significant SSCs.

SFP Drawing:

Line shown on
AP-1000
Fig 9.1-6



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 the 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. Removable steel grating is provided over the inlet to the vertical access tunnel to restrict access to the lower compartments. This grating precludes large debris from being transported into the reactor cavity during ex-vessel cooling scenarios. Figure 5.3-8 depicts the flooded compartments that provide the water for ex-vessel cooling. 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 opening has a minimum flow area of 8 ft^2 and is not susceptible to clogging from debris that can pass through the grating over the inlet to the vertical access tunnel. 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. The damper provides an acceptable pressure drop through the opening during ex-vessel cooling.

DCD subsection 6.3.2.1.3 discusses post-accident operation of the passive core cooling system, which operates to flood the reactor cavity following an accident. DCD subsection 9.1.3 discusses the connections from the refueling cavity to the steam generator compartment that facilitate flooding of the reactor cavity following an accident.

5.3.5.4 Determination of Forces on Insulation and Support System

The forces that may be expected in the reactor cavity region of the AP1000 plant during a core damage accident in which the core has relocated to the lower head and the reactor cavity is reflooded can be based on AP600 test results. The AP600 forces 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 AP1000 is used to estimate forces acting on the rigid sections of insulation.

5.3.5.5 Design Evaluation

A structural analysis of the AP1000 reactor cavity insulation system will be performed to demonstrate that it meets the functional requirements discussed above. The analysis will encompass the insulation and support system and will include a determination of the stresses in support members, bolts, insulation panels and welds, as well as deflection of support members and insulation panels.

9.1.3.3.6 Piping Requirements

Spent fuel pool cooling system piping is made of austenitic stainless steel. Piping joints and connections are welded, except where flanged connections are required as indicated on the spent fuel pool cooling system piping and instrumentation diagram (Figure 9.1-6).

9.1.3.3.7 Reactor Cavity Seal Ring

The AP1000 reactor cavity seal ring is part of the fuel handling system and is a permanent welded seal ring used to provide the seal between the vessel flange and the refueling cavity floor. The reactor cavity seal ring does not use pneumatic seals and is not subject to a gross failure due to loss of a seal.

Leakage is not expected with this design. Leakage past or through the seal would not significantly affect the water level in the refueling canal and would be detected as an increase in water level in the containment sump. Water level in the sump is a key parameter in reactor coolant leak detection.

9.1.3.3.8 Reactor Cavity Connections

The spent fuel pool cooling system contains connections to the refueling cavity to prevent excessive holdup of water in the reactor cavity following an accident. The piping connection facilitates draining of the reactor cavity to the steam generator compartment following a postulated accident. The line connects at the bottom of the reactor cavity and discharges to a steam generator compartment, and contains a manual locked-open isolation valve and two check valves in series. The isolation valve is closed during refueling operations to facilitate flooding of the reactor cavity for refueling operations. Other connections are provided to the refueling cavity to facilitate proper draining, filling, and purification of the reactor cavity to support refueling operations.

9.1.3.4 System Operation and Performance

The operation of the spent fuel pool cooling system for the pertinent phases of plant operation are described in the following paragraphs.

9.1.3.4.1 Normal Operation

During normal plant operation, one spent fuel pool cooling system mechanical train of equipment is operating. The operating train is aligned to provide spent fuel pool cooling and purification. The other train is available to perform the other functions of the spent fuel pool cooling system such as water transfers or in-containment refueling water storage tank purification.

9.1.3.4.1.1 Ion Exchange Media Replacement

The initial and subsequent fill of ion exchange media is made through a resin fill nozzle on the top of the ion exchange vessel. When the media is ready to be transferred to the solid radwaste system, the vessel is isolated from the process flow. The flush water line is opened to the sluice piping and demineralized water is pumped into the vessel through the normal process outlet