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Energy Systems

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DCP/NRC1245
NSD-NRC-98-5556
Docket No.: 52-003

February 4, 1998

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

ATTENTION: T. R. QUAY

SUBJECT: AP600 RESPONSE TO FSER OPEN ITEMS

Dear Mr. Quay:

Enclosed with this letter are 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 to be designated in the Westinghouse status column of OITS.

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 if you have any questions concerning this transmittal.

Susan V Fanto for
Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Enclosure

cc: W. C. Huffman, NRC (Enclosure)
T. J. Kenyon, NRC (Enclosure)
J. M. Sebrosky, NRC (Enclosure)
D. C. Scaletti, NRC (Enclosure)
N. J. Liparulo, Westinghouse (w/o Enclosure)

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Table 1
List of FSER Open Items Included in Letter DCP/NRC1245

FSER Open Item	OITS Number	Westinghouse status in OITS
480.1117F	6541	Confirm W
480.1133F	6557	Confirm W
480.1134F	6558	Confirm W
480.1159F	6538	Action N
720.425F (R1)	6140	Action N
720.456F (R1)	6371	Action N
720.426F (R1)	6141	Action N

Enclosure to Westinghouse
Letter DCP/NRC1245

February 5, 1998

Question: 480.1117F (OITS #6541)

Table 3.3.2-1, Function 19, "Containment Air Filtration System Isolation" receives an isolation signal based on Containment Radioactivity High 1. This function should be operable in modes 1-4 (similar to containment isolation and containment cooling) and during core alterations and movement of irradiated fuel, consistent with Westinghouse standard technical specifications. In addition, Westinghouse should explain why the reference trip setpoint is set at 2 R/hour as compared to the standard technical specifications reference value of 2 times background?

Response:

Consistent with the assumed protection for large break LOCAs, the Applicability (MODES 1, 2, and 3) of ESFAS Function 19.a will be expanded to include:

MODE 4 with the RCS not being cooled by RNS

This change in Applicability makes Condition Q (which ends in MODE 4) inadequate, since it will no longer take the plant out of the specified Applicability. A new Condition which ends in "MODE 4 with the RCS cooling provided by the RNS," be added as Condition Z.

Since for AP600 site boundary dose limits are met, for events occurring during core alterations or movement of irradiated fuel, without taking credit for containment air filtration system isolation, Function 19 is not required during these activities.

The "[≤ 2 R/hour]" specified in Technical Specification Table 3.3.2-1 is the safety analysis value and has been included for reviewer information, only. The Reviewer's Note on the first page of the table explains that the values listed are the safety analysis values and that the actual setpoints will be established in accordance with approved methodologies discussed in WCAP-14606 following selection of plant-specific instrumentation. Therefore, there is no basis for comparison of the AP600 safety analysis value with a typical plant trip setpoint.

SSAR Revision: See attached mark-ups, pages 3.3-26, 3.3- 38, B 3.3-96, B 3.3-97, and B 3.3-115.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Y. Required Action and associated Completion Time not met.	Y.1	If in MODE 5 initiate action to be in MODE 5 with the RCS intact and ~~visible level in pressurizer~~. > 20% PRESSURIZER LEVEL.	Immediately RAI 480.1127F
	AND		
	Y.2	If in MODE 6 with upper internals in place, ~~and cavity level less than full~~, initiate action to be in MODE 6 with the upper internals removed, ~~and the cavity full~~.	Immediately RAI 480.1125
	AND		
	Y.3	Suspend positive reactivity additions.	Immediately

INSERT CONDITION Z

RAI 480.1117F

INSERT Page 3.3-26

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
Z. Required Action and associated Completion Time not met.	Z.1	----------NOTE---------- Flow path(s) may be unisolated intermittently under administrative control. ----------	
		Isolate the affected flow path(s) by use of at least one closed manual or closed and deactivated automatic valve.	6 hours
	OR		
	Z.2.1	Be in MODE 3.	12 hours
	AND		
	Z.2.2	Be in MODE 4 with the RCS cooling provided by the RNS.	30 hours

480.1117F-3

Table 3.3.2-1 (page 10 of 12)
Engineered Safeguards Actuation System Instrumentation

	FUNCTION	APPLICABLE MODES OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
17.	Normal Residual Heat Removal System Isolation						
	a. Containment Radioactivity - High 2	1,2,3(m)	4	B,Q	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6		[≤ 100 R/hr]
	b. Safeguards Actuation	1,2,3(m)	Refer to Function 1 (Safeguards Actuation) for all initiating functions and requirements.				
18.	ESFAS Interlocks						
	a. Reactor Trip, P-4	1,2,3	3 divisions	D,M	SR 3.3.2.3		N/A
	b. Pressurizer Pressure, P-11	1,2,3	4	J,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6		[≤ 1970 psig]
	c. Intermediate Range Neutron Flux, P-6	2	4	J,L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6		[≥ 1E-10 amps]
	d. Pressurizer Level, P-12	1,2,3	4	J,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6		[Above Pressurizer Water Level - Low 1 setpoint of 20%]
	e. RCS Pressure, P-19	1,2,3,4(j)	4	J,M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6		[≥700 psig]
RAI 480.1117F							
19.	Containment Air Filtration System Isolation						
	a. Containment Radioactivity - High 1	1,2,3 4(j)	4	B,~~Q~~ R	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6		[≤ 2 R/hr]
	b. Containment Isolation	Refer to Function 3 (Containment Isolation) for initiating functions and requirements.					

(continued)

(j) With the RCS not being cooled by the Normal Residual Heat Removal System (RNS).

(m) Not applicable for valve isolation functions whose associated flow path is isolated.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

18.d. Pressurizer Level, P-12

The P-12 interlock is provided to permit midloop operation without core makeup tank actuation, reactor coolant pump trip, or purification line isolation. With pressurizer level channels less than the P-12 setpoint, the operator can manually block low pressurizer level signal used for these actuations. When the pressurizer level is above the P-12 setpoint, the pressurizer level signal is automatically enabled and a confirmatory open signal is issued to the isolation valves on the CMT cold leg balance lines. This Function is required to be OPERABLE in MODES 1, 2, and 3.

18.e. RCS Pressure, P-19

The P-19 interlock is provided to permit water solid conditions (i.e., when the pressurizer water level is [>92%]) in lower MODES without automatic isolation of the CVS makeup pumps. With RCS pressure below the P-19 setpoint, the operator can manually block CVS isolation on High 2 pressurizer water level. When RCS pressure is above the P-19 setpoint, this Function is automatically unblocked. This Function is required to be OPERABLE IN MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS. When the RNS is cooled by the RNS, the RNS suction relief valve provides the required overpressure protection (LCO 3.4.15)

19. Containment Air Filtration System Isolation

Some DBAs such as a LOCA may release radioactivity into the containment where the potential would exist for the radioactivity to be released to the atmosphere and exceed the acceptable site dose limits. Isolation of the Containment Air Filtration System provides protection to prevent radioactivity inside containment from being released to the atmosphere.

19.a. Containment Radioactivity - High 1

4 WITH THE RCS NOT BEING COOLED BY THE RNS

Three channels of Containment Radioactivity - High 1 are required to be OPERABLE in MODES 1, 2, 3, and 4 when the potential exists for a LOCA, to protect against radioactivity inside containment

RAI 480.1117F

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

19.a. Containment Radioactivity - High 1 (continued)

being released to the atmosphere. These Functions are not required to be OPERABLE in MODES ~~4,~~ 5, and 6 because any DBA release of radioactivity into the containment in these MODES would not require containment isolation.

MODE 4 WITH THE RCS BEING COOLED BY THE RNS OR

RAI 480.1117F

19.b. Containment Isolation

Containment Air Filtration System Isolation is also initiated by all Functions that initiate Containment Isolation. The Containment Air Filtration System Isolation requirements for these Functions are the same as the requirements for the Containment Isolation. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 3, Containment Isolation, is referenced for initiating Functions and requirements.

20. Main Control Room Isolation and Air Supply Initiation

Isolation of the main control room and initiation of the air supply provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. This Function is required to be OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel because of the potential for a fission product release following a fuel handling accident, or other DBA.

20.a. Control Room Air Supply Radiation - High 2

Two radiation monitors are provided on the main control room air intake. If either monitor exceeds the High 2 setpoint, control room isolation is actuated.

20.b. Battery Charger Input Voltage - Low

Low input voltage to the 1E dc battery chargers will actuate main control room isolation and air supply initiation. This was previously described as Function 15.c.

(continued)

BASES

ACTIONS

X.1, X.2, and X.3 (continued)

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met while in MODES 5 and 6, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by immediately initiating action to be in MODE 5 with the RCS open and ~~a visible~~ level ~~in the pressurizer~~ or to be in MODE 6 with the upper internals ~~in~~ REMOVED. ~~place and the reactor cavity level less than full~~. These requirements minimize the consequences of the loss of decay heat removal by maximizing RCS inventory and maintaining RCS temperature as low as practical. Additionally, the potential for a criticality event is minimized by suspension of positive reactivity additions.

RAI 480.1122F

GREATER THAN 20% PRESSURIZER

RAI 480.1123

REPLACE WITH INSERT Y.1.1, DCP/NRC1119 11/5/97

Y.1, Y.2, and Y.3

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met while in MODES 5 and 6, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. This is accomplished by immediately initiating action to be in MODE 5 with the RCS intact and ~~a visible~~ level ~~in the pressurizer~~ or to be in MODE 6 with the upper internals in place and the reactor cavity level less than full. These requirements minimize the consequences of the loss of decay heat removal by maximizing RCS inventory and maintaining RCS temperature as low as practical. Additionally, the potential for a criticality event is minimized by suspension of positive reactivity additions.

RAI 480.1117F

INSERT Z

SURVEILLANCE REQUIREMENTS

The Surveillance Requirements for each ESF Function are identified by the Surveillance Requirements column of Table 3.3.2-1. A Note has been added to the Surveillance Requirement table to clarify that Table 3.3.2-1 determines which Surveillance Requirements apply to which ESF Functions.

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on

(continued)

Revised Bases - Insert on page B 3.3-115

Y.1.1, Y.1.2, Y.2, and Y.3

If the Required Action and the associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met while in MODE 4, with RCS cooling provided by the RNS, MODE 5, or MODE 6, the plant must be placed in a MODE in which the likelihood and consequences of an event are minimized. If in MODE 4, this is accomplished by placing the plant in MODE 5 within 12 hours. The 12 hours is a reasonable time to reach MODE 5 from MODE 4 with RCS cooling provided by the RNS (approximately 350°F) in an orderly manner without challenging plant systems.

RAI 480.1122F

If in MODE 4 or 5, Required Action Y.1.2 requires initiation of action within 12 hours to open the RCS and establish >20% ~~a visible~~ level in the pressurizer. The 12 hour Completion Time allows transition to MODE 5 in accordance with Y.1.1, if needed, prior to initiating action to open the RCS pressure boundary.

RAI 480.1123F

If in MODE 6, Required Action Y.2 requires the plant to be maintained in MODE 6 and initiation of action to remove the upper internals. ~~and to fill the cavity.~~

Required Actions Y.1 and Y.2 minimize the consequences of the loss of decay heat removal by maximizing RCS inventory and maintaining RCS temperature as low as practical. Opening the RCS pressure boundary assures that cooling water can be injected without ADS operation. Additionally, the potential for a criticality event is minimized by suspension of positive reactivity additions.

INSERT Page B 3.3-115

Z.1, Z.2.1 and Z.2.2

If the Required Action and associated Completion Time of the first Condition listed in Table 3.3.2-1 is not met, the plant must be placed in a condition where the instrumentation Function for valve isolation is no longer needed. This is accomplished by isolating the affected flow path by the use of at least one closed manual or closed and deactivated automatic valve within 6 hours.

If the flow path is not isolated within 6 hours, the plant must be placed in a MODE in which the LCO does not apply. This is accomplished by placing the plant in MODE 3 within 12 hours and in MODE 4 with RCS cooling provided by the RNS within 30 hours.

This Action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the flow path can be rapidly isolated when a need for flow path isolation is indicated.

Question: 480.1133F (OITS #6557)

The discussion of the Containment TS 3.6.1 applicability in the BASES states that "Except in Mode 5 with the loops not full, the time to boiling and core uncovery is significantly reduced due to reduced Reactor Coolant System inventory". This statement does not make sense and should be corrected or explained.

Response:

As agreed at the NRC / Westinghouse meeting of January 28, 1998, the statement will be removed from the bases.

SSAR Revision: See attached mark-up.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Satisfactory leakage rate test results is a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program Leakage Test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. ~~Except in MODE 5 with the loops not full, the time to boiling and core uncovery is significantly reduced due to reduced Reactor Coolant System inventory.~~ The MODES 5 and 6 requirements are specified in LCO 3.6.8, "Containment Penetrations".

480.1133F

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining

(continued)

Question: 480.1134F (OITS #6558)

The Applicability discussion in the BASES for TS 3.6.3 states that , in addition to the pressure and temperature limitations of these modes, there is a large inventory of coolant during Modes 5 and 6 and, therefore, the containment isolation valve specifications are not required during these Modes. The inventory of coolant is no greater during Modes 5 and 6 than during operational Modes 1-4 and is, in some instances, less during reduced inventory and mid-loop operations. This statement does not appear to be accurate and the BASES should be corrected to support the applicability discussions.

Response:

As agreed at the NRC/Westinghouse meeting of January 28, 1998, the bases will be corrected to adopt standard TS wording, i.e., "and large inventory of coolant" will be removed.

SSAR Revision: See attached mark-ups.

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4 a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES, ~~and large inventory of coolant~~. Therefore, containment isolation valves are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. However, containment closure capability is required in MODES 5 and 6. The requirements for containment isolation valves during MODES 5 and 6 are addressed in LCO 3.6.8, "Containment Penetrations."

480.1134F

ACTIONS

The Actions are modified by a Note allowing containment penetration flow paths to be unisolated intermittently under administrative control. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event that the containment isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

(continued)

480.1134F-1

Question: 480.1159F (OITS #6583)

A technical specification equivalent to standard technical specification STS 3.9.4 specifying the status of containment penetration requirements during core alterations and movement of irradiated fuel assemblies within containment should be provided (similar to current AP600 TS 3.6.8 with applicability during core alterations).

Response:

Without taking credit for containment closure, the AP600 site boundary dose limits are met, for events occurring during core alterations or movement of irradiated fuel Therefore, STS LCO 3.9.4 is not applicable to AP600. This difference is based on improvements in analysis methodology.

SSAR Revision: None.

Question: 720.425F (OITS # 6140) Revision 1

FSER OI Pertaining to In-Vessel Steam Explosion:

Bounding approach to melt release rates; consideration of higher release rates and sensitivity studies.
(See copy of enclosure to NRC transmittal letter dated November 4, 1997 for the staff's clarification to this FSER open item).

Response:

The release rates considered in the ROAAM evaluation include, with margin, the rate of approximately 160 kg/s estimated to have occurred in TMI-2. This rather gradual side-pour occurred over a time period of approximately 90 seconds, and gradually burned a hole to the baffle plate having the approximate final dimensions of 60 cm x 150 cm. Such a development is also consistent with the ROAAM evaluation of the AP600. In particular, three points can be implicitly made:

1. The rate found in TMI-2 shows that the initial melt-through area (ROAAM study's intangible) is of limited size. Specifically, since melt velocities would be about the same, the ROAAM study's largest pour area is more than twice the starting pour area at TMI-2.

2. The growth of the pour area was gradual, as expected, according to the mechanisms discussed in the DOE/ID-10541 report. Moreover, the pour area remained coherent as opposed to burning through in multiple locations.

3. The design differences between the AP600 and TMI-2, specifically the presence of a thick second barrier in the AP600 (the reflector), has the effect of promoting the above two items in the direction of even more gradual melt release than occurred in TMI-2.

NRC Follow-on Question:

During the open item closure meeting on January 22, 1998, the staff requested additional information be supplied to the response of this FSER open item. Specifically, the staff's comments were as follows: the discussion of the initial reflector melt-through area does not include a justification to support the statement that the TMI-2 melt-through area was smaller than that used in the ROAAM. The statement that the growth of the hole was gradual over the duration of the pour does not appear to be consistant with the final hole size. Please support the statement that the ROAAM initial melt-through area is conservative with respect to the TMI-2 baffle failure size.

Response to Follow-on Question:

In assessing the relation of the IVE relocation scenario to the experience in TMI-2, the following relevant points can be made:

a) The failure mechanism of the ultimately retaining steel boundary is completely different. In the case of AP600, this boundary is provided by the reflector, a 15-cm-thick stainless steel wall, and the failure occurs by melt-through of the contained, and naturally-convecting melt. In the case of TMI-2, this boundary was the baffle plate, a 1.9-cm-thick stainless steel plate, and failure occurred by impingement of a molten jet released from the melt-containing crucible, some distance away from the baffle plate.

b) It is generally understood that relocation of approximately 30 tons of melt occurred over a time period of approximately 60 seconds. For a 10-cm diameter melt jet, forming by the melt-through of an equivalent area of the crucible boundary (crust), this would produce a velocity of approximately 7 m/s. Such an exit velocity would require a gravity head of approximately 2.5 m which is the distance of failure location from the top of the core (see Figure 720.425F-1). Furthermore, Anderson and Sienicki (Nuclear Technology, Vol. 87, p. 283, August 1989) found that such a jet (with a 200K superheat) would erode through the baffle plate in approximately 20 seconds, after the impingement of 8.8 tons. This is also consistent with the relocated masses found in TMI-2.

c) It is much more appropriate to compare the 10-cm melt-through dimension than the final approximately 50x150 cm^2 hole size in the TMI-2 crucible with the ROAAM IVE analysis of 10 x 20 or 10 x 40 cm^2 size melt-through holes in the reflector. Both are at near the top of the respective melt pools. The ROAAM IVE analysis is considerably bigger but there is no significant melt static head above it, while in TMI-2 there was. In the TMI-2 case, the relocation rate was estimated as approximately 500 kg/s. The ROAAM IVE analysis has taken two scenarios, with rates of 200 and 400 kg/s. Based on the above and considerations of failure size and static head above it, the choices are clearly very conservative in relation to TMI-2.

d) It is also interesting to examine in more detail what the final hole on the TMI-2 baffle was really found to be. It is widely mentioned, as in the NRC question itself, that this hole is ~50 x 150 cm in horizontal and vertical dimensions respectively. Actually, as seen in Figure 720.425F-1, this is highly misleading. For more than half of the vertical distance, the hole is actually less than 20 cm. Also, the hole is narrower near the top, a little more than 20 cm, and it grows in lateral dimension for a distance of ~50 cm. Then it abruptly decreases to less than 20 cm. Finally, the baffle plates in Figure 720.425F-1 are shown "unfolded", while the true azimuthal size of the hole can be seen with the help of Figure 720.425F-2.

In conclusion, this response provides further substance, based on the TMI-2 experience, that the quantification is conservative.

SSAR/PRA Revisions: None.

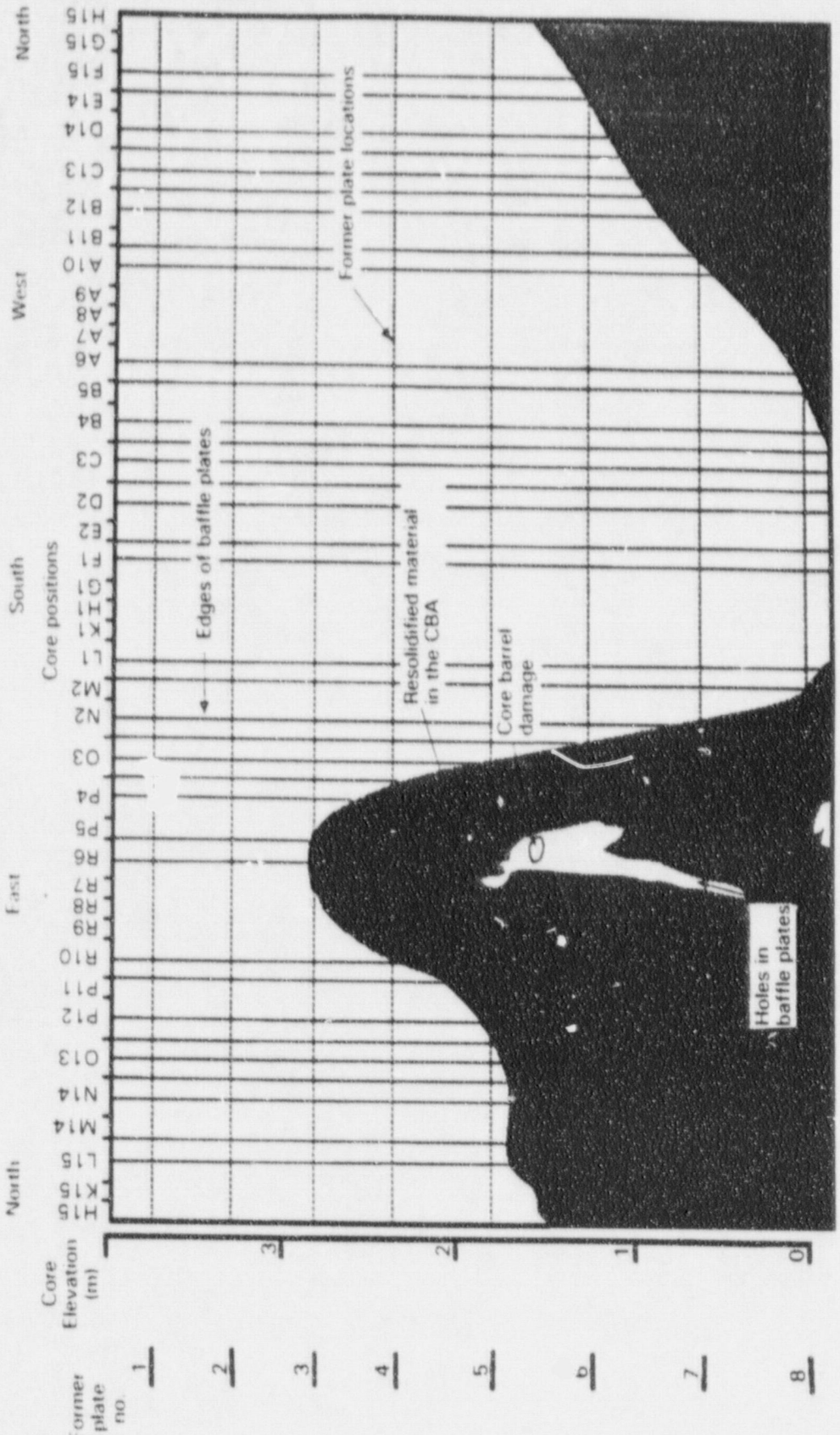

Fig. 3. View of the baffle plates surrounding the core, showing damage patterns.

Figure 720.425F-1

720.425F(R1)-3

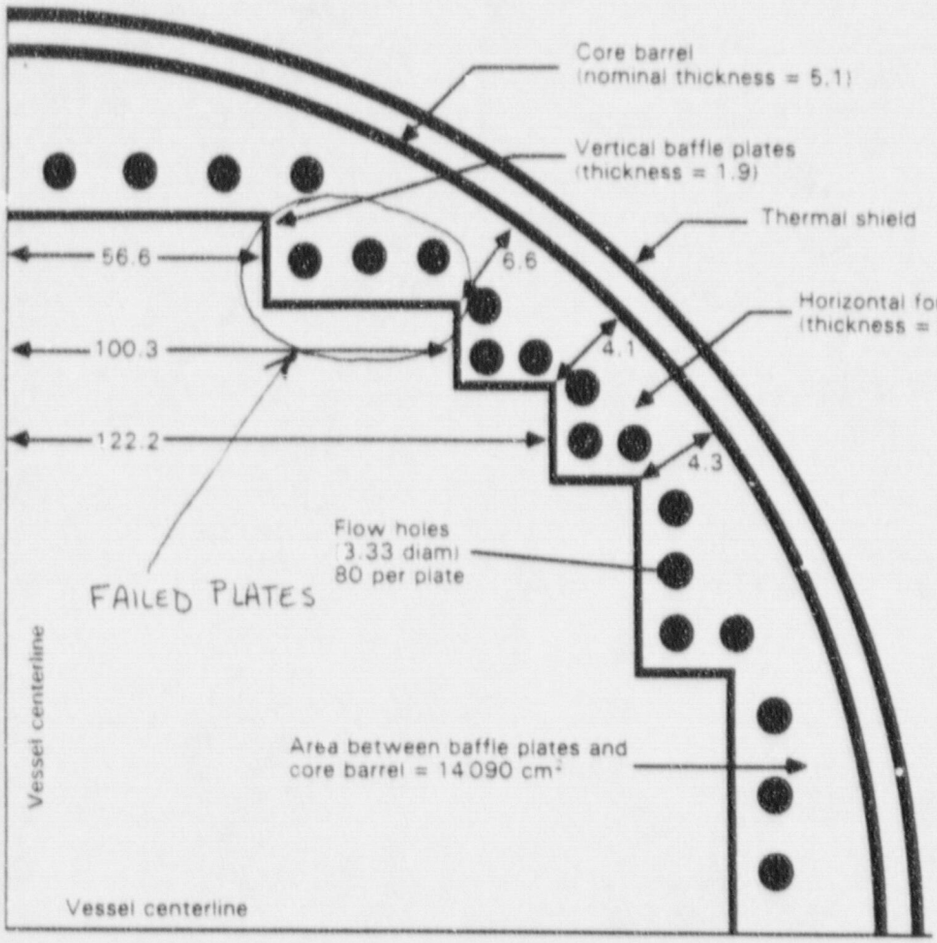

Figure 720.425F-2

Fig. 2. Schematic of the CBA, showing the baffle and former plates. All dimensions are in centimetres.

few millimetres thick on the southwest side. There appeared to be a resolidified crust on the surfaces of the three bottom core former plates and on sections of the core barrel; this crust varied in thickness from ~0.5 to 5 cm. The molten material moved down into the lower CSA through the flow holes in the core former plates. It is estimated that 4.2 tonnes of core debris were retained in this region.[2] The 2.5-cm annulus between the core barrel and the thermal shield was visually inspected, and only fine particulate was observed. No major damage to these components was seen.

The flow holes in the baffle plates just below former plate 6 were observed to have minor ablation damage adjacent to the melted hole in the baffle plate. Advancing around the core, farther from the initial crust failure point, the observed conditions go from slightly ablated holes to clean holes to holes with an outer solidified crust resembling a lava tube to holes completely plugged with solidified corium on the opposite side of the core. This indicates that very nearly the entire volume between former plates 6 and 7 was full of molten corium (~7.8 tonnes) that flowed from the initial melt-through of the baffle plate around the core, cooling as it flowed. The majority of the molten corium then flowed downward through the flow holes in former plate 7 and ultimately into the lower head of the vessel. In the observed areas of this region, the remaining material was in the form of crusts that had solidified on both the horizontal and vertical structural surfaces. Hydraulic analysis of the flow through the 80 flow holes in the former plates indicates an initial draining rate of ~1.8 tonne/s, requiring <8 s to drain the volume. Obviously, the molten corium must have entered this region at a rapid rate (>1.8 tonne/s) to have filled the region between former plates 6 and 7.[8] As the flow from the molten pool in the core through

[8] Another possibility is that the flow holes in the former plates plugged with freezing corium, which subsequently remelted allowing the volume to drain. However, once the corium freezes, there is only the remaining superheat in the molten corium to melt the solidified corium. Analysis shows that the stainless steel would melt first, which is inconsistent with the observed end-state conditions.

720.425F(R1)-4

Question: 720.426F (OITS # 6141) Revision 1

FSER OI Pertaining to In-Vessel Steam Explosion:

Splinter scenario involving downward melt relocation; demonstration as to why the scenario is "physcially unreasonable" or consideration of the scenario within the ROAAM framework.
(See copy of enclosure to NRC transmittal letter dated November 4, 1997 for the staff's clarification to this FSER open item).

Response:

A teleconference was held on November 19, 1997 between NRC (S. Basu, M. Snodderly, J. Sebrosky), DOE (C. Thompson, T. Theofanous - ARSAP), and Westinghouse (J. Scobel, C. Haag) to discuss the NRC's November 4, 1997 letter providing this FSER open item. As was discussed during the telecon, Westinghouse submitted further information on the downward melt relocation in Volume 2 of DOE report titled "Addenda to DOE/ID-10541, -10503, -10504." The DOE report was transmitted to the NRC via Westinghouse letter DCP/NRC1133, dated November 12, 1997. The expanded discussion on downward melt relocation is provided in section "Addendum to Chapter 4" of the DOE report.

NRC Follow-on Question:

During the open item closure meeting on January 22, 1998, the staff requested additional information be supplied to the response of this FSER open item. Specifically, the staff noted the Addendum to Chapter 4 of DOE/ID-10541 does not include a discussion of a potential downward debris relocation to the lower head induced by a primary sideward relocation initiated steam explosion in the lower head which fails the lower core blockage but does not fail the reactor vessel. Please address this postulated downward debris relocation mechanism.

Response to Follow-on Question:

The response to this question is based on the following key factors:

a) The reactor vessel lower internals assembly, which includes the core barrel, reflector, and core support plate, are, at the time of interest, still integral and structurally strong. These constitute the outer envelope of the crucible that contains the melt. Only the uppermost area has melted, but we are interested in the lower part. Also, the lower support structure is integral, and structurally strong.

b) The downcomer cross sectional area is nearly 4 m^2, and allows relatively free venting up and through the cold legs. This would prevent pressurization during premixing. Also in the event of any significant interaction, with sustained pressures capable to set the lower boundary of the crucible (the crusts), or the crucible as a whole, in motion, this vent area would allow large quantities of lower plenary water to be dispersed, together with venting steam, upwards. Note, in this respect, that only a fraction (~30%) of the core support plate area is open (the flow holes), and also, the inertia mass of the whole lower internals assembly (containing the melt), is at least one order of magnitude greater than any lower plenum water mass coupled in the interaction. This means any pressure developed in between these two masses would tend to expel the water rather than move the core.

c) To fail the lower boundary of the crucible (the crusts), pressure must be applied from below that is high enough and sustained enough to cause motion. This can only be done by forcing water on to this boundary, and this can arise only from a sustained strong interaction in the lower plenum. But an immediate consequence of this is, also that another melt-water interaction boundary is fo[illegible]d, at the failing lower boundary of the crucible. This would tend to be self-limiting, as the developing pres[illegible] [illegible]eates a local expansion zone, that again vents downwards, expelling lower plenum water, in a manner that precedes the downward relocation of the melt that would eventually occur. Note that this interaction zone would also contain melt, which would be expelled downwards as well, sustaining the removal of lower plenum water.

d) Throughout all these complicated interactions the structures mentioned under (a) would effectively maintain the retentive property of the crucible, while the core support plate and the internal support structures would effectively prevent a fall-back, gross, contact mechanism. Rather, the fallback would be arrested, and any melt relocation has to occur by gravity, through the holes on the core support plate.

e) By that time hardly any water would have been left in the lower plenum to receive the melt for an explosive interaction. Again, no mechanism that would violate lower head integrity is seen.

PRA Revision: None.

Question: 720.456F (OITS #6371) Revision 1

RAI 720.407 refers to the issue of breach size for the localized failure case, used by Westinghouse in the TEXAS calculations. The localized failure occurs at the transition between the hemispherical lower head and the cylindrical portion of the vessel, presumably due to thermal attack of the vessel wall. It is not clear if the breach size was chosen arbitrarily or was based on calculations of thermal load generated by the molten pool. The basis for assuming a 0.06 m diameter breach for the localized failure case is considered an open item pending a response from Westinghouse, it is noted that the overall conclusion for the localized failure case, i.e., that the containment integrity would not be challenged, is not expected to change as a result of the resolution of this RAI.

Response:

~~As stated in the response to RAI 720.407, the 0.06 meter diameter size of the breach for the localized reactor vessel failure case was arbitrarily chosen. It is not based on any thermal-structural considerations.~~

Appendix B of the AP600 PRA contains a set of deterministic analyses of ex-vessel phenomena to show the consequences of reactor vessel failure on the integrity of the containment structure. The main body of the AP600 PRA provides evidence to show that if the reactor vessel bottom head remains submerged and the reactor coolant system is depressurized, failure of the reactor vessel and transfer of the core material to the reactor cavity is physically unreasonable. In addition, the analyses in the AP600 PRA assume that if the bottom head of the reactor vessel is not submerged or the reactor coolant system is not depressurized, the reactor vessel fails and one of the unspecified ex-vessel phenomena leads to containment failure Using this approach, the risk associated with severe accidents for the AP600 design are shown to be acceptable. Thus, there is no need to provide exhaustive analyses of ex-vessel severe accident phenomena and their uncertainties, to support the AP600 risk assessment quantitative results. The limited set of analyses to show the effects of the various ex-vessel phenomena for the AP600 design were included in the PRA (in Appendix B) in response to an NRC concern that there may be no margin to early containment failures in the event of reactor vessel failure.

The analyses in Appendix B of the PRA include individual evaluations of the effects of ex-vessel steam explosions, direct containment heating and core concrete interactions on the integrity of the containment fission product boundary. At the inception of the evaluation, it was decided that all analyses should have a common base in terms of the mode of reactor vessel failure. In defining the reactor vessel failure mode, it was recognized that the failure mode had a significant degree of uncertainty and that the mode of failure could impact the subsequent analyses of the impact of ex-vessel phenomena on containment integrity. Thus, a large and small failure of the reactor vessel mode were assumed as initial conditions to study each of the ex-vessel phenomena. This approach was chosen in order to show a representative ex-vessel phenomena impact from two distinctly different vessel failure modes that might be postulated. The details of each of the assumed reactor vessel failure modes is provided in Reference B-6 of Appendix B of the AP600 PRA. For the localized small failure case, a value for the initial failure size of 6 cm was taken as a typical failure size that would drain the metallic portion of the in-vessel core debris to the reactor cavity before the oxide portion. No structural or thermodynamic analyses were performed to support the choice of an initial 6 cm failure size; it was arbitrarily chosen to represent a small failure that would drain the metallic portion of the in-vessel molten core debris pool (see the response to RAI 720.457F for the details of the assumed configuration of the in-vessel molten core pool). In keeping with the intent of this limited set of deterministic ex-vessel phenomena analyses (as summarized above), it was not necessary to identify bounding, optimal or best

estimate models and assumptions. The only requirement was that the models and assumptions be mechanistically-based and reasonable. The choice of a 6 cm hole size for the initial failure in the reactor vessel is mechanistically based in that is permits the draining of the metallic portion of the in-vessel molten core debris pool before the oxidic portion; the 6 cm hole size is not unreasonable, based on engineering judgment.

PRA Revisions: None.