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DCP/NRC1249
NSD-NRC-98-5562
Docket No.: 52-003

February 9, 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.

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/NRC1249

FSER Open Item	OITS Number	Westinghouse status in OITS
220.130F (R2)	6435	Confirm W
440.784F (R1)	6366	Confirm W
440.785F (R1)	6367	Confirm W

Enclosure to Westinghouse
Letter DCP/NRC1249

February 9, 1998

Open Item 220.130F (OITS #6435) Response Revision 2

Based on the staff's past licensing review experience, the unevenly distributed construction loads on the foundation mat, especially for the foundation mat with large dimensions and irregular shape, can be very significant and may cause severe foundation cracks. Westinghouse should provide the basis for demonstrating the design adequacy in coping with the unevenly distributed construction loads.

In the meeting on August 4 through 8, 1997, Westinghouse presented its design approach and results for considering the effects of construction settlements in the design of the foundation mat. During this meeting, the effects of construction settlements on developed moments and shears in the NI structures were reviewed and discussed. As a result of the discussion regarding the staff's review findings, the following design procedures for considering loads due to construction sequence and settlements in the foundation mat design are acceptable to the staff:

(1) Obtain stress resultants at critical locations on the NI from the following load cases: (a) Case 1 of PCRA analysis for the deep clay site, (b) Case 2 of PCRA analysis for the sand/clay site, (c) the analyses based on the simplified INITEC's Winkler spring models.

(2) Obtain the maximum values of stress resultants at each of the critical location from the cases in Item 1 above. This list of maximum stress resultant values is then to be considered as the resultant dead load stress resultant solution.

(3) The stress resultants due to dead load from Item 2 above are then to be combined with all other stress resultants to obtain stress resultants in satisfying the various load combination requirements. For each load case, the list of maximum stress resultant values represents the elastic demand on the NI.

(4) The elastic demand is then to be compared with the section capacities of the concrete structural elements of the NI to judge the adequacy of the design.

(5) Because soil bearing pressures calculated from the PCRA and INITEC analyses are not sensitive to the size of finite elements used in the model, they can be used in the localized analysis with more finely discretized finite elements to calculate final design moments and shears.

Westinghouse should follow this procedure to design the NI structures for resisting the loads induced by construction sequence and settlements. In addition, Westinghouse should document this procedure in the SAR. On the basis discussed above, Open Item 3.8.5-10 remains unresolved.

Response: Revision 2

This open item was addressed in Westinghouse's letter of October 17, 1997 (DCP/NRC1092). Results were provided for selected locations using the procedure outlined in the open item. The results showed that the member forces, including those induced by the construction sequence and settlement, are within the design capacity. This confirms the adequacy of the design described in the SSAR.

SSAR Revision:

Add the following at the end of subsection 3.8.5.4.3. This material was included in SSAR, Revision 20. The redline portion will be included in revision 21 A draft revision 21 is also attached..

Member forces in the basemat considering settlement during construction differ from those obtained from the analyses on uniform elastic soil springs described in subsection 3.8.5.4.1. Although the bearing pressures at the end of construction are similar in the two analyses, the resulting member forces differ due to the progressive changes in structural configuration during construction. The design using the results of the analyses of subsection 3.8.5.4.1 provides sufficient structural strength to resist the specified loads including bearing reactions on the underside of the basemat. However, this may require redistribution of stresses locked in during early stages of construction. A confirmatory evaluation was performed ~~for the five critical locations of the nuclear island basemat~~ to demonstrate that the member forces due to design basis loads, including locked-in forces due to construction settlement ~~calculated by elastic analyses~~, remain within the capacity of the section. The evaluation was performed for the following locations which were selected as locations where the effect of locked in member forces were judged to be most significant. The evaluation was performed for the safe shutdown earthquake which governs the design of the basemat.

- North edge of shield building
- South edge of shield building
- North east edge of shield building
- South west edge of shield building
- Middle of north auxiliary building below wall K

The governing scenario is the case with a delay in the auxiliary building construction for the soft soil site with alternating layers of sand and clay. The delay is postulated to occur just prior to the stage where the auxiliary building walls are constructed. The evaluation used the following steps:

~~• Member forces in the basemat just prior to the stage where the auxiliary building walls are constructed were obtained from the construction settlement analyses. Bearing pressures were also obtained at the same stage and at the end of construction.~~

~~• Member forces at the end of construction were obtained by adding the change in member force that occurs after the construction of the auxiliary building walls has started. The increase in design member forces was obtained from the increase in bearing pressure after construction of the auxiliary building walls using the results of the analyses of the fully constructed nuclear island described in subsection 3.8.5.4.1.~~

~~• The bearing pressure and member force due to the safe shutdown earthquake were obtained from the analyses described in subsection 3.8.5.4.1. The member force was added to the dead load member force at the end of construction and compared against the basemat design capacity.~~

1. Member forces in the basemat just prior to the stage where the auxiliary building walls are constructed are extracted from the construction settlement analyses.

2. Bearing pressures under the auxiliary building just prior to the stage where the auxiliary building walls are constructed and at the end of the construction period (including long term settlement) are extracted from the construction settlement analyses. The differences in the bearing pressures at these two time points are calculated.

3. The increase in member forces due to unit bearing pressure is obtained from the analyses described in subsection 3.8.5.4.1. These additional member forces are based on the response of the fully constructed nuclear island.

4. The results of steps 2 and 3 are used to calculate the increase in member forces after the stage where the auxiliary building walls are constructed. This increase in member forces is added to the member forces taken from the construction settlement analyses in step 1. This provides the resultant dead load solution at the end of the construction period (including long term settlement).

5. The bearing pressure increase due to the safe shutdown earthquake is obtained from the analyses described in subsection 3.8.5.4.1.

6. Member forces due to the safe shutdown earthquake are calculated using the results of steps 3 and 5.

7. The member forces due to the safe shutdown earthquake are added to the dead load member forces at the end of construction obtained in step 4 and compared against the basemat design capacity.

The member forces for the load combination of dead load plus safe shutdown earthquake, including the member forces locked-in ~~forces,~~ during various stages of plant construction, are within the design capacity for the five critical locations. The evaluation demonstrates that the member forces including locked-in forces calculated by elastic analyses remain within the capacity of the section.

Add new paragraph as follows at the end of subsection 3.8.5.5:

Settlement during construction results in deformations and member forces in the mat. These deformations and member forces are locked in as the shear walls are constructed. The member forces are calculated in the analyses considering the construction sequences described in subsection 3.8.5.4.3. Post-construction load combinations include the member forces in the basemat locked in during construction.

3.8.5.4.3 Analysis for Loads during Construction

Construction loads are evaluated in the design of the nuclear island basemat. This evaluation is performed for soil sites meeting the site interface requirements of subsection 2.5.4 at which settlement is predicted to be maximum. In the expected basemat construction sequence, concrete for the mat is placed in a single placement. Construction continues with a portion of the shield building foundation and containment internal structure and the walls of the auxiliary building. The critical location for shear and moment in the basemat is around the perimeter of the shield building. Once the shield building and auxiliary building walls are completed to elevation 82′ 5″, the load path changes and loads are resisted by the basemat stiffened by the shear walls.

The analyses account for the construction sequence, the associated time varying load and stiffness of the nuclear island structures, and the resulting settlement time history. To

maximize the potential settlement, the analyses consider a 360 feet deep soft soil site with soil properties consistent with the soft soil case described in subsection 2A.2. Two soil profiles are analyzed to represent limiting foundation conditions, and address both cohesive and cohesionless soils and combinations thereof:

- A soft soil site with alternating layers of sand and clay. The assumptions in this profile maximize the settlement in the early stages of construction and maximize the impact of dewatering.

- A soft soil site with clay. The assumptions maximize the settlement during the later stages of construction and during plant operation.

The analyses focus on the response of the basemat in the early stages of construction when it could be susceptible to differential loading and deformations. As subsequent construction incorporates concrete shear walls associated with the auxiliary building and the shield building, the structural system significantly strengthens, minimizing the impact of differential settlement. The displacements, and the moments and shear forces induced in the basemat are calculated at various stages in the construction sequence. These member forces are evaluated in accordance with ACI 349 using the load factors given in Table 3.8.4-2. Three construction sequences are examined to demonstrate construction flexibility within broad limits.

- A base construction sequence which assumes no unscheduled delays. The site is dewatered and excavated. Concrete for the basemat is placed in a single pour. Concrete for the exterior walls below grade is placed against the vertical sides of the excavation after the basemat is in place. Exterior and interior walls of the auxiliary building are placed in 16 to 18-foot lifts.

- A delayed shield building case which assumes a delay in the placement of concrete in the shield building while construction continues in the auxiliary building. This bounding case maximizes tension stresses on the top of the basemat. The delayed shield building case assumes that no additional concrete is placed in the shield building after the pedestal for the containment vessel head is constructed. The analysis incorporates construction in the auxiliary building to elevation 117′-6″ and thereafter assumes that construction is suspended.

- A delayed auxiliary building case which assumes a delay in the construction of the auxiliary building while concrete placement for the shield building continues. This bounding case maximizes tension stresses in the bottom of the basemat. The delayed auxiliary building case assumes that no concrete is placed in the auxiliary building after the basemat is constructed. The analysis incorporates construction in the shield building to elevation 84′-0″ and thereafter assumes that construction is suspended.

For the base construction sequence, the largest basemat moments and shears occur at the interface with the shield building before the connections between the auxiliary building and the shield building are credited. Once the shield building and auxiliary building walls are completed to elevation 82′ 6″, the load path for successive loads changes and the loads are

resisted by the basemat stiffened by the shear walls. Dewatering is discontinued once construction reaches grade, resulting in the rebound of the subsurface.

Of the three construction scenarios analyzed, the delayed auxiliary building case results in the largest demand for the bottom reinforcement in the basemat. The delayed shield building results in the largest demand for the top reinforcement in the basemat. The analyses of the three construction sequences demonstrate the following:

- The design of the basemat and superstructure accommodates the construction-induced stresses considering the construction sequence and the effects of the settlement time history.

- The design of the basemat can accommodate delays in the shield building so long as the auxiliary building construction is suspended at elevation 117′ 0″. Resumption in construction of the auxiliary building can proceed once the shield building is advanced to elevation 100′ 0″.

- The design of the basemat can accommodate delays in the auxiliary building so long as the shield building construction is suspended at elevation 84′ 0″ feet. Resumption in construction of the shield building can proceed once the auxiliary building is advanced to elevation 100′ 0″.

- After the structure is in place and cured to elevation 100′ 0″, the basemat and structure act as an integral 40 foot deep structure and the loading due to construction above this elevation is not expected to cause significant additional flexural demand with respect to the basemat and the shield building concrete below the containment vessel. Accordingly, there is no need for placing constraints on the construction sequence above elevation 100′ 0″.

The site conditions considered in the evaluation provide reasonable bounds on construction induced stresses in the basemat. Accordingly, the AP600 basemat design is adequate for practically all soil sites and it can tolerate major variations in the construction sequence without causing excessive deformations, moments and shears due to settlement over the plant life.

The analyses of alternate construction scenarios show that member forces in the basemat are acceptable subject to the following limits imposed for soft soil sites on the relative level of construction of the buildings prior to completion of both buildings at elevation 82′6″:

- Concrete may not be placed above elevation 82′ 6″ for the shield building or containment internal structure.

- Concrete may not be placed above elevation 117′6″ in the auxiliary building.

Member forces in the basemat considering settlement during construction differ from those obtained from the analyses on uniform elastic soil springs described in subsection 3.8.5.4.1.

Although the bearing pressures at the end of construction are similar in the two analyses, the resulting member forces differ due to the progressive changes in structural configuration during construction. The design using the results of the analyses of subsection 3.8.5.4.1 provides sufficient structural strength to resist the specified loads including bearing reactions on the underside of the basemat. However, this may require redistribution of stresses locked in during early stages of construction. A confirmatory evaluation was performed ~~for the five critical locations of the nuclear island basemat~~ to demonstrate that the member forces due to design basis loads, including locked-in forces due to construction settlement ~~calculated by elastic analyses~~, remain within the capacity of the section. The evaluation was performed for the following locations which were selected as locations where the effect of locked in member forces were judged to be most significant. The evaluation was performed for the safe shutdown earthquake which governs the design of the basemat.

- North edge of shield building
- South edge of shield building
- North east edge of shield building
- South west edge of shield building
- Middle of north auxiliary building below wall K

The governing scenario is the case with a delay in the auxiliary building construction for the soft soil site with alternating layers of sand and clay. The delay is postulated to occur just prior to the stage where the auxiliary building walls are constructed. The evaluation used the following steps:

- ~~Member forces in the basemat just prior to the stage where the auxiliary building walls are constructed were obtained from the construction settlement analyses. Bearing pressures were also obtained at the same stage and at the end of construction.~~

- ~~Member forces at the end of construction were obtained by adding the change in member force that occurs after the construction of the auxiliary building walls has started. The increase in design member forces was obtained from the increase in bearing pressure after construction of the auxiliary building walls using the results of the analyses of the fully constructed nuclear island described in subsection 3.8.5.4.1.~~

- ~~The bearing pressure and member force due to the safe shutdown earthquake were obtained from the analyses described in subsection 3.8.5.4.1. The member force was added to the dead load member force at the end of construction and compared against the basemat design capacity.~~

1. Member forces in the basemat just prior to the stage where the auxiliary building walls are constructed are extracted from the construction settlement analyses.

2. Bearing pressures under the auxiliary building just prior to the stage where the auxiliary building walls are constructed and at the end of the construction period (including long

term settlement) are extracted from the construction settlement analyses. The differences in the bearing pressures at these two time points are calculated.

3. The increase in member forces due to unit bearing pressure is obtained from the analyses described in subsection 3.8.5.4.1. These additional member forces are based on the response of the fully constructed nuclear island.

4. The results of steps 2 and 3 are used to calculate the increase in member forces after the stage where the auxiliary building walls are constructed. This increase in member forces is added to the member forces taken from the construction settlement analyses in step 1. This provides the resultant dead load solution at the end of the construction period (including long term settlement).

5. The bearing pressure increase due to the safe shutdown earthquake is obtained from the analyses described in subsection 3.8.5.4.1.

6. Member forces due to the safe shutdown earthquake are calculated using the results of steps 3 and 5.

7. The member forces due to the safe shutdown earthquake are added to the dead load member forces at the end of construction obtained in step 4 and compared against the basemat design capacity.

The member forces for the load combination of dead load plus safe shutdown earthquake, including the member forces locked-in ~~forces,~~ during various stages of plant construction, are within the design capacity for the five critical locations. The evaluation demonstrates that the member forces including locked-in forces calculated by elastic analyses remain within the capacity of the section.

3.8.5.5 Structural Criteria

The analysis and design of the foundation for the nuclear island structures are according to ACI-349 with margins of structural safety as specified within it. The limiting conditions for the foundation medium, together with a comparison of actual capacity and estimated structure loads, are described in Section 2.5. The minimum required factors of safety against sliding, overturning, and flotation for the nuclear island structures are given in Table 3.8.5-1.

The basemat below the auxiliary building is designed for shear in accordance with the following supplemental provisions which are based on the requirements for continuous deep flexural members in ACI 318-95.

- The design for shear is based on 11.1 through 11.5 of ACI 349-90 except that the critical section measured from the face of the support is taken at a distance of 0.15 l_n.

- Shear strength, V_n, is not taken greater than $8\sqrt{f_c'}\, b_w\, d$ when l_n/d is less than 2. When l_n/d is between 2 and 5,

$$V_n = 2/3\,(10 + l_n/d)\,\sqrt{f_c'}\, b_w\, d$$

- Minimum vertical shear reinforcement is provided in each bay. The area of vertical shear reinforcement, A_v, is not less than $0.0015\, b_w\, s$ and spacing of shear reinforcement, s, does not exceed d/2, nor 24 in.

- Shear reinforcement required at the critical section is used throughout the span.

The terms φ, V_c, A_v, b_w, s, and f_y are defined in ACI 349.

Settlement during construction results in deformations and member forces in the mat. These deformations and member forces are locked in as the shear walls are constructed. The member forces are calculated in the analyses considering the construction sequences described in subsection 3.8.5.4.3. Post-construction load combinations include the member forces in the basemat locked in during construction.

Question 440.784F (OITS - 6366)

LCO 3.4.15 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.15 specifies that either the RNS suction relief valve, or the RCS depressurized and an RCS vent path of greater than or equal to 5.4 square inches, or both, shall be operable with the accumulators isolated. The operability of the PORV in STS is deleted because AP600 does not have, nor require, a pressurizer PORV for overpressure protection. A NOTE is added to restrict startup of a RC pump when the RCS temperature is greater than 200oF and the pressurizer level is greater than or equal to 92 percent span. This limitation is necessary to limit the heat input transient to within the capacity of the RNS suction relief valve. This is acceptable.

The applicability of LCO 3.4.15 is different from the STS LTOP LCO which is applicable for MODE 4 when all cold leg temperatures are less than 275°F, and MODE 5. The applicability of LCO 3.4.15 is for MODES 4 and 5 operations with the RNS aligned and open to the RCS and the RCS temperature less than 350°F, as well as for MODE 6 when the reactor vessel head is on. Limiting the LCO 3.4.15 applicability to the alignment of RNS to the RCS is necessary to ensure that the RNS suction relief valve is available for pressure protection. LCO 3.4.7 requires operability of the pressurizer safety valves (PSVs) to provide overpressure protection during MODES 1, 2, 3, and 4 with the RNS isolated or RCS temperature greater than or equal to 350°. However, in MODE 5 with RCS temperature less than 200°F, and the RNS is isolated from the RCS, neither LCOs 3.4.7 nor 3.4.15 are applicable to provide overpressure protection. In addition, with the PSV lift setting of greater than or equal to 2460 psig, the PSVs will not be able to provide LTOP protection. The applicant should evaluate or revise LCO applicability to provide suitable LTOP protection with the RNS isolated from the RCS. This is open item.

Response (Revision 1):

The mode applicability will be changed to be consistent with the Standard Technical Specifications. This change will provide LTOP protection when the RCS temperature is below 275°F. Above this temperature, overpressure protection is provided by the RCS safety valves. Technical Specification 3.4.7 is also revised to ensure the safety valves are available above 275°F.

SSAR Revision:

Technical Specifications 3.4.15 and 3.4.7.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 Pressurizer Safety Valves

LCO 3.4.7 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RNS isolated or RCS temperature ≥ ~~350°F~~ 275°F.

.....NOTE.....

The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions.

This exception is allowed for 36 hours following entry into MODE 3, provided a preliminary cold setting was made prior to heatup.

.....

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with RNS aligned to the RCS and RCS temperature < ~~350°F~~ 275.	6 hours 24 hours

440.784F(21)-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify each pressurizer safety valve OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%.	In accordance with the Inservice Testing Program

44C 7E4F(21)-3

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 Pressurizer Safety Valves

BASES

BACKGROUND

The two pressurizer safety valves provide, in conjunction with the Protection and Safety Monitoring System (PMS), overpressure protection for the RCS. The pressurizer safety valves are totally enclosed, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2733.5 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The minimum relief capacity for each valve, 400,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The pressurizer safety valves discharge into the containment atmosphere. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 when the reactor vessel head is on; however, in MODE 4 with the RNS aligned, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.15, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

440.784F(R1)

BASES

BACKGROUND (continued)

The consequences of exceeding the ASME Code, Section III pressure limit (Ref. 1) could include damage to RCS components, increased LEAKAGE, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the SSAR (Ref. 3) that require safety valve actuation assume operation of two pressurizer safety valves to limit increases in the RCS pressure. The overpressure protection analysis (Ref. 2) is also based on operation of the two safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;
b. Loss of reactor coolant flow;
c. Loss of external electrical load;
d. Locked rotor; and
e. Loss of AC power/loss of normal feedwater

Detailed analyses of the above transients are contained in Reference 3. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer Safety Valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig.

The limit protected by this specification is the Reactor Coolant Pressure Boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

(continued)

440.784F(01)-5

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 with the RNS isolated or with the RCS temperature > ~~350~~ 275°F, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 with RNS open and in MODE 5, because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into Modes 3 and 4 with the lift setpoints outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 1 hour. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two pressurizer safety valves are inoperable, the plant must be placed in a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with the RNS aligned to the RCS and RCS temperature < ~~350~~ 275°F within 24 hours.

(continued)

440.784(21)-6

BASES

ACTIONS

B.1 and B.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With the RNS aligned to the RCS, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested one at a time and in accordance with the requirements of ASME Code Section XI (Ref. 4), which provides the activities and Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the values are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7614.3.

2. [WCAP-7769, "Topical Report on Overpressure Protection."]

3. AP600 SSAR, Chapter 15, "Accident Analyses."

4. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

440.784F(81) - 7

Question 440.785F (OITS - 6367)

Reactor Vessel Head Vent (RVHV)

In accordance with the requirement of 10 CFR 50.34(f)(2)(vi), the AP600 design uses the reactor vessel head vent valves to remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation through steam generators resulting from the accumulation of noncondensable gases in the reactor system. The RVHV valves, which can be operated from the main control room to provide an emergency reactor coolant letdown path, can also be used to prevent pressurizer overfill following long-term loss of heat sink events. A flow-limiting orifice is provided downstream of each set of head vent valves to limit the emergency letdown flow rate. The emergency letdown of the reactor coolant is credited in the safety analysis of some design basis events that otherwise could result in the pressurizer overfill. Therefore, a LCO is required for the RVHV system per TS screening Criterion 3. The AP600 TS does not include the RVHV system. This is open item.

Response (Revision 1):

Technical Specification 3.4.17, "Reactor Vessel Head Vent" will be incorporporated in the next revision of the Technical Specifications. Attached is a revision of the proposed Technical Specification 3.4.17 based on NRC staff comments at the 1/28/98 closeout meeting.

SSAR Revision:

Technical Specifications

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Reactor Vessel Head Vent (RVHV)

LCO 3.4.17 The Reactor Vessel Head Vent shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4, not on RNS cooling.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One flow path inoperable.	A.1 Restore flow path to OPERABLE status.	72 hours
B. Two flow paths inoperable.	B.1 Restore at least one flow path to OPERABLE status.	6 hours
C. Required Action and associated Completion Time not met. OR Requirements of LCO not met for reasons other than Conditions A or B.	C.1 Be in MODE 3. AND C.2 Be in MODE 4, with the RCS being cooled by the RNS.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify that each RVHV valve is OPERABLE by stroking them open.	In accordance with the Inservice Testing Program

440.78SF(81)-2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Reactor Vessel Head Vent (RVHV)

BASES

BACKGROUND

The reactor vessel head vent (RVHV) is designed to assure that long-term operation of the Core Makeup Tanks (CMTs) does not result in overfilling of the pressurizer during Condition II Design Basis Accidents (DBA). The RVHV can be manually actuated by the operators in the main control room to reduce the pressurizer water level during long-term operation of the CMTs.

The RVHV consists of two parallel flow paths each containing two RVHV isolation valves in series. The RVHV valves are connected to the reactor vessel head via a common line. The outlets of the RVHV flow paths combine into one common discharge line which connects to a single ADS discharge header that discharges to spargers located in the incontainment refueling water storage tank (IRWST). The RVHV valves are 1 inch valves with DC solenoid operators.

The RVHV valves are designed to open when actuated by the operator, and to reclose when actuated by the operator from the main control room.

The number and capacity of the RVHV flow paths are selected so that letdown flow from the RCS is sufficient to prevent pressurizer overfill for events where extended operation of the CMTs causes the pressurizer water level to increase. Although realistic evaluations of the Condition II non-LOCA events does not result in pressurizer overfill, conservative analyses of some of these events can result in pressurizer overfill if no operator actions are assumed.

APPLICABLE SAFETY ANALYSIS

For Condition II non-LOCA events such as inadvertent passive core cooling system operation and chemical and volume control system malfunction, the use of the RVHV may be required to prevent long-term pressurizer overfill (Reference 1).

For LOCA events, the RVHV is not required.

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BASES

LCO

The requirement that all four RVHV valves be OPERABLE ensures that upon actuation, the RVHV can reduce the pressurizer water level as assumed in the DBA safety analyses.

For the RVHV to be considered OPERABLE, all four valves must be closed and OPERABLE (capable of opening from the main control room).

APPLICABILITY

In MODES 1, 2, 3, and 4 with the RCS not being cooled by the RNS, the RVHV must be OPERABLE to mitigate the potential consequences of any event which causes an increase in the pressurizer water level that could otherwise result in overfilling of the pressurizer.

In Mode 4, with the RCS being cooled by the RNS, and in Modes 5 and 6, operation of the CMTs or CVS will not result in a pressurizer overfill event.

ACTIONS

A.1

If one or two RVHV valves in a single flow path are determined to be inoperable, the flow path is inoperable. The remaining OPERABLE RVHV flow path is adequate to perform the required safety function. A Completion Time of 72 hours is acceptable since the OPERABLE RVHV paths can mitigate DBAs without a single failure.

B.1

If two flow paths are determined to be inoperable, the RVHV is degraded such that the remaining system capacity may not be adequate for some DBA non-LOCA analysis. A Completion Time of 24 hours is permitted to restore at least one flow path. This Completion Time is acceptable considering that the realistic analysis of these non-LOCA events do not result in pressurizer overfill.

C.1 and C.2

If the Required Actions and associated Completion Times are not met or the requirements of LCO 3.4.17 are not met for reasons other than Conditions A or B, the plant must be brought to MODE 4 where the probability and consequences of an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 8 hours and to MODE 4 within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner, without challenging plant systems.

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

This Surveillance requires verification that each RVHV valve strokes to its fully open position. The Surveillance Frequency for demonstrating valve OPERABILITY references the Inservice Testing Program.

REFERENCES

1. AP600 SSAR, Section 15.5, "Increase in Reactor Coolant System Inventory."

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