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Docket No.: STN-52-003

March 4, 1994

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

ATTENTION: R. W. BORCHARDT

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letter January 26, 1994. In addition, revised responses for a number of previously provided responses are included.

A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

These responses are also provided as electronic files in WordPerfect 5.1 format with Mr. Hasselberg's copy.

If you have any questions on this material, please contact Mr. Brian A. McIntyre at 412-374-4334.

Brian A McIntyre

Nicholas J. Liparulo, Manager
Nuclear Safety & Regulatory Activities

/nja

Enclosure

cc: B. A. McIntyre - Westinghouse
F. Hasselberg - NRR

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NTD-NRC-94-4075
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED MARCH 4,1994

RAI No.	Issue
220.026	Structural integrity testing of steel containment
220.031	Stress calculation by ASME criterion
220.035	Corrosion allowance for steel containment plates
230.015R01	Soil-specific analyses
230.026	GDC for seismic Cat II
252.010R01	Piping stresses for different sites
440.026R01	ATWS
440.035R01	LOFTRAN/NOTRUMP AP600 Models

Question 220.26

Provide information on measurements taken at critical locations during pre-operational structural integrity testing (SIT) of the steel containment. This information may be useful in validating the containment analysis methods (Section 3.8.2 of the SSAR).

Response:

The containment vessel will be constructed and tested in accordance with the ASME Code, Section III as described in SSAR Subsection 3.8.2. The ASME Code, Section III does not require that measurements be taken at critical locations during the pre-operational structural integrity testing. No deflection or strain measurements are planned during the structural integrity testing.

Acceptance criteria for the structural integrity test are given in the ASME Code, Section III. They include examination for leakage after reducing the pressure from the test pressure to design pressure. This examination covers joints, connections and regions of high stress but permits a waiver for inaccessible areas such as those portions of the vessel embedded in concrete. This waiver is required for those portions of the vessel embedded in concrete. For the AP600 this waiver will also be requested to exclude visual inspection of surfaces behind the air baffle. Regions behind the air baffle will be examined on a sample basis.

SSAR Revision: NONE

Question 220.31

In SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor Designs," the staff proposed that the containment be evaluated for the credible severe accidents against the stress limits of the ASME Level C Service Limit. Westinghouse states that for tensile stresses at the cylindrical portion of the containment, this results in a pressure capacity equal to 125 psig by ASME stress intensity criterion. However, the staff estimated it as 114 psig using ASME stress intensity criterion based on the theoretical stress calculations.* Explain this discrepancy (Section 3.8.2 of the SSAR).

* Timoshenko, S. and Woinowsky-Krieger, S., Theory of Plates and Shells, pp 4884-485, second edition, 1959, McGraw-Hill,

Response:

Containment vessel membrane stresses due to design internal pressure are shown on sheet 2 of SSAR Figure 3.8.2-5. These were evaluated in accordance with the ASME Code to determine the maximum pressure that could be accommodated within Service Level C. This evaluation considered general membrane, local membrane and bending stresses as specified in the ASME Code. The largest stresses occur at discontinuities and are evaluated as local membrane and bending stresses with an allowable stress of 1.5 times yield. These local stresses did not control the maximum Service Level C pressure. The maximum circumferential stress away from discontinuities is 21.6 ksi for the internal pressure of 45 psi; this is evaluated as a general membrane stress for which the ASME Service Level C stress intensity limit is the yield stress of 60 ksi at ambient temperature. This resulted in a maximum allowable pressure within ASME Service Level C limits of 60/21.6 x 45 = 125 psi.

The internal pressure of 114 psig identified in the question would be correct using an allowable stress equal to yield, provided there is no other discontinuity in the vicinity of the cylinder / head junction. Theoretical stress calculations give a maximum circumferential membrane stress of 23.7 ksi for the internal pressure of 45 psig at a distance of about 51 inches from the junction. However, due to the presence of the crane girder, the maximum circumferential membrane stress occurring in the portion of the shell between the junction and the crane girder is only 22.7 ksi (see SSAR Figure 3.8.2-5, sheet 2 of 4). This is less than the allowable primary local membrane stress. The distance over which the circumferential membrane stress exceeds S_m is about 26 inches which is less than $1.0\sqrt{(Rt)}$ = 36 inches. Under increasing pressure, yield would initiate at this location at a pressure of 45x60/22.7 = 119 psig. However, this yield is local and would not result in excessive deformation. The Service Level C allowables recognize this and allow a stress of 1.5 S_y for primary local membrane stresses computed by elastic analyses.

SSAR Revisions: NONE

Question 220.35

Provide a corrosion allowance to be used for the proposed 60-year plant design life and its technical basis. Also, indicate whether post-weld heating during construction is provided for the steel containment plates (Section 3.8.2 of the SSAR).

Response:

Corrosion of the containment vessel is addressed in the responses to RAIs 252.23 through 252.28. Additional information on the seals at the transition region is provided in the response to RAI 220.25.

Post-weld heat treatment (PWHT) is in accordance with the ASME Code, Section III. In general areas of the vessel, PWHT is not required since the plate thickness does not exceed 1.75 inches. PWHT is required for welds in the thicker penetration insert plates (e.g. equipment hatch and airlocks) and for welds joining nozzles or penetrations with an internal diameter greater than 2 inches. The PWHT is performed in the shop with the exception of the equipment hatches. PWHT of the hatches will be performed in the field at those sites where shipping limitations prevent shipment to the site in one piece.

SSAR Revision: NONE

Question 230.15

Section 3.7.2.1.2 of the SSAR states that "Certain subsystems...are analyzed using the time histories obtained from a series of soil-specific analyses." What are these soil-specific analyses? Provide details of these analyses.

Response (Revision 1):

The soil-specific analyses correspond to the design soil profiles presented in subsection 3.7.1.4. The methods of analysis are presented in Subsection 3.7.3. SSE time history analysis ~~will be performed~~ for the reactor coolant loop piping ~~by December 1993~~ is described in Appendix 3C of the AP600 SSAR, Revision 1.

SSAR Revision: NONE

Question 230.26

Page 3.7-1 of Section 3.7 of the SSAR states that the AP600 standard plant used a three-level seismic classification, i.e., seismic Category I, seismic Category II and non-Category I. Section 3.7 specifies the general design requirements for the seismic Category I items. It also specifies the general design requirements for the non-Category I items. However, the general design requirements were not provided for the seismic Category II items. Provide this information.

Response:

Design requirements for seismic Category II structures were added in SSAR Revision 1 (01/13/94) Subsection 3.7.2.8. Design requirements for seismic Category II piping and supports are described in SSAR Subsection 3.7.3.13.3. Seismic Category II cable tray and HVAC duct supports are designed and analyzed using the same criteria as seismic Category I supports described in SSAR Subsections 3.8.4.1.2 and 3.8.4.1.5.

SSAR Revisions: NONE

Question 252.10

Section 3B of the SSAR discusses the LBB evaluation for the reactor coolant loop piping. The SSAR indicates that two different soil conditions have been considered in deriving piping stresses. Discuss how these piping stresses represent the worst condition of all potential sites within the scope of AP600 applications.

Response (Revision 1):

~~As described in Appendix 3B of the SSAR, two soil conditions for the reactor coolant loop pipe stress analyses were chosen to provide preliminary stresses for a sample of the application of the leak-before-break methodology. Analyses for other soil conditions will be performed by December, 1993. These loop analyses will represent the worst condition of all potential sites within the scope of AP600 applications.~~

Appendix 3C of the AP600 SSAR, Revision 1, describes the reactor coolant loop stress analysis. Based on a comparison of the SSE floor response spectra for three soil conditions, the limiting condition for the reactor coolant loop piping analysis is the hard rock case. A time history analysis for the hard rock case is performed to calculate the SSE loads that are used in the leak-before-break analysis.

SSAR Revision: NONE

Question 440.26

Section 15.8 of the SSAR states that AP600 plant design includes a diverse actuation system, which provides for all of the AMSAC protection features mandated for Westinghouse plants plus a diverse reactor scram, and thus meets the ATWS rule. However, it does not provide an ATWS analysis to demonstrate that AP600 ATWS response is consistent with that considered by the staff in its formulation of the 10 CFR 50.62 design requirements for current plants. Provide such an analysis.

Response (Revision 1):

General Background

This RAI requests that Westinghouse submit a deterministic ATWS analysis. In response to this request, a deterministic complete loss of normal feedwater ATWS analysis has been performed. The purpose of this analysis is to demonstrate that the AP600 ATWS response characteristics are comparable to the responses of other Westinghouse plants and are, therefore, consistent with the bases considered by the staff in formulation of 10 CFR 50.62 design requirements. The analysis methodology used is based on analyses presented in previous Westinghouse submittals (References 440.26-4 & 440.26-5).

For Westinghouse plants, the ATWS rule (10 CFR 50.62) requires the installation of ATWS mitigation systems actuation circuitry (AMSAC), which is separate from the reactor protection system, to trip the turbine and initiate residual heat removal. The AP600 design includes a Diverse Actuation System (DAS) that provides the AMSAC protection features mandated for Westinghouse plants by 10 CFR 50.62 plus, among other functions, a diverse reactor scram. The basis for the ATWS rule requirements, as outlined in SECY-83-293 (Reference 440.26-3), is to reduce the risk of ATWS related core damage to less than 10^{-5} per reactor-year.

Identification of Causes and Accident Description

The most limiting ATWS events for Westinghouse plants were found in a previous study (Reference 440.26-4) to be the heatup transients caused by a reduction in the heat removal capability of the secondary (steam) side of the plant. Because of the strong negative moderator temperature feedback in a PWR, heatup accidents resulting from a loss of heat sink cause the nuclear heat generation rate to decrease until the reactor power matches the heat extraction from the passive residual heat removal system. These events proceed relatively slowly in the AP600 due to the large water inventory in both the primary and secondary sides.

Results for previous Westinghouse designs presented in References 440.26-4 and 440.26-5 indicate that the reactor reaches a steady-state condition with no impairment of reactor coolant system integrity or significant fuel damage. Consistent with the very low probability of an ATWS event, these analyses employ several best-estimate assumptions. Turbine trip and the passive residual heat removal system are assumed to be available to mitigate the event.

The AP600 transient calculations assume actuation of the passive residual heat removal system and a turbine trip as the primary actions needed to mitigate the effects of an ATWS. Both of these functions are assumed to actuate on a signal generated by the DAS when the decreasing steam generator water level reaches the wide range low level setpoint. This same DAS signal also generates a diverse reactor trip that is not modeled in the base case presented in this report. Such a diverse reactor trip function is not needed and therefore not implemented in the standard Westinghouse AMSAC system. Its inclusion in the DAS provides an increased level of protection for the AP600. This function represents an independent means of initiating RCCA insertion, in the unlikely event that the reactor protection system fails to generate a required reactor trip signal.

There are two distinct failure categories identified in the Subsection F.2.22 of the AP600 Probabilistic Risk Assessment (PRA) (Reference 440.26-6) which could prevent the control rods from inserting after receipt of a reactor trip signal during an anticipated transient. The two categories are a failure in either the mechanical or electrical portion of the reactor trip system. For the case of a mechanical failure; turbine trip, startup feedwater, and the passive residual heat removal system are assumed to be available via the control and protection systems. For the case of an electrical failure; the failure in the reactor trip system is conservatively assumed to be common to the entire reactor protection system. Therefore, the DAS must provide turbine trip and passive residual heat removal system actuation on low steam generator wide range level. Since the DAS setpoints are outside the range of normal reactor protection system setpoints, turbine trip and passive residual heat removal system actuation are delayed for the case of an electrical failure and the resulting transient is more severe. Only the limiting case of an electrical failure is presented in this report.

Limiting Criteria

Consistent with previous assessments and the criteria used to define a successful event outcome for PRA purposes, it is conservatively assumed that if any one of the following requirements are not met, unacceptable core damage can occur during an ATWS transient (Reference 440.26-10):

1. The peak reactor coolant system pressure must not exceed the pressure limit corresponding to the service limit stress of the ASME Boiler and Pressure Vessel Code for Level C ("emergency condition") events (Reference 440.26-7). The pressure limit assumed in Reference 440.26-10 is 3200 psig.

2. Reactor coolant system heat removal must be adequate, both before and after the core is brought subcritical. For the AP600, long-term heat removal is provided by the passive residual heat removal system. Note that for the ATWS event, this long-term cooling requirement must be met without RCCA insertion.

3. Actions must be initiated to achieve subcriticality within an acceptable time period. For AP600, operator actions are available to manually initiate boration independent of the reactor protection system.

With respect to achieving subcriticality, even though the DAS provides a diverse reactor trip signal, it is conservatively assumed for analysis purposes that the shutdown condition must be achieved without RCCA insertion. In the AP600, boron injection from the CMTs produces the required shutdown condition. The operator can manually actuate the CMTs via the DAS. An alternative means to achieve subcriticality is for the operator to initiate boration using the makeup pumps in the Chemical and Volume Control System (CVS).

Method of Analysis

The LOFTRAN code (Reference 440.26-2), including the modifications for the AP600 passive safety systems as described in Appendix 15B of the SSAR, is used to compute the reactor transient response to the ATWS event for AP600. The event analyzed is a complete loss of normal feedwater (LONF) with reactor trip signals being generated, but no RCCA insertion actually taking place. Previous studies (References 440.26-4 and 440.26-5) have shown that this event typically produces the maximum reactor coolant system pressure for Westinghouse PWRs.

Major assumptions made in this analysis are:

1. The transient is initialized from nominal full power conditions.

2. An AP600 specific Doppler feedback model for system conditions indicative of an ATWS is input as a function of power and core inlet mass flow.

3. The moderator temperature coefficient (MTC) used in the analysis is -7.3 pcm/°F. This value represents a coefficient, arrived at iteratively, that gives a peak reactor coolant system pressure of approximately 3200 psig during the ATWS. The selection of MTC in this manner is consistent with the analysis methodology used in previous Westinghouse submittals (References 440.26-9 and 440.26-10). The magnitude of the MTC used in this analysis is comparable to that used in the previous submittals and is within the expected range for AP600.

4. The ANS-5.1-1979 decay heat model (+ 2 sigma) is used. (Reference 440.26-8)

5. Both pressurizer safety valves are assumed available. The relief model assumes 3 percent and 10 percent pressure accumulation for steam and water relief, respectively. The AP600 does not include power operated relief valves for the pressurizer.

6. Main feedwater supply to both steam generators falls to zero in 4 seconds, with no main feedwater afterwards.

7. The DAS is assumed to actuate on the wide range steam generator low level DAS signal. The analysis setpoint is considered to be conservatively low, and will therefore delay DAS actuation. Given a fixed MTC, a higher level setpoint and an earlier DAS actuation would produce a lower peak pressure for a given ATWS event. Alternatively, if a case is targeted to produce the peak pressure of 3200 psig, a higher level setpoint would allow meeting the limit pressure with a less negative moderator temperature coefficient.

8. DAS setpoints are typically set outside the range of normal reactor protection system setpoints or, as an alternative, the DAS functions include delays that allow time for the reactor protection system function to actuate. Since a conservatively low DAS wide range steam generator level analysis setpoint is assumed, no additional delay on this signal is assumed.

9. The turbine is assumed to trip 4.0 seconds after the wide range steam generator low level DAS setpoint is reached.

10. The passive residual heat exchanger valves are assumed to be fully opened 10 seconds after the low wide range steam generator level setpoint is reached.

11. The design value of 40 percent steam dump to the condenser is modeled for conservatism.

12. Steamline isolation, which would actuate on the low steamline pressure signal, is conservatively not modeled in the limiting ATWS case for AP600. If assumed in the analysis, steamline isolation would occur prior to the predicted turbine trip (per Item #9, above). Steamline isolation, during this event, has the same beneficial effect on the reactor coolant system pressure transient as a turbine trip. That is, the steamline isolation would "bottle up" the steam generator, thereby reducing the secondary system heat removal rate, and produce a more rapid reactor coolant system heatup. The result is increased reactivity feedback that would produce a more rapid reduction in the core power.

13. Startup feedwater is conservatively assumed to be unavailable.

14. Following turbine trip, steam relief through the spring-loaded steamline safety valves is assumed if the steamline pressure exceeds the safety-valve setpoint (1100 psia) with a 3 percent allowance for accumulation.

15. As an additional conservatism, CMT actuation (safety injection) and the associated reactor coolant pump trip is not modeled for the duration of the transient analyzed. During the analyzed event, the low steamline pressure safety injection signal is generated by the primary protection system, but the resulting CMT actuation and reactor coolant pump trip signals are ignored. Analyzing the event with the pumps operating is consistent with ATWS analyses for standard Westinghouse PWRs that have shown the ATWS cases with loss of off-site power and reactor coolant pump coastdown to be less limiting than those that maintain forced primary system flow. If modeled, a reactor coolant pump trip produces additional heatup and the resulting feedback generates additional negative reactivity. As part of this AP600 ATWS analysis effort, a case with CMT actuation and RCP trip modeled was performed to confirm this assertion.

Results

The ATWS case analyzed for AP600 is a complete loss of normal feedwater flow to both steam generators. The sequence of events for this analysis is presented in Table 1. The first reactor trip setpoint reached is low steam generator water level (narrow range) at 45.5 seconds. Although a reactor trip signal is generated, the ATWS scenario dictates that control rod insertion is assumed to fail. Similarly, the associated turbine trip due to the reactor trip signal generated at this time in the event is ignored due to an assumed failure in the electrical system.

Though the injection of borated water is conservatively not modeled in the analyzed case, the low steamline pressure "S" that could actuate the CMTs, is generated at 62.2 seconds. Following this signal, there is a nominal delay time of about 12 seconds before the CMT valves open and injection flow begins. A similar delay applies to the reactor coolant pump coastdown that is automatically initiated by the reactor protection system in parallel with CMT actuation. If credit for CMT actuation had been taken, boration from the CMT and RCP coastdown would have helped to minimize the pressurization.

The low steam generator water level (wide range) DAS setpoint is reached at 73.4 seconds. Turbine trip and passive residual heat removal system actuation are initiated upon receipt of this signal after delay times of 4 and 10 seconds, respectively. These two functions lead to a successful termination of the heatup transient. The peak reactor coolant system pressure is reached at 119.0 seconds. The predicted peak reactor coolant system pressure is 3198 psia.

The nuclear power transient is presented in Figure 1-1. As the reactor coolant system heatup proceeds, the negative moderator density coefficient produces a decrease in reactor power. The core power falls to a value below the rate of passive residual heat removal as shown in Figure 1-2. Eventually, core power reaches an equilibrium dictated by the combined heat removal capability of the passive residual heat removal system and the steam generators.

Maximum reactor coolant system pressure and T_{AVG} (the average of the inlet and outlet temperatures for a loop) as functions of time are presented in Figures 1-3 and 1-4, respectively. These two figures show that the reactor coolant system heatup and pressurization is terminated in conjunction with the reduction in core power.

Pressurizer pressure, water volume, and relief rate throughout the transient are shown in Figures 1-5, 1-6, and 1-7, respectively. When the pressurizer becomes water solid (97.5 seconds), the entire reactor coolant system begins to pressurize quickly. After the heatup is terminated, the pressurizer pressure is reduced below 2500 psia and the pressurizer safety valves reseat at 195.0 seconds. As the reactor coolant system continues to cooldown, the pressurizer eventually regains steam space (288 seconds).

The total reactor coolant system mass flow as a function of time is shown in Figure 1-8. For conservatism, the reactor coolant pumps are assumed to operate throughout the transient. The total mass flow is maintained above 80% of nominal throughout the transient.

In summary, the AP600 response to a postulated loss of normal feedwater ATWS event is similar to previous Westinghouse PWR designs. The MTC used in the analysis (-7.3 pcm/°F), which gives a peak reactor coolant system pressure of approximately 3200 psig, is consistent with achieving an ATWS core damage frequency well below 10^{-5} per reactor year.

Alternate Case

The DAS also provides a diverse reactor scram, which would still be available despite an electrical failure in the normal reactor protection system. To demonstrate the expected plant response to a loss of normal feedwater event coincident with a failure in the electrical part of the normal reactor trip system, an alternate case was analyzed assuming the presence of the diverse reactor scram initiated by the DAS. The same moderator temperature coefficient as the base case was maintained for the alternative case.

The results of an alternative case which assumes the presence of the diverse reactor scram are presented in Figures 2-1 through 2-8. The time sequence of events is presented in Table 2. The sequence of events is identical to the base case up until the time of diverse reactor scram (75.4 seconds), after which the transient is essentially terminated. The pressurizer never becomes water solid; thus, there is no sharp increase in pressure. The peak reactor coolant system pressure for this case is 2571 psia, which is a 627 psia benefit compared to the base case without the diverse scram. The peak pressure occurs at 64.5 seconds, 9 seconds before the DAS signal is generated. This shows that the diverse reactor scram terminates the pressure transient independent of turbine trip or passive residual heat removal system actuation and eliminates any challenge to the pressure limit.

Conclusions

The analysis results show that the AP600 produces acceptable responses to the limiting pressure ATWS event. The results of the base case are similar to typical Westinghouse PWRs; thus, the AP600 ATWS response is consistent with that considered by the staff in its formulation of the 10 CFR 50.62 design requirements for current plants. The AP600 DAS provides the AMSAC protection features mandated for Westinghouse plants by 10 CFR 50.62 plus, among other functions, a diverse reactor scram. The results of the alternative case, assuming the presence of the diverse reactor scram, demonstrate the added capability for the DAS diverse reactor scram to mitigate the consequences of an ATWS event. Given the presence of the DAS, the results of the PRA, as discussed in Subsection F.2.22 of Reference 440.26-6, show that the ATWS core damage frequency contribution for the AP600 is well below the goal of 10^{-5}. The AP600, therefore, meets the ATWS rule (10 CFR 50.62) and its ATWS core damage frequency safety goal basis.

SSAR Revision: NONE

References

440.26-1. "Simplified Passive Advanced Light Water Reactor Plant Program. AP600 Standard Safety Analysis Report," DE-AC03-90SF18495, June 26, 1992.

440.26-2. Burnett, T. W. T., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

440.26-3. Dircks, W. J., "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," SECY-83-293, USNRC, July 19, 1983.

440.26-4. WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis," August, 1974.

440.26-5. Anderson, T. M., "ATWS Submittal," Westinghouse Letter No. NS-TMA-2182 to S. H. Hanauer of the NRC, December, 1979.

440.26-6. "AP600 Probabilistic Risk Assessment," Contract No. DE-ACO3-90SF18495, June 26, 1992.

440.26-7. "ASME Boiler & Pressure Vessel Code, and American National Standard," ACI Standard 359-83, Section III, Division 1, Subsection NB-3224, July, 1980.

440.26-8. ANSI/ANS-5.1-1979, August 1979, "American National Standard for Decay Heat Power in Light Water Reactors."

440.26-9. WCAP-11992, "Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," December, 1988.

440.26-10. WCAP-11993, "Joint Westinghouse Owners Group/Westinghouse Program: Assessment of Compliance With ATWS Rule Basis for Westinghouse PWRs," December, 1988.

Table 1

Time Sequence of Events for the Loss of Normal Feedwater Anticipated Transient Without Scram Event

Base Case

Event	Time (s)
Main feedwater supply to all steam generators is terminated	0-4
Low steam generator water level (narrow range) reactor trip setpoint reached (failure of RCCA insertion assumed)	45.5
Pressurizer safety valves open	60.5
Low steamline pressure "S" setpoint reached (signal ignored in analysis)	61.7
Low steam generator water level (wide range) DAS setpoint reached	73.4
CMT actuation on low steamline pressure "S" signal conservatively not modeled	73.7
Steam generator tube uncovery	74.5
Turbine trip assumed to occur on DAS generated signal	77.4
Passive residual heat exchanger valves opened	83.4
Pressurizer fills with water	97.5
Peak RCS pressure is reached (3198 psia)	119.0
Steam generator dryout	148.0
Pressurizer safety valves reseat	195.0
Pressurizer regains steam space	288.0

Table 2

Time Sequence of Events for the Loss of Normal Feedwater Anticipated Transient Without Scram Event

Alternate Case: Diverse Reactor Scram Assumed

Event	Time (s)
Main feedwater supply to all steam generators is terminated	0-4
Low steam generator water level (narrow range) reactor trip setpoint reached (failure of RCCA insertion assumed)	45.5
Pressurizer safety valves open	60.5
Low steamline pressure "S" setpoint reached (signal ignored in analysis)	61.7
Peak RCS pressure is reached (2571 psia)	64.5
Low steam generator water level (wide range) DAS setpoint reached	73.4
CMT actuation on low steamline pressure "S" signal conservatively not modeled	73.7
Diverse reactor scram assumed to occur on DAS signal	75.4
Turbine trip assumed to occur on DAS generated signal	77.4
Pressurizer safety valves reseat	79.5
Passive residual heat heat exchanger valves opened	83.4

Figure 1-1
AP600 LONF ATWS: Base Case
Nuclear Power

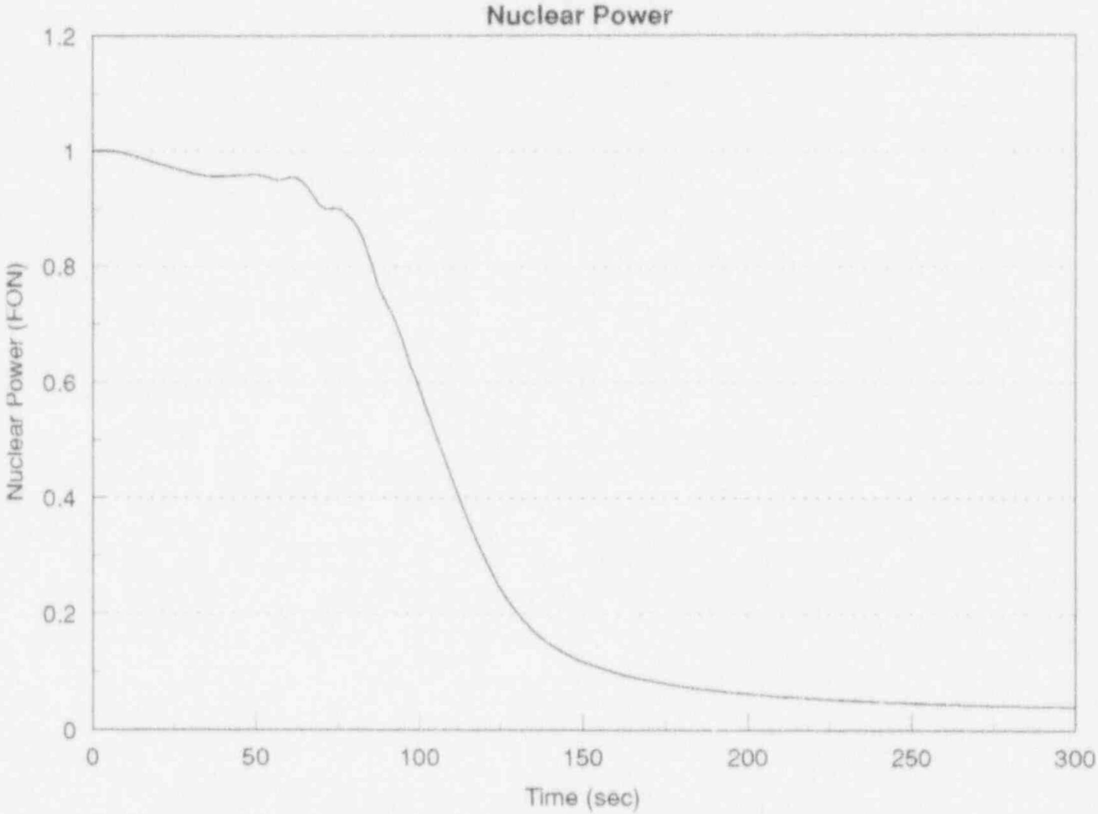


Figure 1-3
AP600 LONF ATWS: Base Case
Maximum RCS Pressure

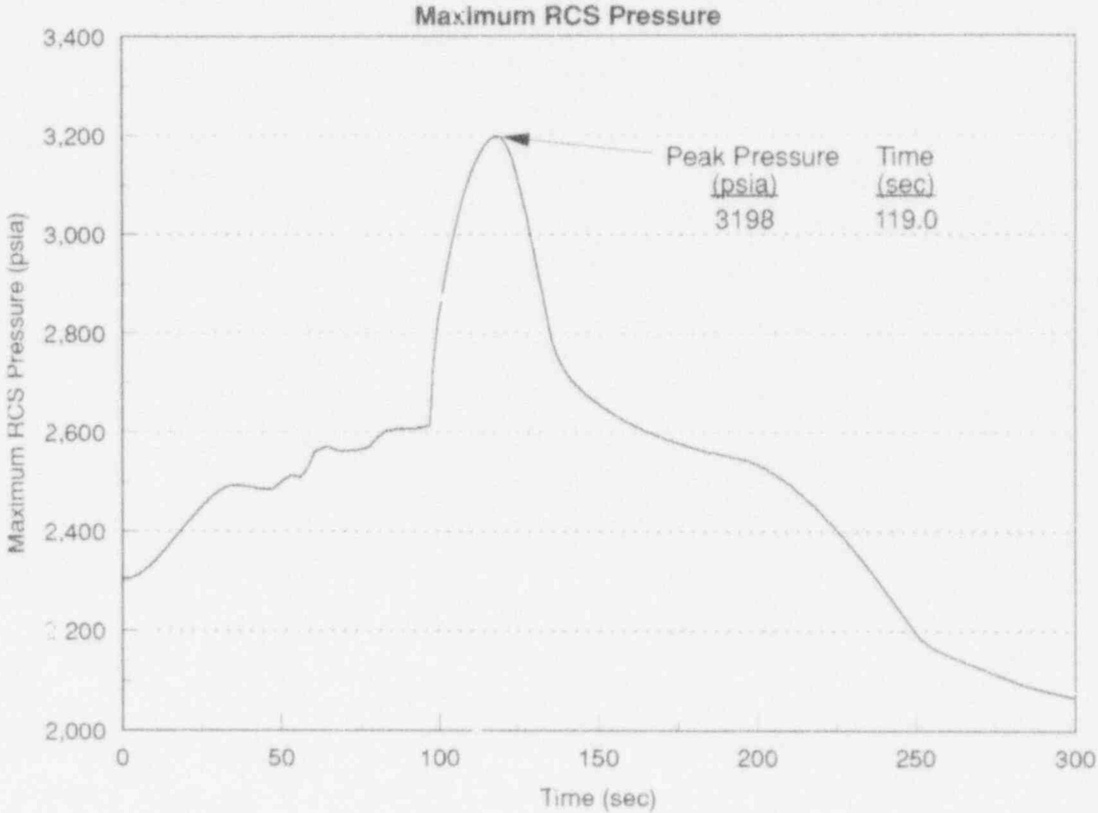


Figure 1-4
AP600 LONF ATWS: Base Case
Tavg

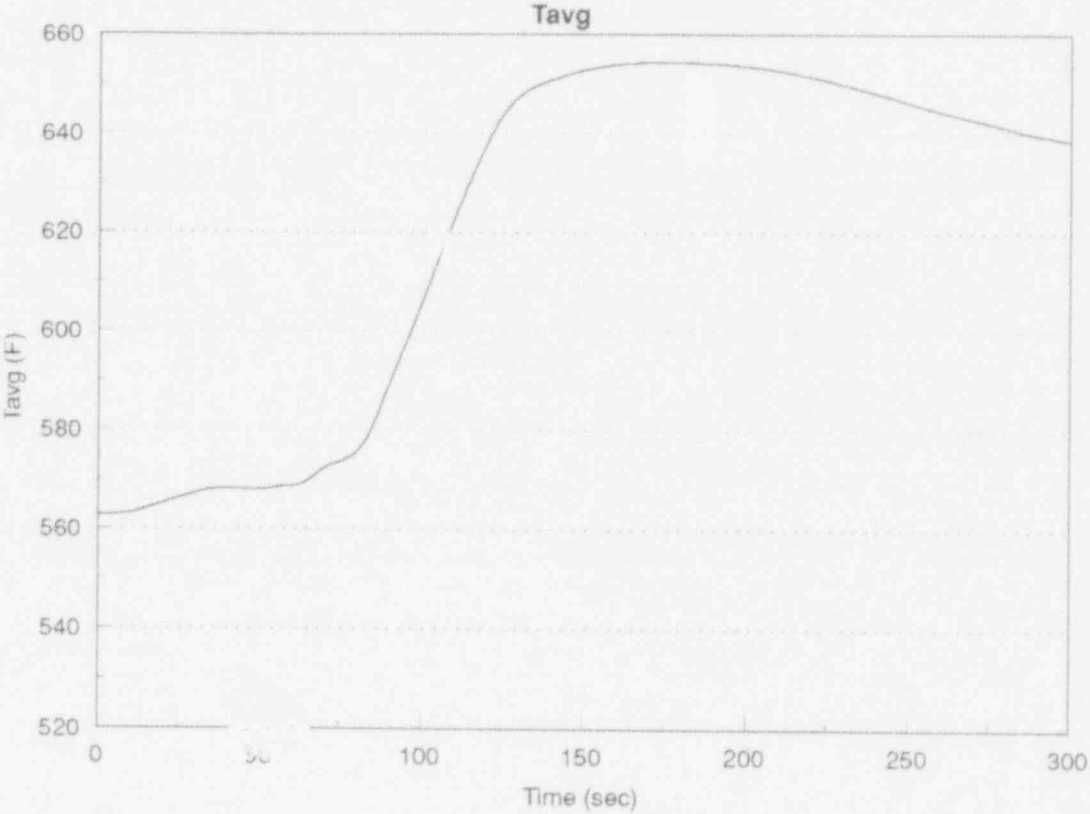


Figure 1-5
AP600 LONF ATWS: Base Case
Pressurizer Pressure

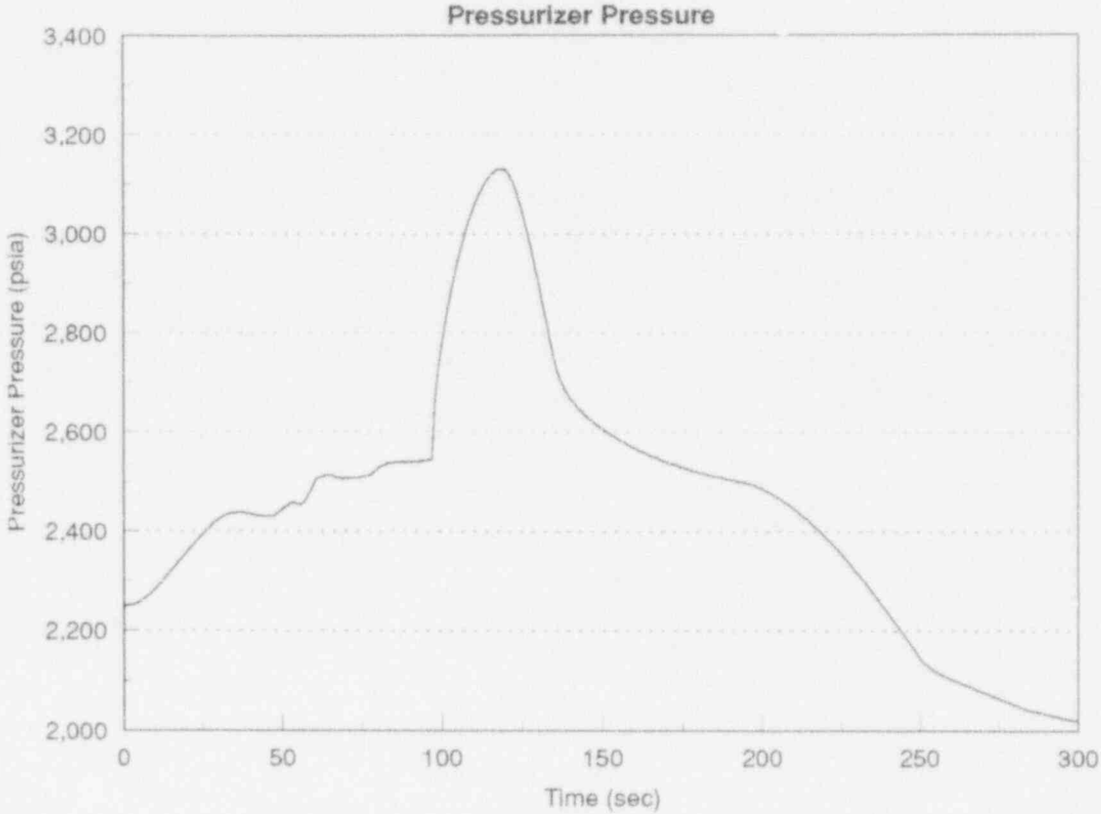


Figure 1-6
AP600 LONF ATWS: Base Case
Pressurizer Water Volume

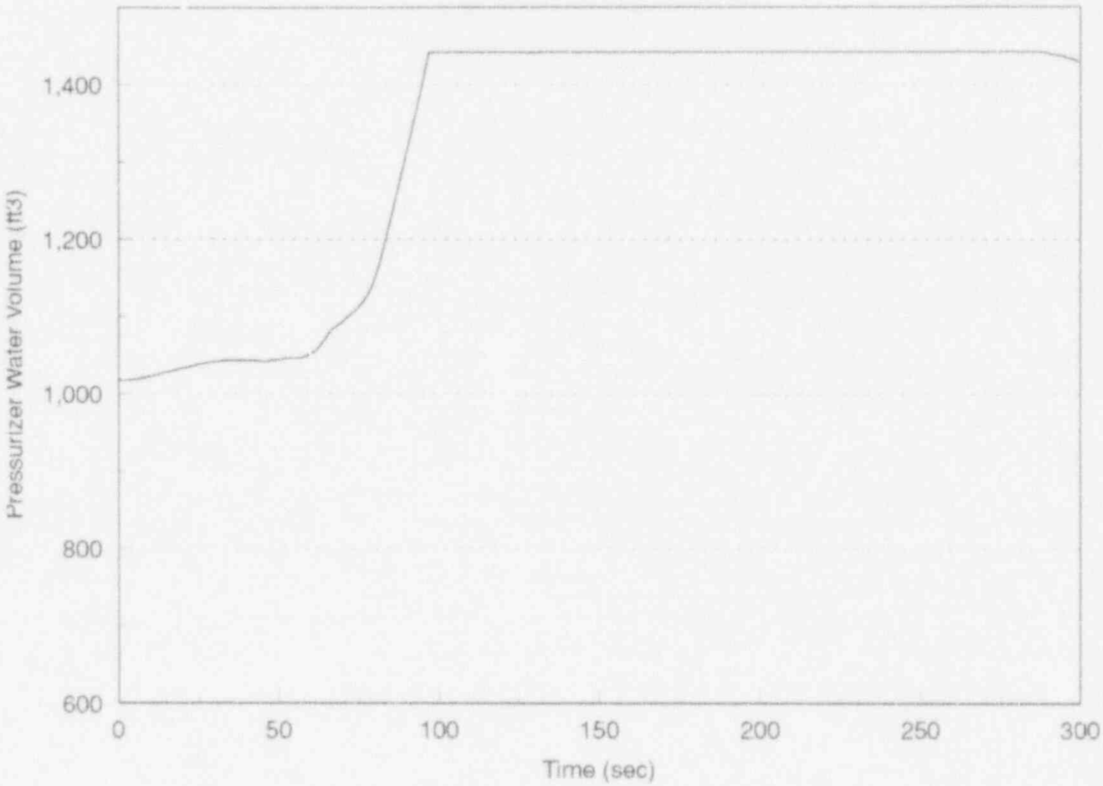


Figure 1-7
AP600 LONF ATWS: Base Case
Pressurizer Steam and Water Relief

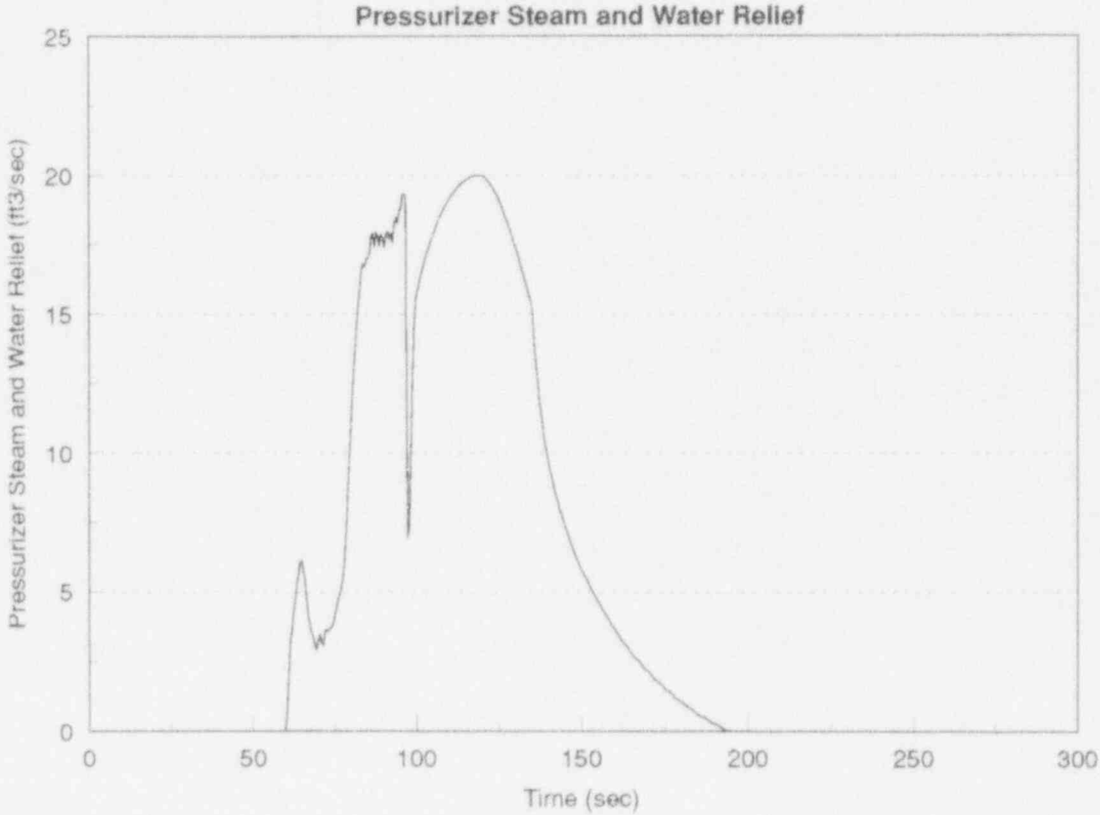


Figure 1-8
AP600 LONF ATWS: Base Case
Core Coolant Mass Flow

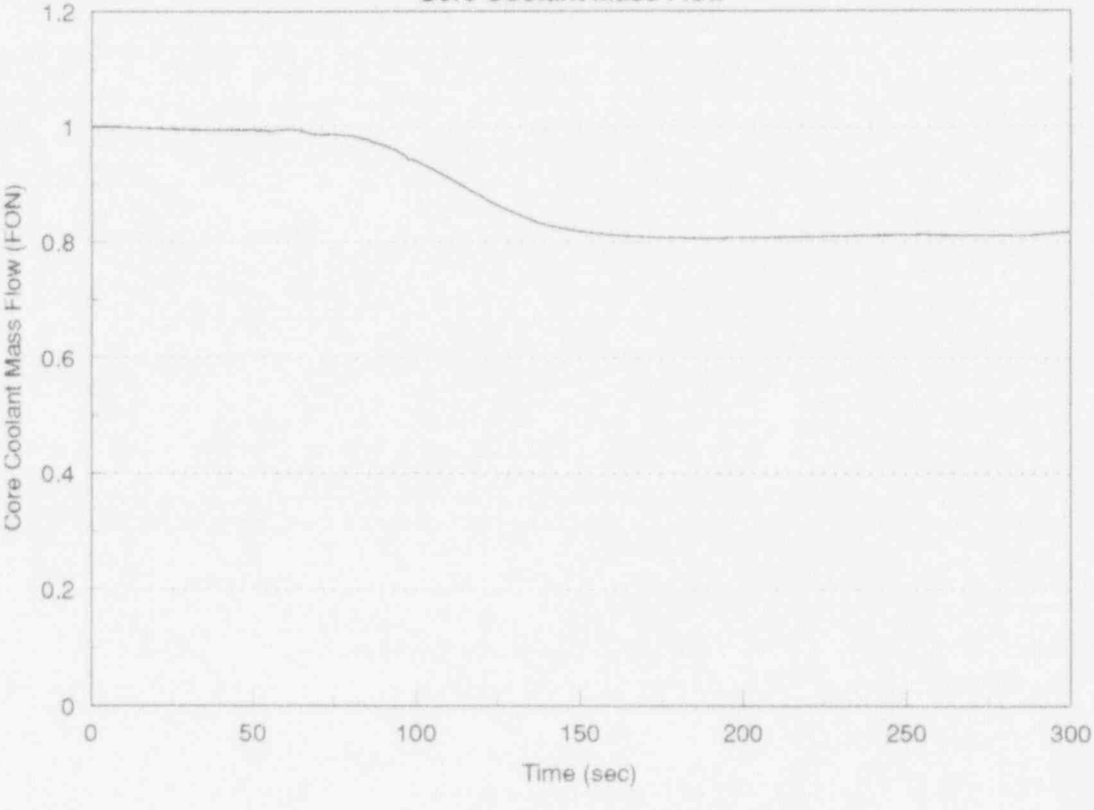


Figure 2-1
AP600 LONF ATWS: Diverse Scram Assumed
Nuclear Power

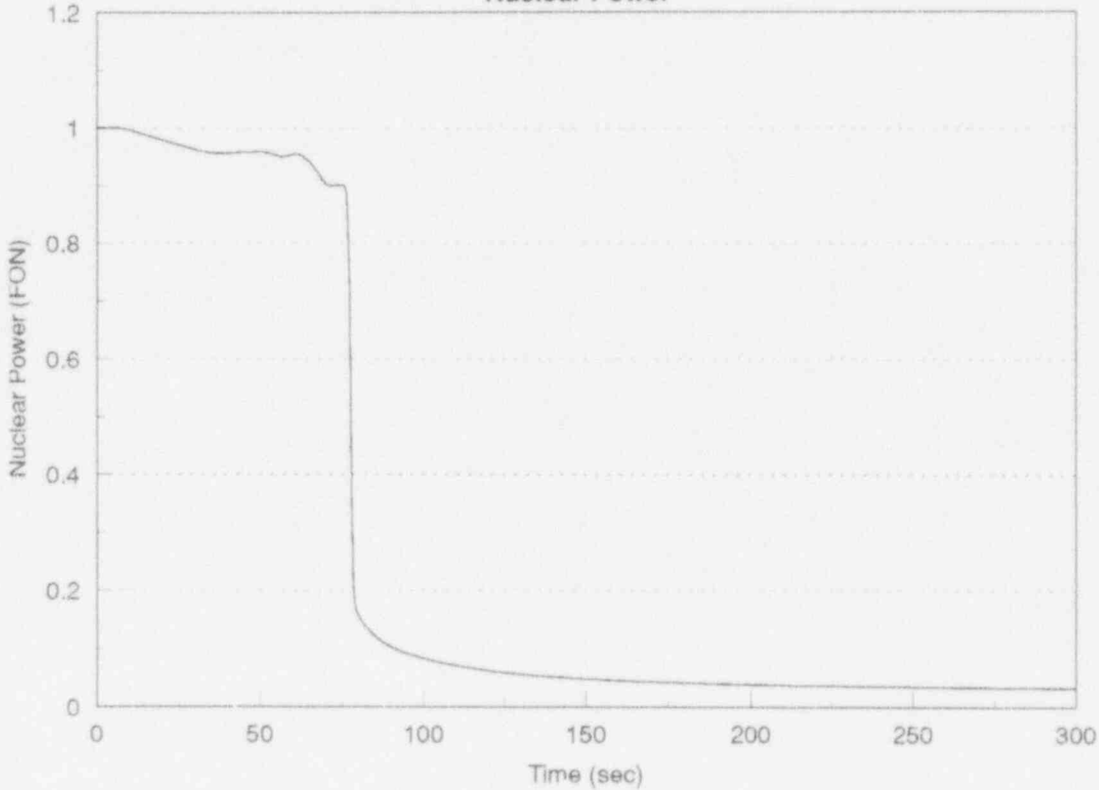


Figure 2-2
AP600 LONF ATWS: Diverse Scram Assumed
Core Heat Flux and PRHR Heat Removal

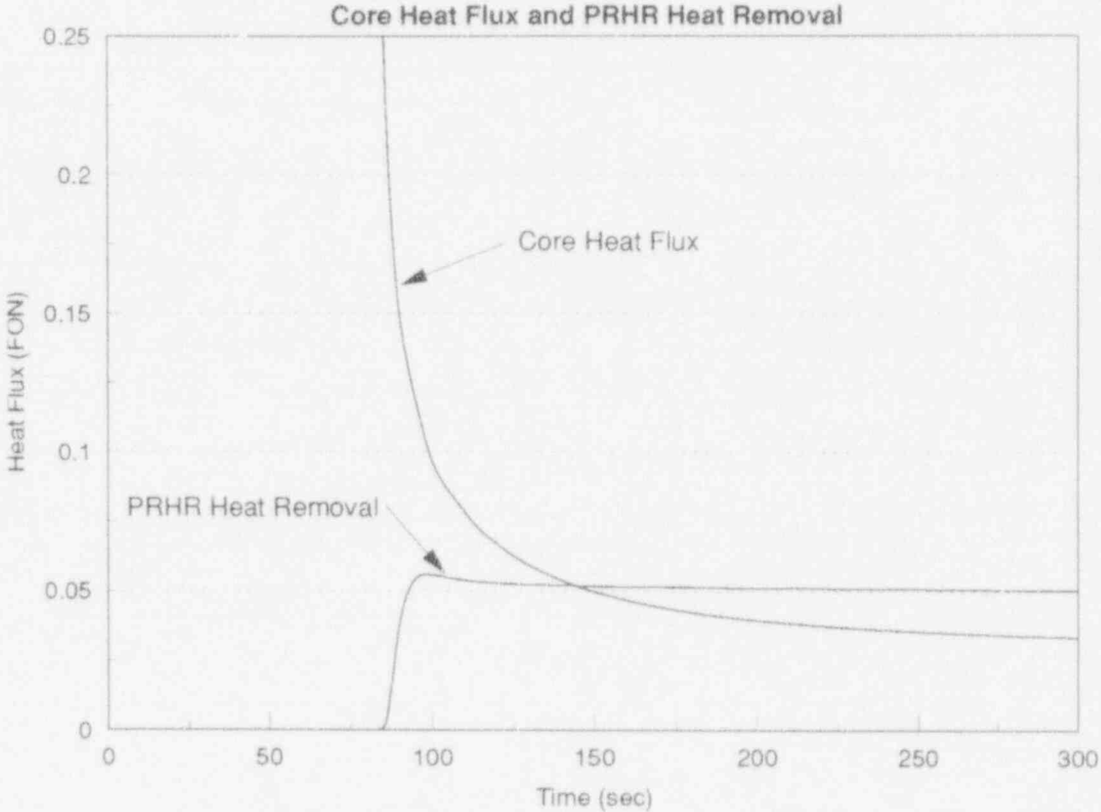


Figure 2-3
AP600 LONF ATWS: Diverse Scram Assumed
Maximum RCS Pressure

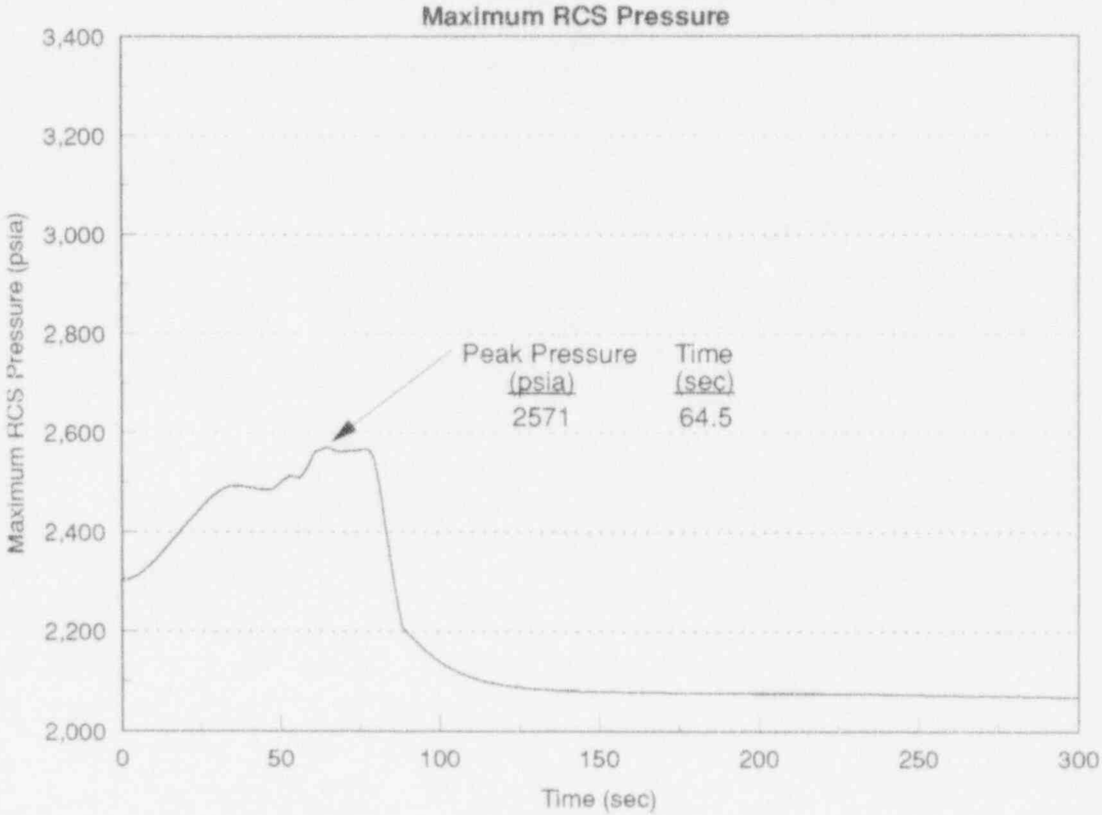


Figure 2-4
AP600 LONF ATWS: Diverse Scram Assumed
Tavg

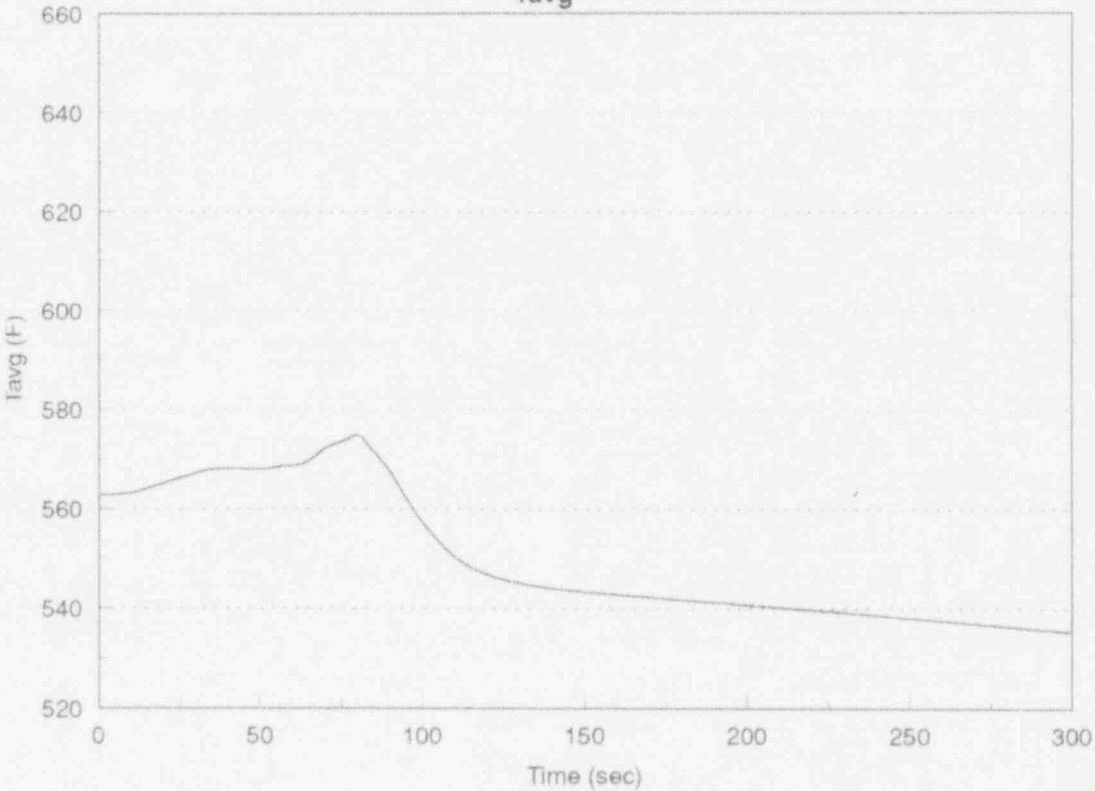


Figure 2-5
AP600 LONF ATWS: Diverse Scram Assumed
Pressurizer Pressure

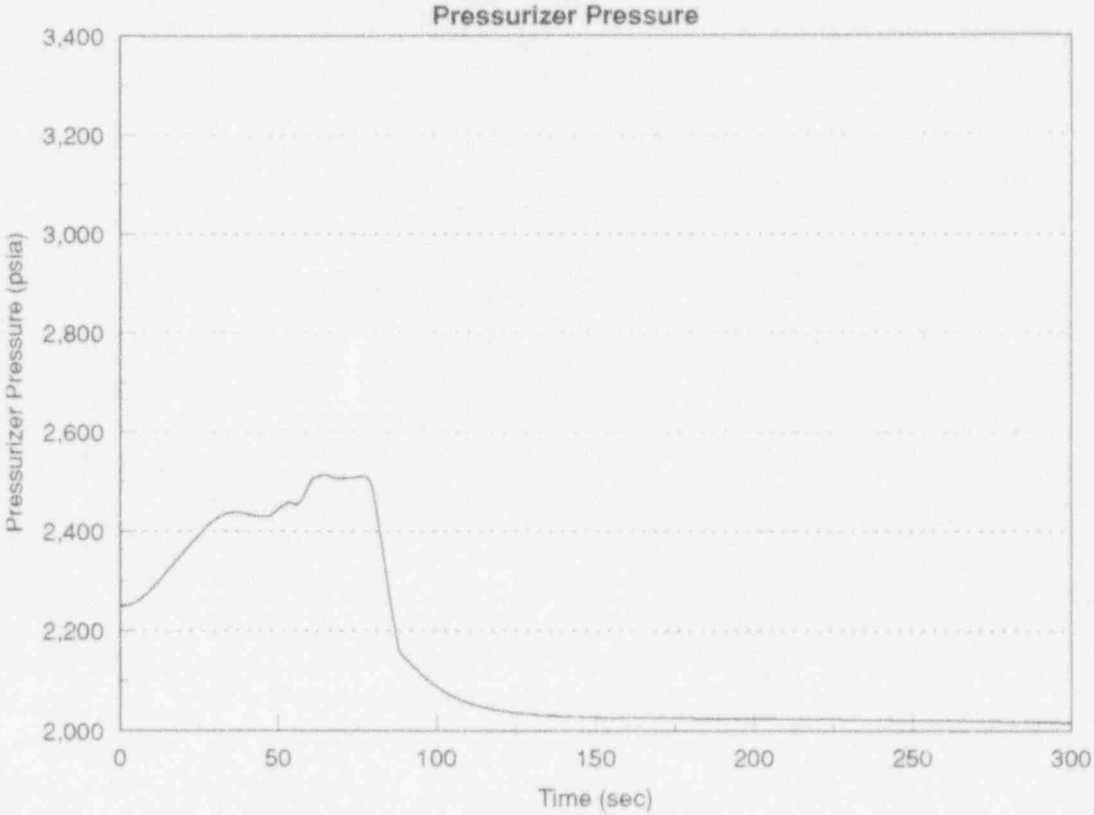


Figure 2-6
AP600 LONF ATWS: Diverse Scram Assumed
Pressurizer Water Volume

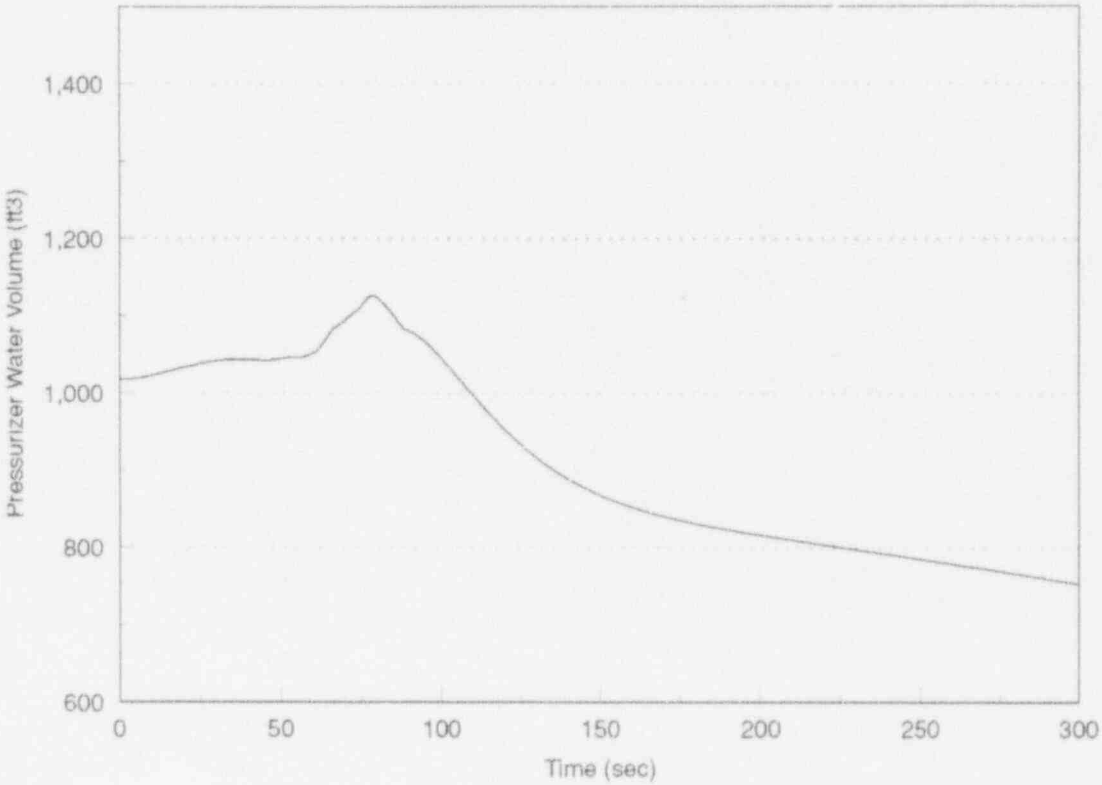


Figure 2-7
AP600 LONF ATWS: Diverse Scram Assumed
Pressurizer Steam and Water Relief

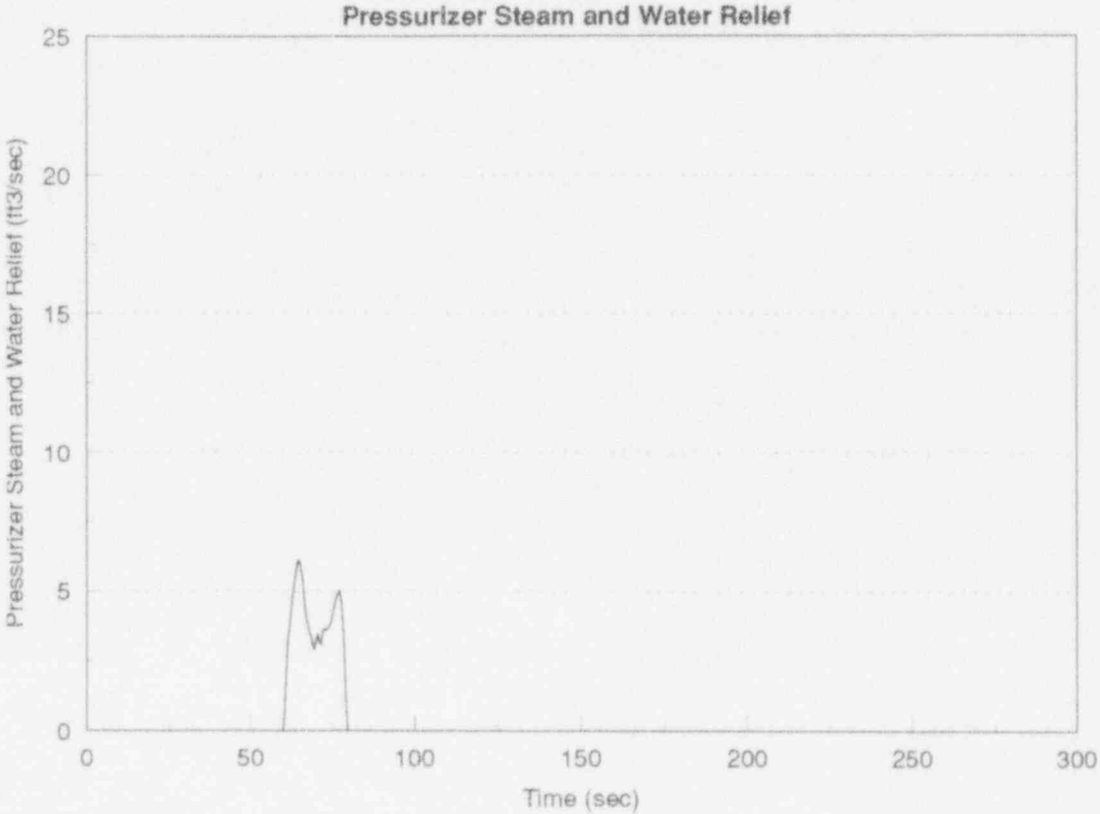
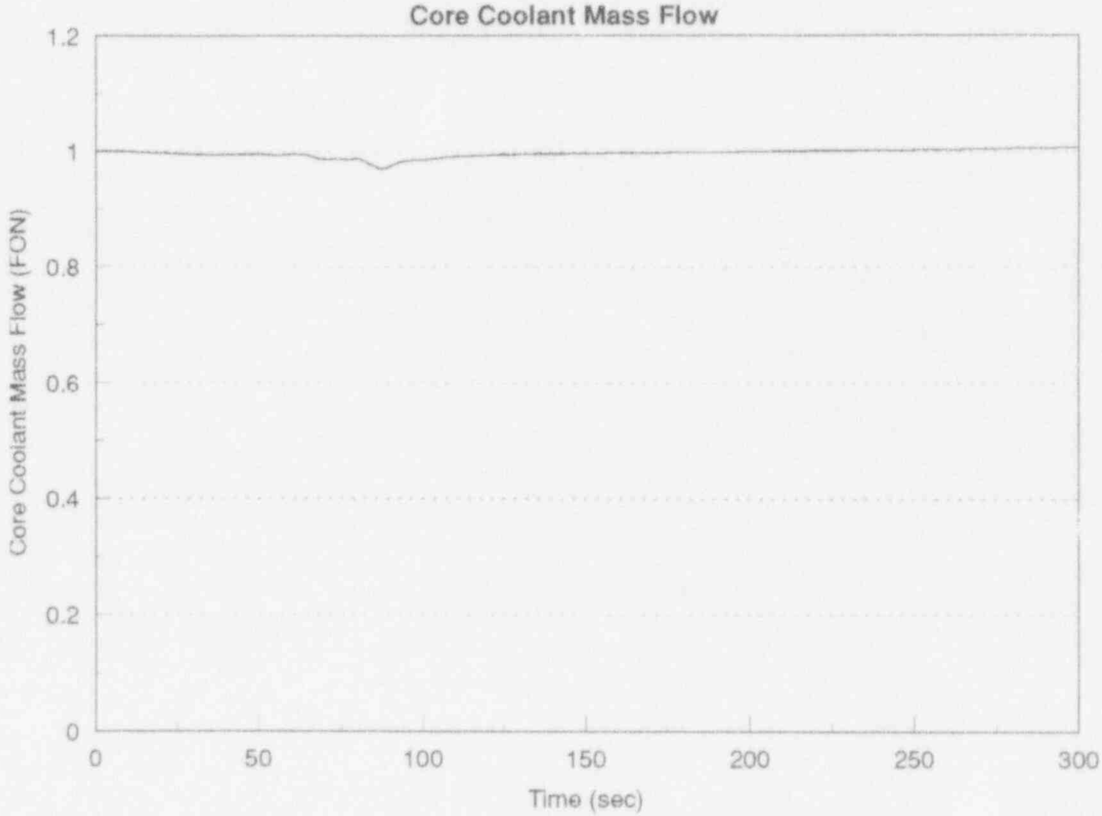


Figure 2-8
AP600 LONF ATWS: Diverse Scram Assumed
Core Coolant Mass Flow



Question 440.35

Chapter 15 of the SSAR references the NOTRUMP small break LOCA model and the LOFTRAN transient analysis model as the methodologies that were used to evaluate small break LOCA and transient events for the AP600. To facilitate its review of Chapter 15, the staff was provided with documentation describing contemporary approved versions of these methodologies. This documentation does not describe model features that address design and behavior phenomena specific to the AP600 design. Provide information that describes the AP600-specific versions of LOFTRAN and NOTRUMP. Identify features that differentiate the AP600-specific versions of these codes from the versions approved for the analysis of conventional Westinghouse designs. Include a comprehensive discussion of the Westinghouse provisions for code qualification of these methodologies for application to the AP600 design, identifying specific methods of qualification and schedules for completion.

Response (Revision 1):

I. NOTRUMP

The NOTRUMP small break LOCA analyses presented in the AP600 SSAR use the approved Appendix K Evaluation Model nodalization of the reactor coolant system (SSAR Reference 15.6.6-13). No AP600 specific coding or logic was created to represent the AP600 passive systems other than introducing the signals that actuate the passive safeguards system components per the plant design. Further, choked flow through the automatic depressurization system is modeled based on the Henry-Fauske model merging into the homogeneous equilibrium model at 10% quality. Otherwise, the documented, existing NOTRUMP fluid node, flow link, metal node and heat transfer models were applied in the modeling of the passive safety systems.

Appendix 15D of the AP600 SSAR provides the detailed AP600 system nodalization for the NOTRUMP code. Subsequent to the performance of the SSAR analysis, two enhancements have been made to the SSAR NOTRUMP model to improve the code's predictive capability. These changes have been applied in the SPES facility NOTRUMP modeling studies. Further enhancements may also be implemented if they lead to better predictions of observed test behaviors (Ref. 440.35-1).

The first change is to employ an eight-node model of the passive residual heat removal (PRHR) heat exchanger. In the SSAR cases the PRHR is activated on a first stage ADS signal; it does not play a major role in RCS depressurization, so a simpler model is adequate. However, for cases in which the PRHR is activated well in advance of the ADS first stage, its heat removal capability can play a significant role in the transient. To obtain a better prediction of the PRHR performance an eight-node model has been derived which utilizes five horizontal tube nodes at the top of the heat exchanger, followed by two vertical nodes and a final node for the horizontal exit section. The PRHR inlet is nodalized in a fine mesh to obtain a good simulation of the heat transfer in the tube section where the temperature difference and condensation potential are at their maxima. Separate heat transfer correlations are applied to define the primary side condensation heat transfer in the horizontal and vertical tube nodes, as specified in Reference 440.35-2.

The second change is to eliminate the surge line entry/exit horizontal stratified flow link connections used in SSAR analysis. While the horizontal stratified flow link nodalization leads to an overly conservative core uncovery prediction in the SSAR, it also leads to non-physical draining and pressure behavior in the pressurizer. To improve the prediction of pressurizer phenomena these links were replaced throughout the transients with single flow links. The hot leg/surge line flow link is now a continuous contact flow link. A point contact flow link connects the bottom of the pressurizer vertically at the surge line entrance.

II. LOFTRAN & LOFTTR2

The LOFTRAN code and LOFTTR2 code were modified to model the AP600 by the addition of specific models for the passive residual heat removal heat exchanger and for the core makeup tank (CMT). Otherwise, the NRC-approved version of the LOFTRAN computer code (SSAR Reference 15.5.4-1) and associated modeling techniques have been applied. The PRHR and CMT models in both codes are identical. Reference 440.35-3 provides a detailed description of the AP600 passive residual heat removal model and the core makeup tank model which have been added to LOFTRAN and LOFTTR2. This information will also be included in the LOFTRAN and LOFTTR2 code validation report to be submitted after completion of the SPES facility tests.

III. CODE QUALIFICATION PLANS

Qualification of the AP600 NOTRUMP and LOFTRAN/LOFTTR2 modeling will be accomplished by prediction of the pertinent test facility results. Further information concerning the qualification plans for both codes is provided in Reference 440.35-1.

Initially the core makeup tank models in NOTRUMP and LOFTRAN/LOFTTR2 will be refined and validated based upon the core makeup tank component test results. The refined CMT model will then be confirmed in the simulation of integral systems tests in the SPES facility and the Oregon State University low pressure facility.

In using NOTRUMP to predict the integral and ADS facility tests, adjustments will be made if needed to model the pertinent physical phenomena on a system basis before the blind test predictions are performed. The code qualification as well as the "single blind" predictions of the designated SPES and Oregon State tests will be submitted to the NRC in the code validation report after the completion of the Oregon State tests.

References:

15.6.6-13 Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-P-A (Non-Proprietary), August 1985.

15.5.4-1 Burnett, T.W.T. et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), April 1984.

440.35-1 Letter ET-NRC-93-3976, N. J. Liparulo to the NRC, September, 1993.

440.35-2 Holman, "Heat Transfer", McGraw-Hill Book Company, 1986, p.498 and p.494.

440.35-3 Letter NTD-NRC-94-4047, N. J. Liparulo to the NRC, February, 1994.

SSAR Revision: NONE

PRA Revision: NONE