DCP/NRC1558

March 28, 2003

Attachment 3

Westinghouse Revised Non-Proprietary Responses
to US Nuclear Regulatory Commission
Requests for Additional Information
dated March 2003

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.022 (Response Revision 1)

Question:

Section 4.4.1.1.2 states that for those transients that use the VIPRE-W computer program and the WRB-2M correlation, the Revised Thermal Design Procedure (RTDP) design limits are 1.25 for the typical cell and 1.25 for the thimble cell for Core and Axial Offset Limits, and 1.22 for the typical cell and 1.21 for the thimble cell for all other RTDP transients, and that these values may be revised when plant specific uncertainties are available.

A. Discuss the differences between the RTDP design departure from nucleate boiling ratio (DNBR) limits for (1) core and axial offset limits, and (2) other RTDP transients, respectively.

B. Provide the derivations of these RTDP design DNBR limits, including the uncertainties of all parameters used in the derivation.

C. Provide the instrument uncertainty methodology and the assumed uncertainty values of various components of the instrument for the measurements of the parameters included in the RTDP.

Westinghouse Response:

A. RTDP Design Limits are calculated using parameter uncertainties and DNBR sensitivities to these parameters for a number of conditions as illustrated in WCAP-11397-P-A, "Revised Thermal Design Procedure." The magnitude of the DNBR sensitivities to the various parameters is dependent upon the conditions analyzed. The calculations that are associated with Core Limit conditions gave higher DNBR Design Limits than those associated with the other RTDP conditions. To maximize margin, separate DNBR Design Limits are used for each set of conditions.

B. The calculation of the RTDP DNBR Design Limits follows that illustrated in WCAP-11397-P-A, "Revised Thermal Design Procedure." The following values were used for sigma: Power = 1.0%, Tin = 3 degrees F, Pressure = 30 psi, Flow = 1.25%, Bypass = 0.5%, FdH = 0.0386, FdHE1 = 0.0182, VIPRE code = 0.02, Transient code = 0.005. The values for Power, Tin, Pressure and Flow were assumed since the plant instrumentation to measure these has not been detailed. These are typical bounding values. The calculations will be revised when the plant is built. Experience has shown that any changes in these parameters are expected to have a minor impact (less than 1%) on the design limits. The value for FdH is based on a 4% uncertainty and a FdH value equal to 1.587. These DNBR Design Limits were based on the WRB-2M DNB correlation which was based on 241 data points with a mean of 1.0008 and a sample standard deviation of 0.0652.

C. The instrumentation uncertainty methodology will be similar to that used for AP600 in WCAP-14605, "Westinghouse Setpoint Methodology for Protection Systems", April 1996.

The assumed 2-sigma instrumentation uncertainties associated with the sigmas in (B) above are: Power = 2.0%, Tin = 6 degrees F, Pressure = 60 psi and Flow = 2.5%.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

NRC Additional Comments:

A. The response to Item A stated that the magnitude of the DNBR sensitivities to the various parameters is dependent upon the conditions analyzed, and the calculations that are associated with the Core Limit conditions gave higher DNBR design limits than those associated with the other RTDP conditions.

Provide the sensitivity factor values of each parameter included in the RTDP process for the core limit conditions and the other RTDP conditions.

B. In the response to item B, Westinghouse provided the sigma values for the uncertainties of various parameters. Westinghouse did not provide the derivations of the design DNBR limits, nor the sensitivity factors of the parameters. Using the sensitivity factors in the sensitivity factor values of the sample calculation in Table 3-1 of WCAP-11397-P-A, and the sigma values Westinghouse provided, the RTDP design DNBR limits would be 1.226 and 1.215 for the typical cell and thimble cell, respectively, compared to 1.22 and 1.21, respectively, stated in the DCD.

Are the sensitivity factors in WCAP-11397 sample calculations used in the AP1000 RTDP design DNBR limits calculations? What are the bases for using these sensitivity factors, which were based on the THINC-IV code and the WRB-1 correlation, to the AP1000 design using the VIPRE-01 code and the WRB-2M correlation?

C. The staff SER approving WCAP-11397 included seven restrictions, e.g., inclusion of sensitivity factors in the safety analysis report, and re-evaluation of the sensitivity factors for any changes in DNB correlation, thermal-hydraulic code, or parameter values, etc.

Provide your evaluation that the use of RTDP process for the AP1000 design comply with the seven restrictions in the WCAP-11397 SER.

D. Westinghouse stated that the uncertainties values for power, T_{in}, pressure, and flow were assumed typical bounding values, and that the calculations will be revised when the plant is built.

Would this be a COL interface item? If the revised calculation results in higher design DNBR limits, will it invalidate the Chapter 15 safety analyses? What are the safety analysis DNBR limits used for the AP1000?

E. The response to Item C stated that the instrumentation uncertainty methodology will be similar to that used for the AP600 in WCAP-14605.

Provide reference to the methodology used for AP1000. Has it been approved?

Westinghouse Additional Response:

A. The AP1000 sensitivity factor values of each parameter included in the RTDP process and the derivation of the design DNBR limits are listed in Tables 440.022R1-1 and 440.022R1-2 for the core limit conditions and the non-core limit conditions, respectively. The RTDP sensitivity factors were calculated specifically for AP1000 at different DNB limiting conditions, using the VIPRE-01 code and the WRB-2M DNB correlation. The DNBR limit is the highest value obtained from a wide set of the sensitivity factors.

B. See the response to Item A above.

C. The RTDP application for the AP1000 design is in full compliance with the seven conditions from the SER on WCAP-11397. Each of the seven conditions was considered in the safety evaluation and is addressed below, and a reference is provided to the specific section in the safety evaluation or other appropriate documentation where the condition is discussed.

SER Condition 1:

Sensitivity factors for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.

Response:

Sensitivity factors were evaluated using the WRB-2M DNB correlation and the VIPRE-01 code for parameter values applicable to AP1000 core conditions. These sensitivity factors are proprietary information and, therefore, are not included in the Safety Analysis Report. The AP1000 sensitivity factors are provided in the response to Item (B).

SER Condition 2:

Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397 outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.

Response:

See Responses to Condition 1 and Condition 3. The sensitivity factors have been evaluated and the use of Equation 2-3 of WCAP-11397 has been justified for application to AP1000 core design.

SER Condition 3:

If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be re-evaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.

Response:

Equation (2-3) of WCAP-11397-P-A and the linearity approximation made to obtain Equation (2-17) have been shown to be valid for the combination of WRB-2M and the VIPRE-01 code which was used for the application of RTDP to AP1000 core design. The sensitivity factors, operating parameters, and the VIPRE-01 model used in this application do not differ significantly from those used in WCAP-11397-P-A.

SER Condition 4:

Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained.

Response:

The plant specific variances and distributions for this application are justified for AP1000 as discussed in response to Items (D) and (E).

SER Condition 5:

Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions.

Response:

Nominal initial conditions were only applied to DNBR analyses that used RTDP.

SER Condition 6:

Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.

Response:

Bounding nominal conditions were used in the DNBR analyses that were based on RTDP.

SER Condition 7:

The code uncertainties specified in Table 3-1 (± 4 percent for THINC-IV and ± 1 percent for transients) must be included in the DNBR analyses using RTDP.

Response:

The same code uncertainties specified in Table 3-1 of WCAP-11397-P-A were included in the DNBR analyses using RTDP.

D. The core thermal limits in Safety Limit 2.1.1 and the DNB parameters in LCO 3.4.1of the AP1000 Technical Specifications identify limits for power, temperature, pressure, and flow. The Technical Specification limit values for SL 2.1.1 and LCO 3.4.1 are contained in the Core Operating Limits Report (COLR) since these parameters may vary for each fuel cycle. The limits specified in the COLR are a combination of the values assumed in the safety analyses and the associated instrumentation uncertainties for each parameter. As stated in DCD section 7.1.6, the COL Information Item specification of these uncertainties is established as part of the required calculation to determine the associated Technical Specification setpoints, as discussed in Item E.

Based on experience, the instrumentation uncertainties identified in Item B of the original response are expected to be typical values that bound both the specified and delivered uncertainties for the plant instrumentation. In the unlikely event that the assumed uncertainty values in Item B are exceeded when the plant is built, the calculated COLR limits could be increased slightly to accommodate any additional uncertainties for the installed instrumentation beyond the originally assumed uncertainty values.

If the revised COLR limit results in any unacceptable plant limitations, there are several options. The COL applicant can attempt to modify or install different instrumentation that can meet the originally assumed uncertainties. Another option is to analytically offset any additional instrumentation uncertainty by reallocating the available margin from existing overconservatism in the safety analyses. The third option is to re-perform any safety analyses that would be affected by the increased instrumentation uncertainty resulting in a change to the calculated limiting DNBR parameters. However, it is unlikely that any of these options would be required since the originally assumed instrumentation uncertainties are expected to be bounding.

E. DCD Section 7.1.6 references the COL Information Item to use WCAP-14605 as the methodology that will be used to calculate the setpoints for AP1000 protective functions. Section 7.2.7 of the NUREG-1512, "Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design," states that "WCAP-14605 provides sufficient information on instrument setpoints for the COL applicant to establish setpoints for plant-specific equipment, and therefore, is acceptable."

See the response to RAI 440.103 for a discussion of the approach for identifying the allowable values and the trip setpoints for AP1000 Technical Specifications Tables 3.3.1-1 (Reactor Trip System Instrumentation Set Points) and 3.3.2-1 (Engineered Safeguards Actuation System Instrumentation Set Points).

Measurement uncertainties for the trip and ESFAS instrumentation, as well as for other Technical Specification limits where instrumentation uncertainties affect the limits, cannot be determined until the plant-specific setpoint calculation is completed by the COL applicant, once the actual instrumentation has been selected for the plant.

The AP1000 Technical Specification Trip Setpoint values specified in brackets must be replaced, following the plant-specific setpoint study, with the actual Trip Setpoints. In addition, some COLR values are also calculated as part of the setpoint study. Upon selection of the plant specific instrumentation, the Setpoints and COLR values will be calculated in accordance with the setpoint methodology described in WCAP-14605 (P) and WCAP-14606 (NP), "Westinghouse Setpoint Methodology for Protection Systems, AP600," April 1996.

WCAP-14605 and WCAP-14606 are AP600 documents that describe the methodology, which is also applicable to AP1000, as discussed in DCD 7.1.6. AP1000 has some slight differences in instrument spans as a result of the higher power level. The plant-specific setpoint calculations will reflect the latest licensing analysis/design basis and may incorporate NRC-accepted improvements in setpoint methodology.

Table 440.022 R1-1
AP1000 RTDP Sensitivity Factors and WRB-2M Design Limit DNBR
for Core Limit Conditions

Parameter	Nominal or Best Estimate (μ)	Standard Deviation (σ)	σ/μ	Typical Cell Sensitivity (S)	$S^2(\sigma/\mu)^2$	Thimble Cell Sensitivity (S)	$S^2(\sigma/\mu)^2$
Power (Fraction)							
Tin (°F)							
Pressure (psia)							
RCS Flow (Fraction)							
Core Flow - Bypass (Fraction)							
$F^N_{\Delta H}$							
$F^E_{\Delta H,1}$							
Subchannel Code							
Transient Code							

a,c

a,c

Table 440.022 R1-2
AP1000 RTDP Sensitivity Factors and WRB-2M Design Limit DNBR
for Non-Core Limit Conditions

Parameter	Nominal or Best Estimate (μ)	Standard Deviation (σ)	σ/μ	Typical Cell Sensitivity (S)	$S^2(\sigma/\mu)^2$	Thimble Cell Sensitivity (S)	$S^2(\sigma/\mu)^2$
Power (Fraction)							
Tin (°F)							
Pressure (psia)							
RCS Flow (Fraction)							
Core Flow - Bypass (Fraction)							
$F^N_{\Delta H}$							
$F^E_{\Delta H,1}$							
Subchannel Code							
Transient Code							

a,c

a,c

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.001 (Response Revision 1)

Revision 0 Question:

Reference, Volume 6, Section 3.9.2.3, Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions, Pg. 3.9-31, last paragraph:

Westinghouse (W) proposes that the assessment of RPV internals flow-induced vibrational response is done using a combination of analysis and testing, as specified in Regulatory Guide (R.G.) 1.20. However, W also proposes that the entire vibration assessment program, including the predictive analysis portion, will be performed by the Combined License (COL) applicant. This proposal is repeated in DCD Section 3.9.8.1 (Volume 6, Pg. 3.9-93) citing consistency with R.G. 1.20 as a basis for deferral of the performance of the entire vibration assessment program to the COL applicant.

The NRC staff is not in complete agreement with this proposal for the following reasons. Title 10 of the Code of Federal Regulations (10 CFR) Part 52.47(a)(2) requires that applications for standard design certification must contain a level of design information sufficient to enable the Commission to reach a final conclusion on all safety questions associated with the design before the certification is granted. Delaying the predictive analysis portion of the vibration assessment program to the COL applicant stage of plant construction does not provide the staff with a level of design information sufficient to reach a final conclusion regarding adequacy of the RPV internals design. Conformance with R.G. 1.20 alone, does not necessarily fulfill the requirements of 10 CFR 52.47(a)(2) for certification of the adequacy of the standard design of the RPV internals, primarily because the R.G. 1.20 scheduling requirements for the submittal of analytical results to the staff occurs much too late to support the standard design certification process.

The staff's position on this issue is that the detailed, predictive analysis portion of the RPV internals flow-induced vibration analysis program should be provided for staff review during the design certification process, and not be deferred to the COL applicant stage of actual plant construction. It is recognized that the other phases of the comprehensive R.G. 1.20 vibration assessment program, i.e., vibration measurement and physical inspection, must be done later by the COL applicant to confirm the predictive analysis results. However, the staff considers the results of the predictive analysis phase of this program to be the kind of detailed information necessary for the staff to make a determination of adequacy of the AP1000 RPV internals design for purposes of final design certification.

Please provide technical documentation of the predictive analysis phase of the vibration assessment program. The technical details should be provided with descriptions of the analytical methods used including computer models, results of the analyses summarized in tabular format, and comparisons of calculated stresses to ASME Code allowables for the major components of the RPV internals design. An example of the presentation of the type of analytical data requested is W topical report WCAP-14761, "AP600 Reactor Internals Flow-Induced Vibration Assessment Program." This type of topical report would also be appropriate for presentation of key details of the AP1000 prototype RPV internals design necessary for staff review at the standard design certification stage.

Revision 0 Westinghouse Response:

A flow-induced vibration assessment of the AP1000 reactor vessel internals will be performed and the results will be documented in an AP1000 Vibration Assessment Report which will be provided to the NRC to support AP1000 Design Certification. This report will be similar in scope to WCAP-14761, "AP600 Reactor Internals Flow-Induced Vibration Assessment Program". The predictive analysis portion of the reactor vessel internals flow-induced vibration analysis program will be included in the report.

The AP1000 Vibration Assessment Report (WCAP-15949) will be referenced in DCD Section 3.9.2.3 and will be included in the Referenced Material list in DCD Table 1.6-1.

The reactor internals vibration assessment will be deleted from the Combined License Information in DCD Section 3.9.8.1 and from the Combined License Information listed in DCD Table 1.8-2.

Design Control Document (DCD) Revision:

From DCD page 3.9-31: (These changes have been incorporated into DCD Revision 3)

3.9.2.3 Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions

The vibration characteristics and behavior due to flow-induced excitation are complex and not readily ascertained by analytical means alone. Assessment of vibrational response is done using a combination of analysis and testing. Comparisons of results obtained from reference plant vibration measurement programs have been used to confirm the validity of scale model tests and other prediction methods as well to confirm the adequacy of reference plant internals regarding flow induced vibration. The flow-induced vibration assessment **is documented in WCAP-15949 (Reference 18).**~~will be performed by the Combined License applicant.~~

From DCD Page 3.9-92: (These changes have been incorporated into DCD Revision 3)

3.9.8 Combined License Information

3.9.8.1 Reactor Internals Vibration Assessment and Predicted Response

~~The flow induced vibration assessment of the reactor internals will be performed by the Combined License applicant. The assessment will be consistent with the guidelines of Regulatory Guide 1.20.~~ Information including predicted vibration response and allowable response will be provided prior to the preoperational vibration testing of the first AP1000 consistent with the guidance of Regulatory Guide 1.20.

From DCD page 3.9-94, Section 3.9.9 References: (These changes have been incorporated into DCD Revision 3)

18. "AP1000 Reactor Internals Flow Induced Vibration Assessment Program," WCAP-15949-P (Proprietary) and WCAP-15949-NP (Non-proprietary), November 2002.

From DCD page 1.6-5, Table 1.6-1: (These changes have been incorporated into DCD Revision 3)

Table 1.6-1 (Sheet 4 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
3.8	WCAP-13891	AP600 Automatic Depressurization System Phase A Test Data Report, May 1994
	WCAP-14324	Final Data Report for ADS Phase B1 Tests, April 1995
	WCAP-15613	AP1000 PIRT and Scaling Assessment, March 2001
3.9	WCAP-7765-AR	Westinghouse PWR Internals Vibrations Summary Three-Loop Internals Assurance, November 1973
	WCAP-8766 (P) WCAP-8780	Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant, May 1976
	WCAP-8516-P (P) WCAP-8517	UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations, March 1975
	WCAP-10846 (P)	Doel 4 Reactor Internals Flow-Induced Vibration Measurement Program, March 1985
	WCAP-10865 (P) WCAP-10866	South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment, February 1985
	WCAP-8708-P-A (P) Volumes 1 and 2 WCAP-8709-A Volumes 1 and 2	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8446 (P) WCAP-8449	17x17 Drive Line Components Tests - Phase 1B 11, 111 D-Loop Drop and Deflection, December 1974
	WCAP-9693 (P)	Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants, June 1980
	WCAP-15949-P (P) **WCAP-15949-NP**	**AP1000 Reactor Internals Flow Induced Vibration Assessment Program, November 2002**

(P) Denotes Document is Proprietary

From DCD page 1.8-12, Table 1.8-2: (These changes have been incorporated into DCD Revision 3)

Table 1.8-2 (Sheet 2 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
2.5-13	Subsurface Instrumentation	2.5.4.6.11
2.5-14	Stability of Slopes	2.5.5
2.5-15	Embankments and Dams	2.5.6
3.3-1	Wind and Tornado Site Interface Criteria	3.3.3
3.4-1	Site-Specific Flooding Hazards Protective Measures	3.4.3
3.5-1	External Missile Protection Requirements	3.5.4
3.6-1	Pipe Break Hazards Analysis	3.6.4.1
3.6-2	Leak-Before-Break Evaluation of as-Designed Piping	3.6.4.2
3.6.3	Leak-Before-Break Evaluation of as-Built Piping	3.6.4.3
3.7-1	Seismic Analysis of Dams	3.7.5.1
3.7-2	Post-Earthquake Procedures	3.7.5.2
3.7-3	Seismic Interaction Review	3.7.5.3
3.7-4	Reconciliation of Seismic Analyses of Nuclear Island Structures	3.7.5.4
3.7-5	Seismic Analyses of Nuclear Island Structures at Soil Sites	3.7.5.5
3.8-1	Containment Vessel Design Adjacent to Large Penetrations	3.8.6.1
3.8-2	Passive Containment Cooling System Water Storage Tank Examination	3.8.6.2
3.8.3	Design Summary Reports - Structures	3.8.6.3
3.8-4	As-Built Summary Report	3.8.6.4
3.9-1	**Reactor Internal Vibration Response**	**3.9.8.1**
3.9-2	Design Specification and Reports	3.9.8.2
3.9-3	Snubber Operability Testing	3.9.8.3
3.9-4	Valve Inservice Testing	3.9.8.4
3.9-5	Surge Line Thermal Monitoring	3.9.8.5
3.9-6	Piping Benchmark Program	3.9.8.6

3.10-1	Experience-Based Qualification	3.10.6
3.11-1	Equipment Qualification File	3.11.5
4.2-1	Changes to Reference Reactor Design	4.2.5

PRA Revision:

None

NRC Additional Comments on WCAP-15949:

The following items in WCAP-15949 need further clarification:

1. (Editorial) Pg. 3-5, Figure 3-1: Three different reactor lower internals configurations are shown, but not specifically identified. AP1000 appears to be the one on the right; please identify the other two.

2. (Editorial) Pg. 5-16, Table 5-6. The Calculated Maximum (Stress) Gradient reported for the guide tubes, top location at 409 Hz, is not consistent with the other data reported. Please confirm that this should read 0.54e-4 psi/in, instead of 0.54 e04 psi/in.

3. (Editorial) Pg. 6-4, second line. The units for pump rotating frequency are expressed in mils. Please clarify.

4. Pg. 6-5, first bullet, and pg. 6-8, last paragraph. The acronym 'CFX' is used. Please provide additional explanation of what CFX refers to; e.g., is it a computer code, a computer model of a specific plant configuration, etc.?

5. (Editorial) Pg. 6-7, fifth paragraph. The minimum high cycle fatigue margin (3.16) reported for the guide tubes is not consistent with data listed in Table 6-5. Please clarify.

6. Pg. 6-9, Section 6.4.2.1. The calculation of required bolt preload concludes that the resulting needed preload <u>per</u> <u>bolt</u> is about 28,600 lb. Table 6-8, Net Preload Acting on Lower Flange of Core Shroud, reports a <u>net</u> bolt preload of 33,022 lbs, suggesting that this is the total preload for all sixteen bolts, which does not appear to be consistent with the preload required <u>per</u> <u>bolt</u>, from pg. 6-9. Please clarify.

7. Pg.6-13, Table 6-3. The guide tube (location H-14) steady flow load calculated by the CFD analysis is significantly lower than the corresponding load from the scale model results. Please provide some justification for this result, which is not consistent with other data trends reported in Table 6-3.

8. (Editorial) Pg. 7-2, fifth bullet. The locations of the two strain gages discussed are not included in Fig. 7-1, Location of Transducers for the AP1000 Preoperational Vibration Measurement Program. Please clarify.

9. Pg. 7-11, Fig. 7-3. These figures are graphically confusing; what is intended? what is the frame of reference, or coordinate system?

10. (Editorial) Superscripts, including the lower case letters 'a', 'b', and 'c', are used throughout the text and tables without accompanying footnotes explaining their meaning. Examples include Pg. 1-1, last paragraph, Pgs. 3-4, 4-2, and 6-2, Tables 4-2 through 4-7, Tables 5-3 through 5-6, etc. Please clarify the meaning of these superscripts.

Westinghouse Response to Additional Comments on WCAP-15949:

The changes indicated below will be incorporated in Revision 1 of WCAP-15949.

1. Page 3-5: Identification of the plant design for each of the three reactor internals are, left to right: 3 Loop 12 ft core, 3 Loop 14 ft core, and AP1000.

2. Page 5-16: The correct number is 0.54e-4 psi/in.

3. Page 6-4: The correct unit is Hertz (Hz).

4. Page 6-5 and 6-8: CFX is a computer code from AEA Technology Engineering Software Limited.

5. Page 6-7: The minimum high cycle fatigue margin for the guide tubes is 3.7 as given in Table 6-5.

6. Page 6-15: Table 6-8 title will be changed to indicate the given preload is per bolt. Also the sign on the hydraulic drag force on the core shroud will be corrected.

7. Page 6-13, Table 6-3: In general the ratios of CFD to measured 1/7 scale model mean loads vary between 0.83 and 1.19 or less than ± 20%. Given that the CFD to measured prototype ratios are well above unity, and the margins of the guide tubes using CFD are greater than 3, then the 17% uncertainty does not significantly change the design margin.

   This explanation will be added to the discussion on page 6-7.

8. Page 7-9: Figure 7-1 will be revised to show the location of the strain gages on the outer wall on the core barrel at the connection to the lower core support plate.

9. Page 7-11: The purpose of Figure 7-3 is to show that the core shroud beam modes are very similar both with and without the core included in the model. Figure 7-3 will be revised to show a different angle of the core shroud model to help clarify the mode shape. An

indication of the coordinate system will also be added to the figure.

10. Page 1-1: "[ ]" indicates that the enclosed data is proprietary data. The superscripts are defined as follows:
    "a" – Information reveals the distinguishing aspects of a process or component, structure, tool, method, etc.
    "b" – Information consists of supporting data relative to a process or component, structure, tool, method, etc.
    "c" – Information, if used by a competitor, would reduce competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, etc. of a similar product.
    "g" – Information is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner

The above information will be added as a footnote to page 1-1.

WCAP-15949 Revision:

The attached marked-up pages reflect the changes to the WCAP described above.

Design Control Document (DCD) Revision:

From DCD Revision 3 page 1.6-5, Table 1.6-1:

Table 1.6-1 (Sheet 4 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
3.8	WCAP-13891	AP600 Automatic Depressurization System Phase A Test Data Report, May 1994
	WCAP-14324	Final Data Report for ADS Phase B1 Tests, April 1995
	WCAP-15613	AP1000 PIRT and Scaling Assessment, March 2001
3.9	WCAP-7765-AR	Westinghouse PWR Internals Vibrations Summary Three-Loop Internals Assurance, November 1973
	WCAP-8766 (P) WCAP-8780	Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant, May 1976
	WCAP-8516-P (P) WCAP-8517	UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations, March 1975
	WCAP-10846 (P)	Doel 4 Reactor Internals Flow-Induced Vibration Measurement Program, March 1985
	WCAP-10865 (P) WCAP-10866	South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment, February 1985
	WCAP-8708-P-A (P) Volumes 1 and 2 WCAP-8709-A Volumes 1 and 2	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8446 (P) WCAP-8449	17x17 Drive Line Components Tests - Phase 1B 11, 111 D-Loop Drop and Deflection, December 1974
	WCAP-9693 (P)	Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants, June 1980
	WCAP-15949-P (P) WCAP-15949-NP	AP1000 Reactor Internals Flow Induced Vibration Assessment Program, ~~November 2002~~**April 2003**

From DCD Revision 3 page 3.9-94, Section 3.9.9 References:

(P) Denotes Document is Proprietary

1. "AP1000 Reactor Internals Flow Induced Vibration Assessment Program," WCAP-15949-P, **Revision 1** (Proprietary) and WCAP-15949-NP, **Revision 1** (Non-proprietary), ~~November 2002~~**April 2003**.

PRA Revision:

None

REPLACE WITH ATTACHED FIGURE

RAI 210.001 Revision 1

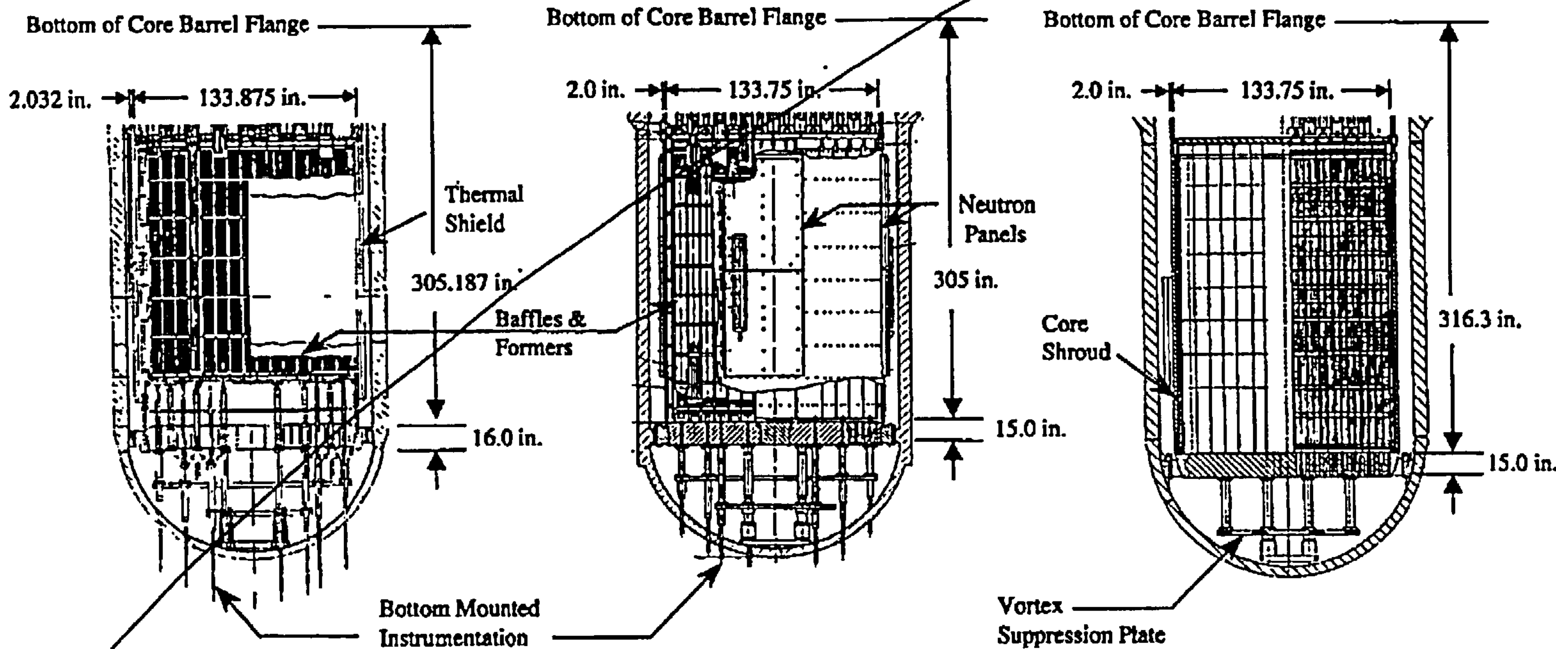
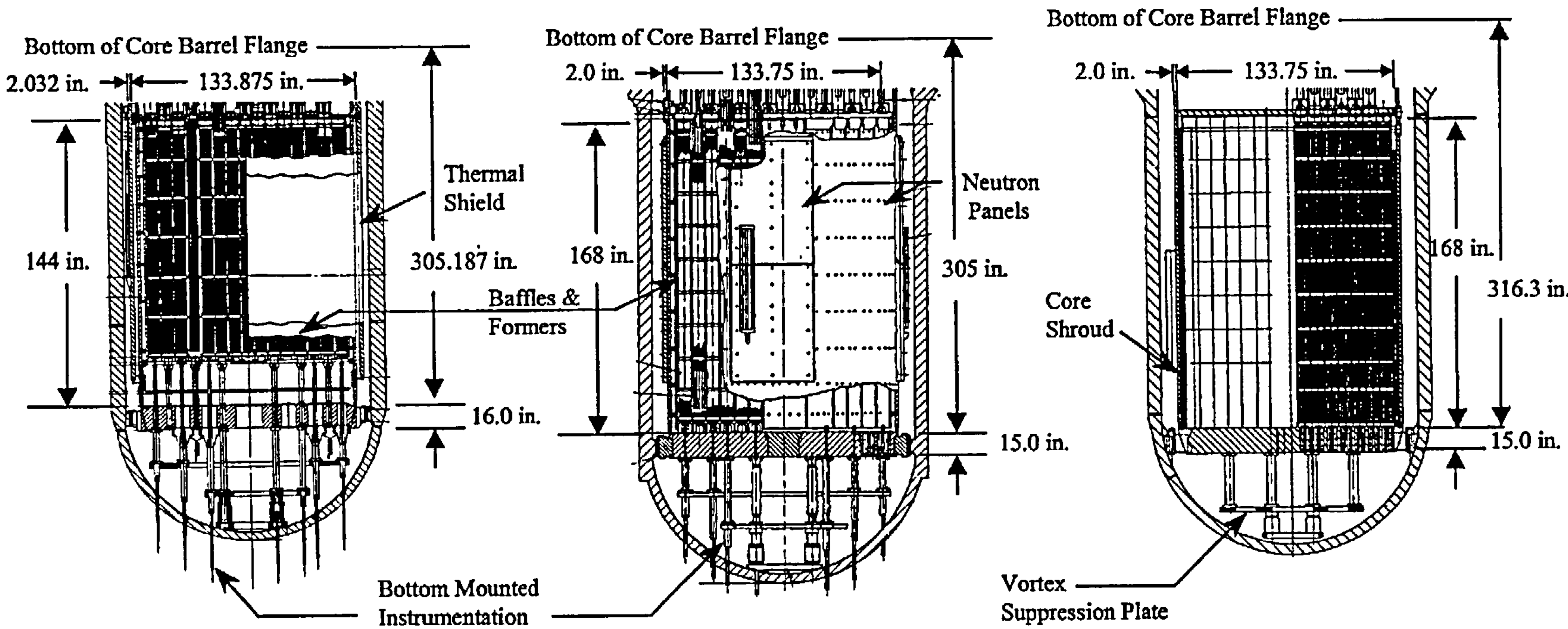


Figure 3-1. Design Differences in Lower Internals

RAI Number 210.001 R1-11



3 Loop, 12 ft. Core

3 Loop, 14 ft. Core (XL)

AP1000

Table 5-6 (cont.) ACSTIC Results for Upper and Lower Support Plates, Over 372 to 454 Hz Range, Hot, Full-Power Condition

b,c

Component	Frequency (Hz)	Location	Calculated Maximum Gradient		
			AP1000	Sequoyah 1	Sizewell B
Guide tubes	204	Top	0.003 psi/in	0.190 psi/in @ 316 Hz	0.1744 psi/in @ 178.86 Hz
		Middle	0.90e-3 psi/in	0.190 psi/in @ 316 Hz	0.1316 psi/in @ 178.86 Hz
		Bottom	0.51e-3 psi/in	0.190 psi/in @ 316 Hz	0.0159 psi/in @ 178.86 Hz
	409	Top	~~0.54e04~~ psi/in 0.54e-4	0.190 psi/in @ 316 Hz	0.1744 psi/in @ 178.86 Hz
		Middle	0.84e-5 psi/in	0.190 psi/in @ 316 Hz	0.1316 psi/in @ 178.86 Hz
		Bottom	0.81e-5 psi/in	0.190 psi/in @ 316 Hz	0.0159 psi/in @ 178.86 Hz
Support columns	204	Top	0.003 psi/in	0.190 psi/in @ 316 Hz	0.1744 psi/in @ 178.86 Hz
		Middle	0.90e-3 psi/in	0.190 psi/in @ 316 Hz	0.1316 psi/in @ 178.86 Hz
		Bottom	0.29e-3 psi/in	0.190 psi/in @ 316 Hz	0.0159 psi/in @ 178.86 Hz
	409	Top	0.50e-4 psi/in	0.190 psi/in @ 316 Hz	0.1744 psi/in @ 178.86 Hz
		Middle	0.13e-4 psi/in	0.190 psi/in @ 316 Hz	0.1316 psi/in @ 178.86 Hz
		Bottom	0.31e-4 psi/in	0.190 psi/in @ 316 Hz	0.0159 psi/in @ 178.86 Hz

0.5 is the vertical displacement (mils) of the upper support plate due to the pump rotating frequency (Hz ~~mils~~). This is the calculated value of the center section of the system model upper support plate multiplied by the ratio of the finite element model to system model upper support plate stiffness.

0.884 is the displacement to load correlation ($mils/psi_{load}$).

0.025 is the maximum pressure from the pump first blade passing frequency (psi).

0.025 is the maximum pressure from the pump second blade passing frequency from (psi).

25 is a factor to account for increased coincidence of the first and second blade passing frequency components with a structural mode.

This deflection, multiplied by the maximum stress to USP center deflection ratio gives the maximum stress, $\sigma_{USP,max}$, of the AP1000 upper support skirt and flange:

$\sigma_{USP,max} = 3.974 \times (102.9);$
$\sigma_{USP,max} = 409$ psi

The calculated stress for the upper support plate, compared to the allowable alternating stress of 13,000 psi at 10^{11} cycles from the ASME Code Figure I-9.2.2 Curve C, gives a margin of safety of 30.7.

6.3.3.2 Upper Support Plate and Skirt Transition

Using the calculated deflection above, the perforated region and skirt transition margins are calculated.

For the perforated region, the maximum stress to displacement ratio is 16.97 psi/mil. Multiplying this by the previously calculated deflection results in a maximum stress of 67.44 psi. This alternating peak stress is well below the allowable alternating stress of 13,000 psi at 10^{11} cycles from the ASME Code Figure I-9.2.2 Curve C.

At the transition locations of the skirt, the maximum stress to displacement ratio is 55.43 psi/mil. Multiplying this by the previously calculated deflection results in a maximum stress of 220 psi. This alternating peak stress is also well below the ASME allowable alternating stress.

6.3.3.3 Guide Tubes and Support Columns

<u>Methodology</u>

The guide tubes in the AP1000 upper internals are similar to the 17x17 AS guide tubes, and the support columns are identical to those in standard plants such as Doel 3 so that flow-induced vibration and load correlations can be confidently applied to the AP1000 components.

To calculate the high cycle fatigue margins of the guide tubes and support columns, the following methodology was used:

- A CFD model (Reference 5-3) of the 3XL upper internals flow paths during the hot function test was made on the CFX computer code. The extent of this model was from the upper end of the core cavity to the outward end of the reactor vessel outlet nozzles.

- The mean flow loads on the most highly loaded guide tubes and support columns in the 3XL design were calculated and compared to measured values. No adjustments were necessary.

  The mean flow loads on the most highly loaded AP1000 components were calculated on a CFD model that corresponded closely to that of the 3XL design. The AP1000 upper internals have more guide tubes and support columns than the Doel 3 plant as discussed in section 3.1. The CFX model includes the AP1000 number of guide tubes and support columns as well as the AP1000 outlet nozzle geometry.

- The moments and forces at key locations of the AP1000 components were determined by scaling the measured 3XL responses to the AP1000 flow condition.

- The mean loads, and the moments and forces for quasi-static, first mode, and second mode frequency bands were determined from the RMS signal levels of the strain gages mounted on the guide tubes and support columns in the 3XL upper internals prototype measurement program.

- For the quasi-static vibration, the moment inferred from the strain gages was scaled to the AP1000 component by ratioing the moments produced at the strain gage elevation by the mean flow loads.

- For the modal responses, the moments inferred from the strain gages were scaled by the ratio of the square root of the product of the mean force squared times the eigenvector squared summed over the length of the component.

- The moments and shear forces due to the flow loads for each frequency band discussed above were added as the square root of the sum of the squares at key locations.

- The RCP-induced excitations were added algebraically at each location.

- The moments due to core plate motion were added as the square root of the sum of the squares to the forces from directly impinging flows.

- The moments and shear forces at each cross section were compared to acceptable values to determine the component factor of safety for high cycle fatigue. The acceptable values were obtained from a finite element stress model and ASME code allowable stresses.

Measured and calculated steady flow loads for the three-loop standard configuration (Doel 3) are compared in Table 6-3. The measured loads were obtained from Reference 6-1. As shown in the table, steady flow loads predicted from 3XL 1/7 scale model data were somewhat greater than those measured using the Type 1 support column. Therefore,

$$\beta = 0.37 + 1.76\left[\frac{8.466\text{ in.}}{5.48\text{ in.}}\right] = 3.0$$

$\beta > \beta_{min}$, so 2.4 will be used to find U_c.

$$U_c = (f_n)(D)(\beta)\left[\frac{(m_0)(\delta_n)}{(\rho\rho)(D)^2}\right]^{1/2}$$

where:

f_n = 1st mode natural frequency (62 Hz)
D = outside diameter (7.46 in)
β = fluidelastic constant (2.4)
m_0 = mass per unit length (4.442 lbm/in)
δ_n = logarithmic decrement of damping (0.176)
ρ = fluid density (0.0245 lbm/in^3)

Substituting:

$$U_c = (62)(7.46)(2.4)\left[\frac{(4.442)(0.176)}{(0.0245)(7.46)^2}\right]^{1/2} = 841\,\text{in}/\text{s} = 70\,\text{ft}/\text{s}.$$

Similarly, the support column has the following values for the variables:

f_n = 1st mode natural frequency (102 Hz)
D = outside diameter (3.5 in)
β = fluidelastic constant (2.4)
m_0 = mass per unit length (4.442 lbm/in)
δ_n = logarithmic decrement of damping (0.088)
ρ = fluid density (0.0245 lbm/in^3).

Substituting these values into the velocity equation:

$$U_c = (102)(3.5)(2.4)\left[\frac{(4.442)(0.088)}{(0.0245)(3.5)^2}\right]^{1/2} = 977\,\text{in}/\text{s} = 81.4\,\text{ft}/\text{s}$$

By comparison, the maximum value of the velocity calculated by CFD analysis using the CFX computer code (Reference 5-3) in the region of the guide tube and support column is 27.6 ft/s. Because this velocity is much lower than 70 ft/s and 81.4 ft/s, there is a margin of 2.54 for the guide tube and 2.95 for the support column, respectively.

Guide tube natural frequencies occur at frequencies close to the first and second harmonics of the RCP blade passing frequencies in the AP1000 design. Although greater separation of these frequencies is desired, it is noted that: Analysis of guide tubes for other plants made for similar differences between pump blade passing frequencies and guide tube natural frequency have shown that the resulting guide tube stresses are well within allowable values.

Detailed stress analysis for the 17x17AS guide tube was performed in Reference 6-2. In this document, the flange-to-enclosure weld throat primary and secondary tensile and shear stresses are related to a bending moment. These stresses are scaled to the corresponding AP1000 bending moment. For welds, per the ASME Code, Subsection NG, a stress concentration factor of 4 is used. The total stress intensity, corrected for the elastic modulus used for the ASME code design fatigue curve results in a high cycle fatigue alternating stress intensity value. The allowable value is taken from the ASME code, Figure I-9.2.2 for the endurance limit of welds and adjacent base metal at 10^{11} cycles. The result indicates a margin (allowable/(actual −1)) of 5.4.

A similar procedure is used to determine the margin at the continuous section card/enclosure weld and the bottom flange/enclosure weld.

Using relationships developed for other purposes, the force on support pin is determined and the stress at the highest stress location is determined. Based on a finite element model of the pin.

The resulting margins, summarized in Table 6-5 indicate a minimum guide tube fatigue margin high cycle fatigue of ~~3.16~~ to code allowable stresses. The margin of safety for all guide tube fatigue assessments is acceptable. [3.7]b/c

INSERT A →

Upper Support Column High Cycle Fatigue Analysis

To determine the high cycle fatigue margins for the upper support columns, the moments due the quasi-static, modal core plate and RCP responses were combined at locations of interest. The margin to ASME code allowable values was then determined at each location. The results, listed in Table 6-6 indicate a minimum margin of 1.03, which is acceptable.

6.3.3.4 Fluidelastic Stability Evaluation for Guide Tubes and Support Columns

From Appendix N of the ASME Boiler and Pressure Vessel Code (1998 Edition, 2000 Addenda), the minimum fluidelastic constant for any array type is $\beta = 2.4$. For a square array (Reference 6-3):

$$\beta = 0.37 + 1.76\frac{T}{D}$$

Between a guide tube and support column,

$$T = \text{pitch} = 8.466 \text{ in.}$$

$$D = \text{width(avg)} = \frac{7.46 \text{ in.} + 3.50 \text{ in.}}{2} = 5.48 \text{ in.}$$

Insert A

In general the ratios of CFD to measured 1/7 scale model mean loads varies between 0.83 and 1.19 or less than ± 20%. Given that the CFD to measured prototype ratios are well above unity, and the margins of the guide tubes using CFD are greater than 3, then the 17% uncertainty does not significantly change the design margin.

Table 6-7 Local Flow Turbulence[1] Vibratory Forces on Core Shroud Bolts

Bolt Number	Shear-X	Shear-Y	Shear-Z	$(F_x^2 + F_y^2)^{1/2}$
1	1050.7	980.7	363.1	1045.7
2	1208.2	547.4	634.6	838.1
3	1187.7	668.9	424.1	792.0
4	1451.5	380.6	761.3	851.1
5	1683.7	355.8	625.2	719.4
6	1711.1	586.0	309.7	662.8
7	1753.9	413.1	568.3	702.6
8	1509.4	807.2	374.4	889.8
9	1075.4	966.8	408.0	1049.3
10	1063.1	510.8	760.8	916.4
11	1111.7	749.4	473.5	886.5
12	1346.5	409.7	884.8	975.1
13	1873.9	356.1	597.0	695.1
14	1991.0	485.4	310.0	575.9
15	1932.2	439.5	509.0	672.5
16	1472.0	858.8	333.3	921.2
Sum of forces on all bolts	204.8	6935.9	4647.8	8349.2

1. The vibratory loads due to turbulence excitation of the system fundamental acoustic mode and due to RCP speed-related excitations must still be added in.

Table 6-8 Net Preload Acting on Lower Flange of Core Shroud per Bolt for Hot Full Power

Load Source	Load (lbs)
Core shroud weight	46479/16 = 2905
Core shroud buoyancy force	-4183/16 = -261
Hydraulic lift force on the core shroud	-(73915 + 14402)/16 = -5520
Hydraulic drag force on the core shroud	-988/16 = -62
Bolt preload	33022
Net preload	30084

REPLACE WITH ATTACHED FIGURE

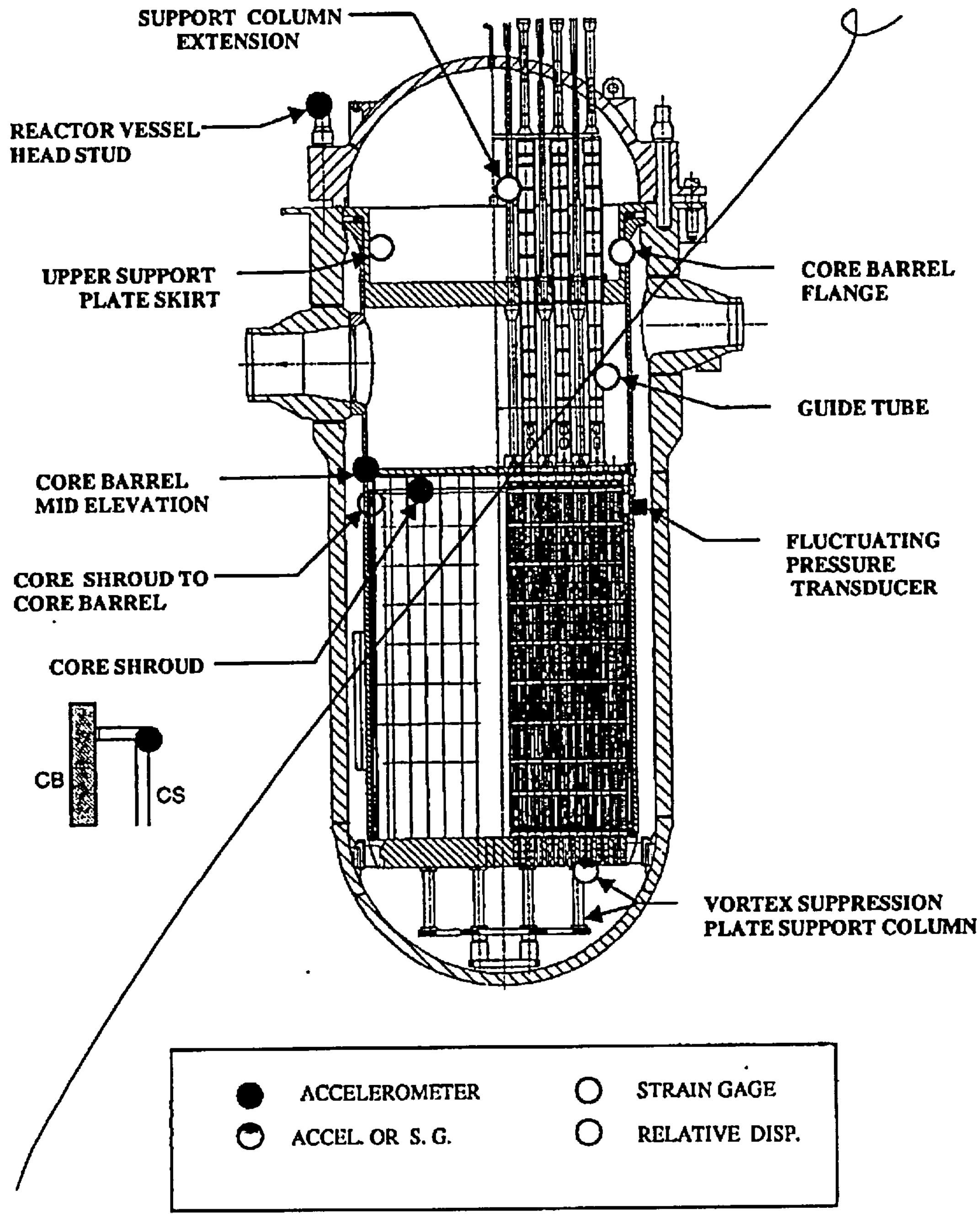
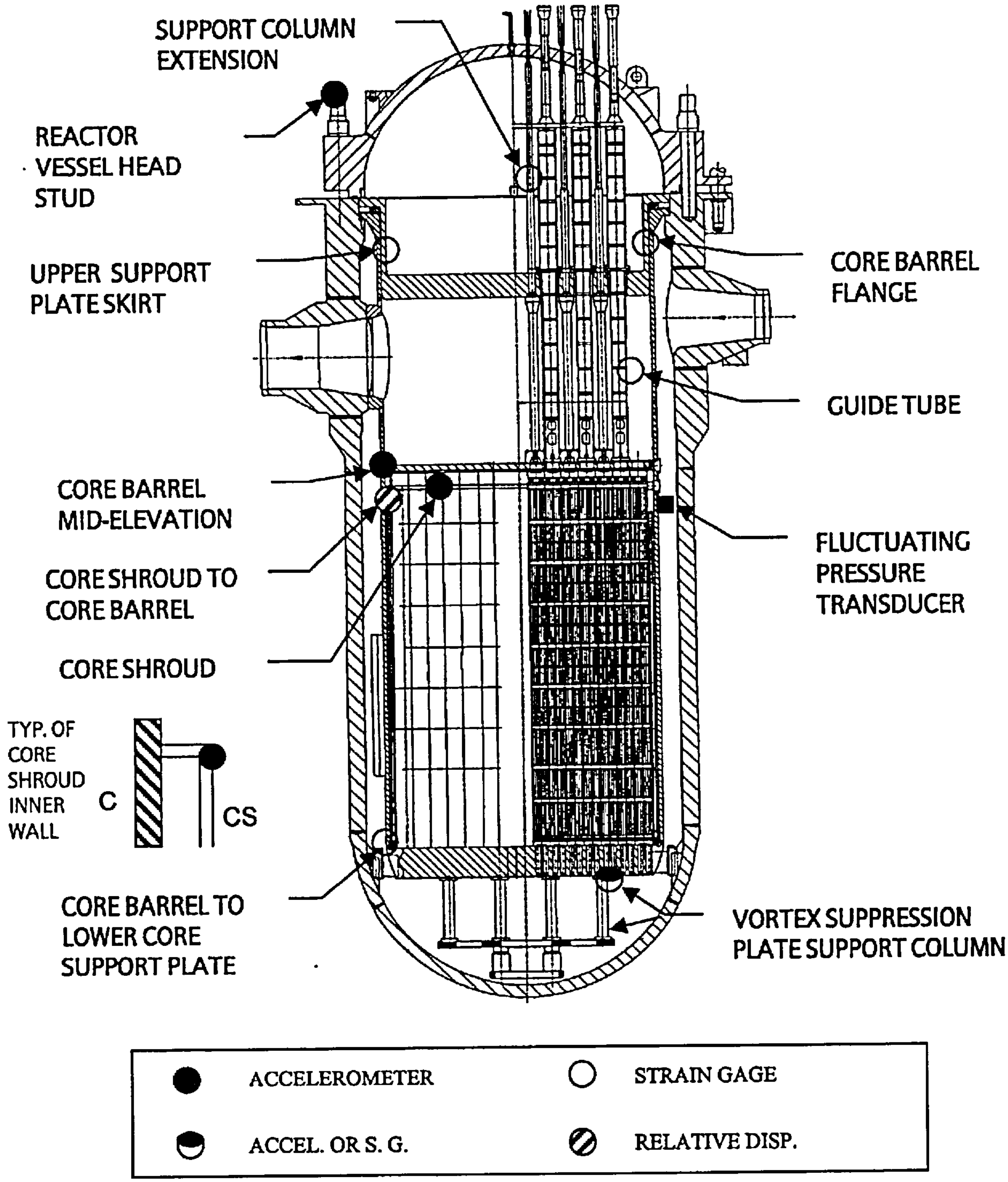


Figure 7-1 Location of Transducers for the AP1000 Preoperational Vibration Measurement Program


SUPPORT COLUMN EXTENSION
REACTOR VESSEL HEAD STUD
UPPER SUPPORT PLATE SKIRT
CORE BARREL FLANGE
GUIDE TUBE
CORE BARREL MID-ELEVATION
CORE SHROUD TO CORE BARREL
CORE SHROUD
FLUCTUATING PRESSURE TRANSDUCER
TYP. OF CORE SHROUD INNER WALL
C
CS
CORE BARREL TO LOWER CORE SUPPORT PLATE
VORTEX SUPPRESSION PLATE SUPPORT COLUMN
ACCELEROMETER
STRAIN GAGE
ACCEL. OR S. G.
RELATIVE DISP.

REPLACE WITH ATTACHED FIGURE

FREQ =10.092

(with core)

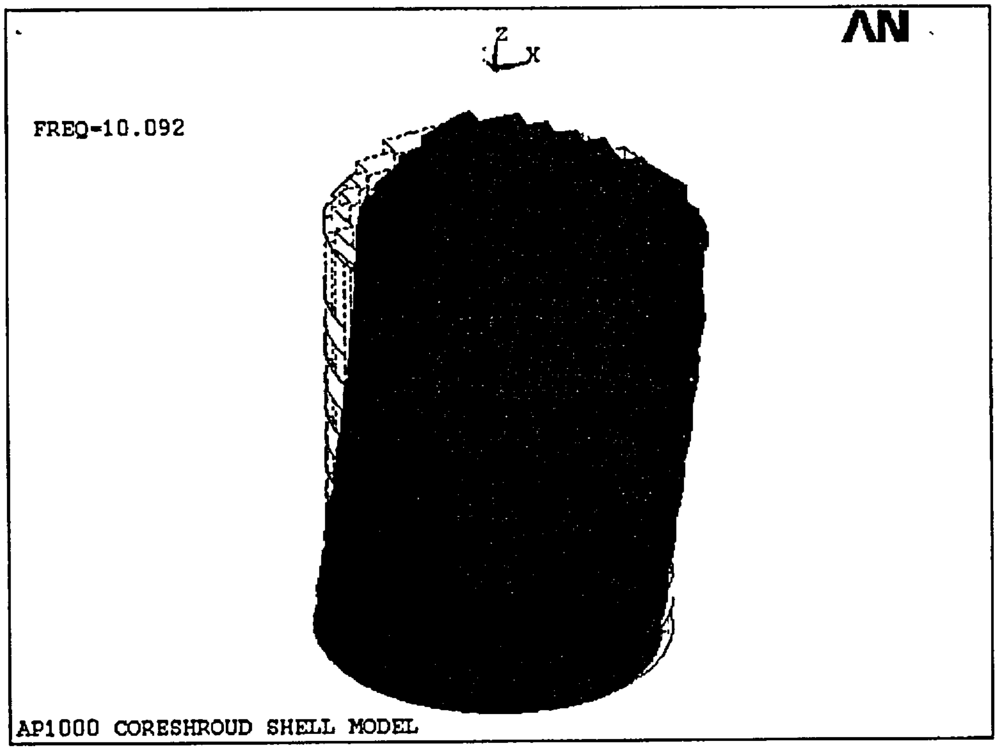
[redacted]

FREQ =10.517

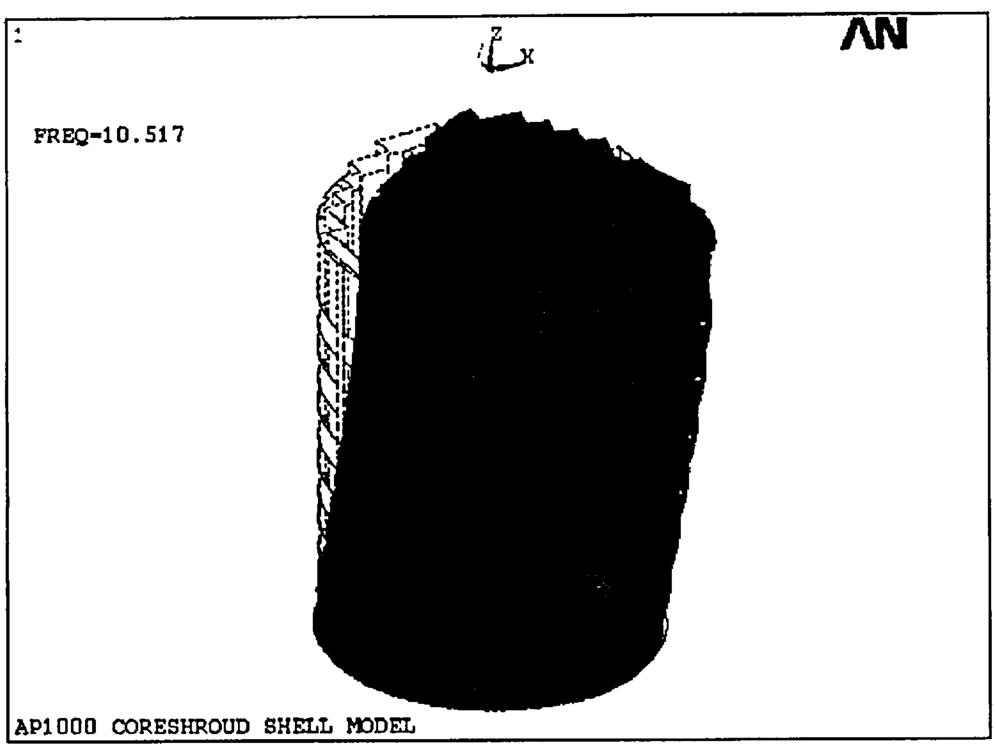
(without core)

[redacted]

Figure 7-3 Core Shroud Beam Modes With and Without Core



With Core



Without Core

1 INTRODUCTION

With respect to the reactor internals preoperational test program, the first AP1000 plant reactor vessel internals are classified as prototype as defined in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.20, Revision 2. AP1000 reactor vessel internals do not represent a first-of-a-kind or unique design based on the general arrangement, design, size, or operating conditions. The units referenced as supporting the AP1000 reactor vessel internals design features and configuration have successfully completed vibration assessment programs, including vibration measurement programs. These units have subsequently demonstrated extended satisfactory inservice operation. Additional background on the Westinghouse position with regard to NRC Regulatory Guide 1.20 is provided in Westinghouse Nuclear Safety Position Papers (References 1-2 and 1-3).

The vibration assessment approach for the AP1000 is believed to meet the intent of NRC Regulatory Guide 1.20 and is similar to the approach taken by Westinghouse on previous plants. Westinghouse performed a similar vibration assessment in support of the AP600 Design Certification. The AP600 vibration assessment was documented in Reference 1-4. This report provides a similar format and content as was provided for the AP600 assessment.

The purpose of this AP1000 reactor internals vibration assessment program is to demonstrate structural adequacy with respect to flow- and pump- induced vibrations. Estimates of flow-induced vibration levels and forces (or relative values) of the AP1000 plant are made on the basis of scale model tests, tests on prototype reactors, and results of analytical calculations. Based on this information, the vibratory behavior of the reactor internals is well characterized and the vibration amplitudes are sufficiently low for structural adequacy of the components.

The H. B. Robinson no. 2 plant has historically been established as the prototype design for three-loop plant internals and was instrumented and tested during hot functional testing. The test and analysis results of the three-loop configuration of Reference 1-3 demonstrate that the vibration levels of the reactor internals components are low and that the vibrations are adequately characterized to ensure structural integrity. These results are further augmented by References 1-5, 1-6, 1-7, and 1-8 to address the effects of successive hardware improvements in Westinghouse designs, which are discussed in the following sections.

The AP1000 reactor internals are generally similar to subsequent three-loop, 12- and 14-foot core designs (specifically Doel 3 and Doel 4), which have incorporated these improvements, and on which instrumented plant test programs have been completed. The dimensions of the AP1000 core barrel wall thickness and reactor vessel to core barrel downcomer annulus are similar to those of Doel 3 and Doel 4. The AP1000 guide tube and support column designs are the same as the designs used in Doel 3 and Doel 4. The upper internals components vibration responses were measured at Doel 3 (Reference 1-9), and the lower internals were measured at Doel 4 (Reference 1-10).

The total reactor mechanical design flow rates in these instrumented tests were approximately [314,600 gpm][b] (1) and [322,500 gpm][b] as compared to the AP1000 value of [327,600 gpm][b] mechanical design flow. Table 1-1 lists the ratios of the AP1000 to Doel 4 plant velocities at various locations based on flow rates of [322,500 gpm][b] for Doel 4 and [327,600 gpm][b] for the AP1000 design. The tabulated velocities show similar values for the AP1000 and Doel 4 reactors with the exception of the inlet nozzle

ADD FOOTNOTE AS SHOWN ON ATTACHED PAGE

Footnote to Page 1-1

(1) "[ ]" indicates that the enclosed data is proprietary.
The superscripts are defined as follows:
"a" – Information reveals the distinguishing aspects of a process or component, structure, tool, method, etc.
"b" – Information consists of supporting data relative to a process or component, structure, tool, method, etc.
"c" – Information, if used by a competitor, would reduce competitor's expenditure of resources or improve the competitor's advantage in the design, manufacture, etc. of a similar product
"g" – Information is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.012 (Response Revision 1)

Question:

The application indicates that the reactor coolant pump pressure housing will be made from SA 351 or SA 352 CF3A material and that reactor coolant pressure boundary valve bodies may be castings of SA 351 CF3A. The application also indicates that cast austenitic stainless steel will not exceed a ferrite content of 30 FN (Ferrite Number). CASS reactor coolant pressure boundary components are subject to thermal embrittlement. Please provide additional information discussing the impact of this aging effect on the integrity of these components along with a discussion of how this thermal embrittlement mechanism has been considered in the design and material selection for these components. Also, please discuss the need for potential licensees of AP1000 plants to perform inspections to detect this aging effect. (Section 5.2.3)

NRC Follow-On Comment:

The Westinghouse response to this question is acceptable subject to clarification of the method used to calculate the δ-ferrite. This clarification needs to be included in the DCD.

Westinghouse Response (Revision 1):

Selection of the low carbon grade casting, CF3A, is a preventive measure considered to control the impact of thermal aging embrittlement. Also, the ferrite content of the material will be controlled by the material specifications to be below 20 FN. **Calculation of ferrite content will be based on ASTM A800 (Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof).** Based on our experience with the casting materials, there should be no significant impact of thermal aging on the integrity of the components when the carbon and ferrite content of the castings are controlled.

Inspection to detect any thermal aging effects will be part of the in-service inspection program that is the responsibility of the Combined License Applicant of each AP1000. The program will reference the edition and addenda of the ASME Code Section XI to be utilized and will be submitted to the NRC. The Combined License Applicant commitment to provide the in-service inspection program is included in DCD Section 5.2.6.2.

DCD Section 5.2.3.1 will be revised to limit the ferrite content of cast austenitic stainless steel to a maximum of 20 FN.

DCD Section 5.2.3.1 will be revised to include the standard used to calculate ferrite content.

Design Control Document (DCD) Revision:

From DCD page 5.2-9, Section 5.2.3.1: (Already Incorporated in DCD Revision 3)

Table 5.2-1 material specifications are the materials used in the AP1000 reactor coolant pressure boundary. The materials used in the reactor coolant pressure boundary conform to the applicable ASME Code rules. Cast austenitic stainless steel does not exceed a ferrite content of ~~30~~**20** FN.

From DCD Revision 3 page 5.2-9, Section 5.2.3.1:

Table 5.2-1 material specifications are the materials used in the AP1000 reactor coolant pressure boundary. The materials used in the reactor coolant pressure boundary conform to the applicable ASME Code rules. Cast austenitic stainless steel does not exceed a ferrite content of 20 FN. **Calculation of ferrite content is based on ASTM A800.**

From DCD Revision 3 page 5.2-30:

5.2.7 References

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

2. EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report, Interim Report, April, 1982.

3. Logsdon, W. A., Begley, J. A., and Gottshall, C. L., "Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals," WCAP-9292, March 1978.

4. Golik, M. A., "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems," WCAP-7477-L (Proprietary), March 1970, and WCAP-7735 (Nonproprietary), August 1971.

5. Enrietto, J. F., "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," WCAP-8324-A, June 1975.

6. Enrietto, J. F., "Delta Ferrite in Production Austenitic Stainless Steel Weldments," WCAP-8693, January 1976.

7. **ASTM A800, ~~"~~"Standard Practice for Steel Casting, Austenitic Alloy, Estimating Ferrite Content Thereof~~"~~".**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.021 (Revision 1 Response)

Question:

In the AP600 review, RAIs 251.2 through 251.23 pertain to RCP flywheel integrity. In addition, WCAPs-13734 and 13735, "Structural Analysis Summary for the AP600 Reactor Coolant Pump Flywheel," were submitted as supplemental information for the revised response to question 251.11. Confirm that these responses and the WCAPs are applicable to the AP1000 application as it pertains to RCP flywheel integrity. Should aspects of these responses or reports not be applicable, provide updated information to address the AP600 RAIs as applicable to AP1000 RCP flywheel integrity. (Section 5.4.1)

Note: AP600 RAIs 251.2 through 251.23 were issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its responses to these RAIs in letters dated January 14, May 24, and May 28, 1993 (NUDOCS Accession Nos. 9301250260, 9306020387, and 9306020220, respectively).

NRC Follow-On Comments:

In response to AP600 RAI 251.3, the applicant states that the critical flaw size for the AP1000 flywheel design was approximately 1 inch, as opposed to 2 inches critical flaw size that was reported in response to the RAI for the AP600 designs. However WCAP-15994-P, Revision 0, indicates that the critical flaw size for the AP1000 and AP600 designs are virtually the same (1.075 inches for AP1000 vs 1.16 inches for the AP600 design). There appears to be an error in the AP 600 RAI response. The applicant needs to clarify what the actual critical flaw sizes are for the two flywheel designs.

In response to AP600 RAI 251.8, the applicant's response for the AP1000 flywheel design is virtually the same as that for the AP600 flywheel design, with the exception that the response for the AP1000 increases the kinetic energy for flywheel fragments that could potential impact the RCP structure from 10% of the tensile energy-absorbing capability in AP600 RCP structure to 15% of the tensile energy-absorbing capability in AP1000 RCP structure. However, there is some confusion in the wording of the final paragraph of the applicant's AP1000 response to AP600 RAI 251.8 as to whether potential flywheel fragments will penetrate the flywheel enclosure or not, and whether they could impact the surrounding RCP structure.

In response to AP600 RAIs 251.17 and 251.19, the applicant discusses the relationship of the structural integrity of the RCP flywheel enclosures to revolution of the flywheels at normal and design operating speeds. In this case, the applicant's responses for AP1000 are virtually the same as the previous AP 600 design responses to the RAIs. However, in these AP1000 responses to these AP600 RAIs, the applicant has a statement that the flywheel impact on the enclosure was assessed for a postulated rupture of the AP600 flywheel under normal and design revolution speeds and that the impact on the AP600 enclosure welds was within acceptable ASME limits. The applicant then concludes that since the AP1000 flywheel design is similar to that for the AP600 flywheels, "it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits" during revolutions of the AP1000 flywheel at normal speeds (as addressed in the AP1000 response to AP600 RAI 251.17) and design speeds (as addressed in the AP1000 response to AP600 RAI 251.19). The applicant needs to state whether this is based solely on engineering judgement or whether the applicant actually did the corresponding analyses at normal speed and design speed for the AP1000 design.

Westinghouse Response (Revision 1):

Responses to AP600 RAIs 251.2 through 251.23 specific to the AP1000 design are given below. The format is to repeat the AP600 question and provide a response specific to the AP1000 design. WCAP-15994-P, **Revision 1** (Proprietary), WCAP-15994-NP, **Revision 1** (Non-Proprietary), "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel", November 2002, has been issued to supplement some of the responses given below. In the responses below, this WCAP is referred to as Reference 1.

When the responses below refer to other AP600 RAI responses, the reference is to the AP1000 response to the AP600 RAI as given here in the overall response to this AP1000 RAI.

AP600 RAI 251.2

Westinghouse proposes to use a depleted uranium alloy casting in an Inconel alloy welded enclosure to construct the pump flywheel. These materials are not addressed in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity". Provide technical justifications for the use of these materials (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.2

As noted in Subsection 5.4.1 of the DCD, the AP1000 canned motor reactor coolant pump uses a fundamentally different approach to demonstrate safe operation of the flywheel than the design approach for which Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 were developed. Of prime importance in the consideration of flywheel integrity is minimizing the potential for generation of missiles from the flywheel in conformance with the requirements of General Design Criteria 4. The AP1000 approach is to demonstrate that fragments from a postulated flywheel fracture do not penetrate the surrounding pressure boundary and thus do not become missiles. See the AP1000 response to AP600 RAI 251.11 for additional information on the analysis of the retention of flywheel fragments. This basis of containing postulated fragments is the same as for the rotor and other rotating parts in previous shaft seal pump designs. The approach behind the recommendations of Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 is to minimize the potential for a flywheel fracture by extensive testing and inspection of the flywheel.

Although conformance with the recommendations in the regulatory guide is not the only means to demonstrate safe operation of the pump, many of the recommendations are followed in the design and fabrication of the flywheel assembly for operational reliability. Since the AP1000 design does not rely on flywheel material integrity to minimize the potential for the generation of missiles, the quality assurance requirements inherent in the use of ASME Code pressure boundary quality material as suggested by the Standard Review Plan are not required. The design requirements for the flywheel assembly materials are selected to provide a high level of operational reliability. The basis for the design requirements for the flywheel assembly materials is outlined below.

The flywheel assembly is a uranium-alloy casting or forging surrounded by a nickel-chromium-iron alloy enclosure. The material strength used for the analyses that demonstrate flywheel integrity is based on the material specification outlined in AP1000 DCD Table 5.4-2. The material toughness is demonstrated by the yield strength and elongation. See the AP1000 response to AP600 RAI 251.3 for additional information on the fracture toughness properties of the uranium alloy. Since the uranium alloys to be used in the flywheel were not developed for use as pressure boundary materials, ASME Code material specifications do not exist. See the AP1000 response to AP600 RAI 251.23 for additional information on the material specification. Nevertheless, quality assurance practices can confirm that the minimum material requirements are met. The nickel-chromium-iron Alloy 690 material used in the enclosure is a commonly used material with established material specifications.

The uranium alloy does not come in contact with the reactor coolant. The Alloy 690 enclosure material has been shown to be compatible with reactor coolant in other applications. The operating temperature of the coolant surrounding the flywheel assembly is substantially less than the reactor coolant system operating temperature, so stress corrosion cracking of the Alloy 690 is not expected to be an issue. See the AP1000 response to AP600 RAI 251.21 for additional information on the resistance to stress corrosion cracking of the flywheel enclosure.

AP600 RAI 251.3

Westinghouse indicates that the fracture toughness guidelines in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14 are not applicable to depleted uranium alloy castings. Provide information on the fracture toughness properties for this material and propose fracture toughness requirements with technical justifications (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.3

The fracture toughness of the uranium alloy casting is approximately 50 ksi √in. between 100°F and 200°F based on available data. Over the same temperature range the minimum impact energy (Charpy V-notch) is 10 foot-pounds. The material specification for the flywheel material includes a requirement for this minimum impact energy. The material specification does not include a fracture toughness requirement, but the properties and processing specified define a material that meets the 50 ksi √in. minimum.

Calculation of the critical flaw sizes (Reference 1) is based on the 50 ksi √in. fracture toughness. The minimum critical flaw size is greater than 1 inch for a full-length axial crack on the inner diameter. This flaw size was calculated for assembly plus design conditions (125% overspeed).

The minimum critical flaw sizes for a full-length axial crack reported in WCAP-15994, Revision 0 for both the AP600 (1.16 inches) and AP1000 (1.075 inches) flywheels are correct. The critical flaw size for a semi-elliptical (critical radius) crack is greater than 2 inches for both the AP600 and AP1000 flywheels.

AP600 RAI 251.4

Provide information on the fabrication process and resulting quality for the depleted uranium alloy casting (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.4

The melting of the depleted uranium alloy for the flywheel casting or forging billet is done under vacuum or inert atmosphere to provide a high quality product. The vacuum or inert atmosphere prevents reaction of the uranium with air and minimizes the potential for the formation of voids. Because of the density of the uranium alloy, slag and other impurities tend to float to the top of the molten metal and porosity in the cast material is not a problem. The molds for the casting are treated to minimize the contamination of the uranium with carbon. The rest of the manufacturing process is controlled to minimize the contamination of the uranium alloy with carbon and hydrogen. Excessive carbon reduces the ductility of the uranium alloy. Hydrogen contamination may induce delayed cracking. Because of the thickness of the flywheel, the final heat treatment is a solution anneal in a vacuum furnace followed by a slow cooling. Other heat treatments such as annealing followed by water quenching and aging hardening are not appropriate for a thick uranium alloy flywheel. See the AP1000 response to AP600 RAI 251.22 for additional discussion of the heat treatment.

AP600 RAI 251.5

Section 1A of the SSAR indicates that the AP600 design meets the guidelines of Regulatory Position 1.d in Regulatory Guide 1.14. However, the flywheel, including the enclosure welds, will not be inspected. Discuss how the flywheel design meets Regulatory Position 1.d.

Westinghouse AP1000 Response to AP600 RAI 251.5

The uranium alloy flywheel is not subject to welding operations, including repair welding, or any other finishing operations that use thermal methods. The component parts of the enclosure are connected together with flexible, full-penetration welds. These welds are inspected following fabrication by ultrasonic testing and liquid penetrant testing. ASME Code, Section III criteria for structural welds are used as guidelines to establish welding and inspection requirements. See the AP1000 response to AP600 RAI 251.14 for additional information on the analysis and inspection of the enclosure flexible welds. The enclosure represents only a small fraction of the energy in a rotating flywheel assembly. The locations of the flexible welds are such that there is minimal effect on the fracture analysis.

AP600 RAI 251.6

Regulatory Positions 2.c, 2.d, and 2.e in Regulatory Guide 1.14 recommends that an analysis be submitted for staff review. Provide the analysis with appropriate technical justifications. Further, because no inservice inspection for the flywheel is being proposed, describe the flaw size assumed in its analysis (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.6

Regulatory Positions 2.c., 2.d., and 2.e. in Regulatory Guide 1.14 recommend that analyses be conducted to predict the critical speed for ductile failure, nonductile failure, and excessive deformation of the reactor coolant pump flywheel. As noted in Subsection 5.4.1 of the DCD and the AP1000 response to AP600 RAI 251.2, the approach to demonstrate safe operation of the AP1000 canned motor reactor coolant pump flywheel differs from the design approach for which Regulatory Guide 1.14 was developed. The AP1000 design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of the analysis of critical flywheel failure speeds.

The analysis completed for the flywheel structure evaluates the stress intensity levels at the normal speed and at the design speed of 125 percent of normal. The calculated stress levels are evaluated against ASME Code, Section III, Subsection NG stress limits and the recommended stress limits in Positions 4.a. and 4.c. of Standard Review Plan 5.4.1.1. of one-third and two-thirds of yield stress for normal speed and design speed, respectively. The margin inherent in these limits provides an appropriate degree of margin to failure at the normal and design speeds. See Reference 1 for additional information on the evaluation of stress in the flywheel assembly.

The flaw size assumed in the evaluation of fracture toughness is described in the AP1000 response to AP600 RAI 251.3.

AP600 RAI 251.7

Section 1A of the SSAR indicates conformance with Regulatory Position 2.f in Regulatory Guide 1.14. Provide information to support this statement.

Westinghouse AP1000 Response to AP600 RAI 251.7

As noted in Subsection 5.4.1.3.6.3 of the AP1000 DCD and in the AP1000 response to AP600 RAI 251.8, the design speed (125 per cent of normal speed) envelopes all expected and postulated overspeed conditions including overspeeds due to postulated pipe ruptures See the AP1000 response to AP600 RAI 251.8 for a discussion of the size of postulated pipe ruptures also. This limitation on the potential overspeed along with the design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of the analysis of critical flywheel failure speeds. The analysis completed for the flywheel stress report evaluates the stress intensity levels at the normal speed and the design speed of 125 percent of normal. The calculated stress levels satisfy the ASME Code, Section III, Subsection NG stress limits. The calculated primary stress levels are less than the recommended stress limits in Positions 4.a. and 4.c. of Standard Review Plan 5.4.1.1 of one-third and two-thirds of yield stress for normal speed and design speed, respectively. See Reference 1 and the AP1000 responses to AP600 RAIs 251.16, 251.17, 251.18, and 251.19 for additional information on the evaluation of stress in the flywheel assembly.

The flywheel structural analysis verifies that the failure modes outlined in Positions 2.c, 2.d, and 2.e of Regulatory Guide 1.14 do not occur at the design speed. The flywheel stress evaluation noted above demonstrates an appropriate margin against these failure modes. In addition, the design of the canned motor pump mitigates the effects of hypothetical failures by these modes, as outlined below.

The AP1000 response to AP600 RAI 251.11 discusses the containment of fragments from a postulated flywheel fracture. The mode of failure, ductile or nonductile, would not alter the capacity of the surrounding pump structure to absorb the energy of the fragments and prevent the generation of missiles from the flywheel assembly.

Regulatory Guide 1.14 defines excessive deformation as any deformation that could cause separation of the flywheel from the shaft. Because of the restriction of the lateral movement of the flywheel assembly by the surrounding structure and axial movement by the thrust bearings, the loss of shrink fit would not be expected to result in substantial movement of the flywheel assembly or significant separation of the assembly from the shaft. This restriction in movement of the flywheel assembly and the adjacent location of the journal bearing to the flywheel assembly minimize the potential for a structural failure of the shaft during a hypothetical overspeed transient sufficient to result in excessive deformation.

Neither separation of the flywheel assembly from the shaft nor structural failure of the shaft would result in a loss of safety-related function of the canned motor pump during an overspeed transient. That safety-related function is the maintenance of the primary pressure boundary. Neither separation of the flywheel assembly nor structural failure of the shaft would degrade the pressure boundary of the pump. The safety-related function of providing flow during coastdown of the pump is not germane during an overspeed transient.

AP600 RAI 251.8

Section 1A of the SSAR indicates conformance with Regulatory Position 2.g in Regulatory Guide 1.14, relating to the flywheel overspeed due to postulated pipe rupture. Section 5.4.1.3.6.3 of the SSAR appears to assume the application of leak-before-break (LBB) for all high-energy piping 10 cm (4 in) in diameter or larger. Since the outcome of the staff's review of the application of LBB to the AP600 design is uncertain, the staff recommends that Westinghouse discuss how the flywheel conforms with RG 1.14 if the criteria of Section 3.6.2 and BTP MEB 3-I is used to determine pipe break size.

Westinghouse AP1000 Response to AP600 RAI 251.8

As was the case for AP600, for AP1000 nominal pipe sizes of 6" and larger are qualified for elimination of post-rupture dynamic analysis through application of leak-before-break criteria. Therefore, the largest break analyzed to determine the dynamic response of the AP1000 reactor coolant pump is that of a 4" pipe (e.g. pressurizer spray line, first stage ADS line).

The overspeed analysis of the AP1000 reactor coolant pump flywheel is based on the design speed of 125 per cent of normal speed. The AP1000 pipe rupture overspeed is expected to be enveloped by the design speed since the reactor coolant main loops and all of the branch line piping with a nominal diameter of 6 inches and greater are being qualified for LBB. The pipe rupture overspeed is expected to be substantially less than any of the calculated critical flywheel failure speeds.

As noted in the AP1000 response to AP600 RAI 251.2, the approach used to demonstrate the safe operation of the flywheel is containment of the fragments from a postulated fracture by the surrounding pump ~~structure~~ **pressure boundary. In the evaluation of the postulated flywheel fracture, the flywheel enclosure is conservatively neglected. This approach maximizes the energy of the flywheel fragments that are assumed to impact the pump pressure boundary.** For a postulated flywheel fracture at the flywheel design speed there is a large amount of margin in the calculated capability of the pump ~~structure~~ **pressure boundary** to contain flywheel fragments. **In the AP600 pump design, the flywheel fragments resulting from a fracture had only 8.5% of the energy required to penetrate the pump pressure boundary. Although the energy of the flywheel fragments in the AP1000 design has increased to 12% of the energy required to penetrate the pump pressure boundary, the conclusion that the fragments from a postulated flywheel fracture will not penetrate the pressure boundary is still valid.** Thus even in the event of a postulated failure of a flywheel during a hypothetical break of a reactor coolant loop pipe, it is not expected that additional breaks in the reactor coolant pressure boundary would be created nor would missiles be generated by the flywheel.

AP600 RAI 251.9

Section 1A of the SSAR indicates that Westinghouse is taking exception to Regulatory Position 4.a in Regulatory Guide 1.14. Propose an alternative to this position with appropriate technical justifications.

Westinghouse AP1000 Response to AP600 RAI 251.9

A spin test is done on the flywheel assembly after the enclosure is welded closed. Inspection of the flywheel inside the assembly is not practical. Because of the density of the uranium, radiographic examination is also not a practical option.

The uranium alloy flywheel is ultrasonically inspected following final machining and prior to assembly of the enclosure around the flywheel. The ultrasonic inspection conforms to the requirements of the ASME Code, Section III, paragraph NB-2574, for ferritic steel castings, including the use of the procedures outlined in SA-609 (ASTM-A-609). See the AP1000 response to AP600 RAI 251.13. Machined surfaces of the uranium flywheel undergo liquid penetrant inspection prior to final assembly. The liquid penetrant inspection conforms with the requirements of the ASME Code, Section III, paragraph NB-2576, including the use of the procedures outlined in SA-165 (ASTM-A-165).

In-process controls during the assembly of the enclosure onto the flywheel are used to provide for the quality of the completed assembly. The spin test of the completed assembly confirms the quality of the flywheel assembly. Since the basis for safe operation of the flywheel assembly is the retention of the fragments from a postulated fracture by the structure of the pump, inspection subsequent to the spin test is not necessary for safe pump operation.

AP600 RAI 251.10

Performance of inservice inspection of the flywheel should be considered. If the ISI procedures in Section 5.4.1.1 of the SRP are not applicable to uranium flywheels, propose alternative inservice inspection procedures with appropriate technical justifications (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.10

Inservice inspection of the uranium alloy flywheel would be very labor intensive and involve significant radiation exposure. Since the surrounding structure of the pump would contain flywheel fragments even in the worse case fracture, inservice inspection would do little to increase the safety of pump operation. The technical justification of no inservice inspection is the analysis that shows that the fragments of a fractured flywheel would not penetrate the pressure boundary of the pump to become missiles (see the AP1000 response to AP600 RAI 251.11). On this basis a flywheel fracture is an operational reliability consideration rather than a safety-related consideration. The use of inspections and in-process controls during fabrication of the flywheel assembly and a spin test of the completed assembly also provide verification of the initial quality of the assembly. The use of vibration monitoring of the pump during operation provides an indication of rotating part stability and thus integrity. This allows any necessary maintenance to be performed as needed for operational reliability.

As noted in the AP1000 response to AP600 RAI 251.2 the design approach to the flywheel in the AP1000 canned motor reactor coolant pump is fundamentally different than that for previous shaft seal reactor coolant pump designs. The canned motor pump design was selected for several safety related and operational reasons. Inherent in the design of a canned motor reactor coolant pump is the location of the flywheel assembly within a pressure housing and the flywheel enclosure in contact with reactor coolant. To make the flywheel readily accessible for an inservice inspection of marginal utility, many advantages of the canned motor pump would have to be foregone. Routine inservice inspection of the flywheel is neither recommended nor advantageous.

AP600 RAI 251.11

Section 1A of the SSAR states that a flywheel rupture will be contained within the stator shell. Provide an analysis and technical justifications supporting this statement.

Westinghouse AP1000 Response to AP600 RAI 251.11

The canned motor reactor coolant pump has an outer shell that comprises the pressure boundary. The shell is analyzed to demonstrate that in the event of a postulated flywheel fracture, the surrounding pump structure is sufficient to prevent missiles from leaving the pump. The analysis considers that portion of the shell, including the flange, and motor end cap around the flywheel assembly between the top and bottom elevations of the assembly as the barrier to missile generation. The structural analysis summary is documented in Reference 1 and is outlined below.

The analysis of the capacity of the surrounding pump structure to contain the fragments of a postulated flywheel failure is done using the energy absorption equations of Hagg and Sankey (Hagg, A. C., and Sankey, G. O., "The Containment of Disk Burst and Fragments by Cylindrical Shells," ASME Journal for Power, April 1974, pp. 114-123). The containment of missile-like metal disk fragments is by a two-stage process. Stage 1 involves inelastic impact and transfer of momentum to include an effective target mass. To show that the fragments do not perforate the surrounding structure, the energy dissipated in plastic compression and shear strain and the local impact area must be sufficient to account for the loss in kinetic energy of the system. For the nonperforation case the process enters Stage 2, which involves dissipation of energy in plastic tension strain over extended volumes of shell material. For containment, the energy dissipated in plastic strain in Stage 2 must account for the residual kinetic energy on the system. In predictive calculations it is more conservative to consider Stage 2.

For the AP1000 reactor coolant pump analysis, the uranium insert in the flywheel assembly is assumed to fracture at the design speed of 125 percent of normal speed. The worst-case scenario of fragment size and number was derived analytically, using methods from Hagg and Sankey to determine the mass and velocity combination that would produce the most severe impact on the surrounding pressure boundary components. The following conservative assumptions are also made:

1. End plates and welds of the flywheel enclosure and the coolant surrounding the flywheel assembly have negligible energy-absorbing capability.
2. Only the mass in the stator shell and flange and the motor end cap between the elevation of the top and bottom of the flywheel assembly are considered to absorb energy.
3. Closure bolts and joint effects were not considered to be affected.
4. The minimum material properties were used.

The analysis results show that the fragments impact the surrounding pump structure with a kinetic energy of less than 15 percent of the tensile energy-absorbing capability of the surrounding pump structure. Thus the components around the flywheel contain the flywheel fragments using only a small portion of the energy-absorbing capability. The energy absorbed by the flywheel enclosure is small compared to the surrounding pump structure and was not considered in the calculation of flywheel fragment containment within the pump pressure boundary.

AP600 RAI 251.12

Section 1A of the SSAR indicates that a "small" flywheel rupture or leak in the enclosure will not result in stresses in the pressure boundary to cause a break. Provide information to clarify what is the intent of the term "small" flywheel rupture. The staff is concerned with the rupture of the flywheel into large fragments of high energy.

Westinghouse AP1000 Response to AP600 RAI 251.12

The canned motor pump design is evaluated for a spectrum of postulated uranium flywheel fractures. A fracture that ruptures the flywheel enclosure is bounded by the analysis of the worst-case fracture (Reference 1) that shows that the fragments are contained as noted in AP1000 DCD section 5.4.1.3.6.3. A fracture that deforms the enclosure enough to bring it in contact with the surrounding structure is bounded by the analysis described in AP1000 DCD section 5.4.1.3.6.2. A small fracture in the context of the DCD discussion is one that may unbalance the assembly, but any resulting fragment is contained by the enclosure without sufficient deformation to result in interference with the surrounding structure. The discussion of these faults on the low end of the spectrum are included for completeness of the discussion of postulated flywheel fractures.

AP600 RAI 251.13

Section 5.4.1.3.6.3 of the SSAR indicates that ultrasonic inspection of the uranium following final machining will be based on ASTM A388 as modified for uranium. Identify any modifications to the application of ASTM A388 to the AP600 design with appropriate technical justifications. In addition, demonstrate that this preservice inspection is equivalent to that in Section III of the ASME Code.

Westinghouse AP1000 Response to AP600 RAI 251.13

ASTM A388, which is a standard for use of ultrasonic inspections on steel forgings, is not given as the standard for ultrasonic inspection of the uranium following final machining in AP1000 DCD section 5.4.1.3.6.3. ASTM A609, which is a standard for use of ultrasonic inspections on ferritic steel castings, will be used as the standard for ultrasonic inspection of the uranium flywheel. Changes to the practices specified in the standard to account for use on uranium include the use of uranium reference blocks and potential additional restrictions on the couplants used. The size and frequency of transducers may also be different than the standard, although the inspection of a prototype flywheel casting was done with a transducer size and frequency in the range designated in the standard. Areas of the standard that are not dependent on the type of material inspected, such as personnel qualification requirements, surface conditions, procedures, and data reporting should not have to be modified. See the AP1000 response for AP600 RAI 251.9 for additional discussion of inspection of the uranium flywheel.

It is not the intent that the inspection of the uranium alloy flywheel be equivalent in every respect to inspections required of components built to the requirements of the ASME Code, Section III. The requirements for the flywheel are chosen to provide high operational reliability. There are no pressure boundary functions associated with the flywheel assembly that require the use of the ASME Code.

AP600 RAI 251.14

Demonstrate that the construction of the flywheel enclosure meets Section III of the ASME Code, including inspection (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.14

Since the flywheel enclosure is not a pressure boundary and is not relied upon to contain fragments from a postulated flywheel fracture, there is no requirement to meet the requirements of the ASME Code, Section III for construction of the enclosure. Additionally, the enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The function of the enclosure is to isolate the uranium alloy from the reactor coolant circulating in the reactor coolant pump. A leak in the enclosure could result in an out-of-balance condition for the flywheel assembly or, over the long term, the possible introduction of depleted uranium into the reactor coolant. Neither of these events represents a catastrophic failure and both would be addressed by other systems. Sensors in the pump detect vibration of the pump and the chemical and volume control system includes provisions to reduce contaminants in the reactor coolant. The uranium would be detected by periodic sampling of the reactor coolant by the primary sampling system.

The ASME Code, Section III criteria for structural welds are used to establish welding requirements and inspection requirements for the enclosure. As noted in the AP1000 DCD section 5.4.1.3.6.3, the welds are subject to dye penetrant and ultrasonic tests. The ASME Code Subsection NG stress limit criteria are used as guidelines to evaluate the stress in the enclosure components and the flexible welds for normal and design speeds. The use of the ASME Code, Section III to establish design, fabrication, and inspection requirements was selected to provide operational reliability and availability.

AP600 RAI 251.15

Demonstrate that the design overspeed of the flywheel is at least 10% above the highest anticipated overspeed (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.15

The requirement for the AP1000 is that the design speed (125 percent of normal speed) be greater than or equal to anticipated overspeed conditions due to electrical faults and overspeed conditions due to postulated pipe breaks. Anticipated overspeed conditions are those due to electrical faults including turbine overspeed events. Because of design of the turbine control system (see AP1000 DCD section 10.2.2), reactor coolant pump overspeed resulting from an electrical fault is expected to be less than the design speed. See the AP1000 response to AP600 RAI 251.8 for a discussion of flywheel overspeed due to postulated pipe rupture.

Since the basis for safe operation of the pump with respect to flywheel integrity is the containment of flywheel fragments by the pump structure rather than the prevention of fracture (see the AP1000 responses to AP600 RAIs 251.2 and 251.11), a 10% margin between calculated overspeed and the design speed is not necessary to assure safe operation.

AP600 RAI 251.16

Show that the combined stresses for the uranium flywheel at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, is less than 1/3 of the minimum specified yield strength (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.16

The flywheel structural analysis verifies that the primary stresses in the uranium due to centrifugal forces at the normal operating speed are less than one-third of the minimum yield strength. The combination of primary and secondary stresses is evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors (Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at normal operating speed. See Reference 1 for analysis details.

AP600 RAI 251.17

Discuss how the limit in AP600 RAI 251.16 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.17

The evaluation of the flywheel enclosure does not use the limit of one-third of minimum yield strength as a criterion for normal operating speed. The flywheel enclosure **only** prevents contact of coolant with the uranium flywheel, **and is not considered to be a "reactor coolant pressure boundary"**. No credit is taken in the analysis of the flywheel missile generation for the retention of the fragments by the enclosure, and the flywheel enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The evaluation of the stress in the flywheel enclosure components and the flexible welds connecting the components for normal and design speeds uses the criteria in Subsection NG of the ASME Code as a guideline. The ASME Code limits are satisfied for analyzed stresses of the flywheel enclosure at the normal operating speed (see Reference 1).

The AP600 and AP1000 flywheel enclosure designs are very similar. **The radial thickness of the two enclosures is approximately the same.** The stresses **calculated** in the AP1000 flywheel enclosure **shell components** at normal operating speed are similar to those in the AP600 flywheel enclosure. The maximum radial displacement ~~at~~ **of** the AP1000 flywheel enclosure welds is also **very** similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds ~~were~~ **have been** shown to meet the ASME Code limits during operation at the normal operating speed. Therefore, **because of the similarities in enclosure configuration, operating conditions, enclosure shell stresses, radial thickness, and maximum weld area radial displacement between the AP600 and AP1000,** it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during normal operating conditions. **Stress analyses of the flywheel enclosure welds under normal operating speed conditions will be performed during detailed design of the AP1000 reactor coolant pump.**

A discussion of the flywheel enclosure weld stresses will be included in Revision 1 of Reference 1.

AP600 RAI 251.18

Show that the combined stresses for the uranium flywheel at the design overspeed, due to centrifugal forces and the interference fit, is less than 2/3 of the minimum specified yield strength (Section 5.4.1).

Westinghouse AP1000 Response to AP600 251.18

The flywheel structural analysis verifies that the combined stresses in the uranium flywheel due to centrifugal forces and the interference fit at the design speed of 125 percent of normal speed are less than the limit of two-thirds of the minimum yield strength. The combination of primary and secondary stresses is also evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors (Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at the design speed (see Reference 1).

AP600 RAI 251.19

Discuss how the limit in AP600 RAI 251.18 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.19

The evaluation of the flywheel enclosure does not use the limit of two-thirds of minimum yield strength as a criterion for design speed conditions. The criteria in the ASME Code, Section III, Subsection NG, are used as a guideline for stress limits. The ASME Code limits are satisfied for analyzed stresses in the flywheel enclosure at the design speed. See Reference 1 and the AP1000 response for AP600 RAI 251.17.

The AP600 and AP1000 flywheel enclosure designs are very similar. **The radial thickness of the two enclosures is approximately the same.** The stresses **calculated** in the AP1000 flywheel enclosure **shell components** at design speed are similar to those in the AP600 flywheel enclosure. The maximum radial displacement ~~at~~ **of** the AP1000 flywheel enclosure welds is also **very** similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds ~~were~~ **have been** shown to meet the ASME Code limits during operation at the design speed. Therefore, **because of the similarities in enclosure configuration, operating conditions, enclosure shell stresses, radial thickness, and maximum weld area radial displacement,** it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during operation at the design speed. **Stress analyses of the flywheel enclosure welds under design speed conditions will be performed during detailed design of the AP1000 reactor coolant pump.**

A discussion of the flywheel enclosure weld stresses will be included in Revision 1 of Reference 1.

AP600 RAI 251.20

Demonstrate that the shaft and the bearings supporting the flywheel will be able to withstand any combination of loads from normal operation, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake (Section 54.1).

Westinghouse AP1000 Response to AP600 RAI 251.20

The containment of fragments from a postulated fracture of the flywheel is not dependent on the support of the shaft and flywheel by the bearings. Postulated failures of the bearings and shaft would result in the rotating assembly being slowed to a stop. Bearing or shaft failures would be indicated by vibration or temperature sensors. A postulated failure of a bearing or shaft that allowed excessive lateral movement would result in contact between one or more rotating parts and the surrounding structure thereby slowing the rotation. A postulated failure of a bearing or shaft that allowed excessive axial movement would not remove the restriction provided by the pump internals, including the impeller and suction adapter. Thus a failure that would allow axial movement would not result in significant movement of the flywheel assembly.

Based on this information, the effect of these loads on the shaft and bearings is of interest with regard to operational reliability but not with regard to safe operation. The shaft and bearing supports are evaluated for loads due to seismic events.

AP600 RAI 251.21

Identify the materials for the flywheel enclosure and associated welds. Provide technical justification to show that the flywheel enclosure and associated welds are resistant to stress corrosion cracking, especially if Inconel 600 or 182 materials will be used (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.21

The material of construction of the flywheel assembly enclosure is nickel-chromium-iron Alloy 690. The material for the welding filler metal is nickel-chromium-iron Alloy 52. Since the coolant surrounding the flywheel assemblies is normally at a relatively low temperature (approximately 165 F) and Alloy 690 has shown good resistance to stress corrosion cracking in applications at the higher reactor coolant system temperatures, primary water stress corrosion cracking in the flywheel assembly would not be expected.

AP600 RAI 251.22

Demonstrate that the uranium flywheel is resistant to stress corrosion cracking or other potential degradation mechanisms in a reactor coolant environment (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.22

The uranium alloy flywheel is sealed in the nickel-chromium-iron alloy enclosure and is not in contact with the reactor coolant or other fluid. See the AP1000 response to AP600 RAI 251.14 for additional discussion of the enclosure flexible welds. The uranium alloy flywheel is heat treated by solution annealing in a vacuum furnace and slowly cooled. This heat treatment minimizes the potential for residual stresses. The heat treatment process also removes hydrogen from the material to reduce the potential for hydrogen embrittlement. Since the depleted uranium alloy is not in contact with reactor coolant or any other fluid and operates at a relatively low temperature, degradation of the material is not expected.

AP600 RAI 251.23

Table 5.4-2 in the SSAR lists the flywheel material specifications. Provide the technical basis for these specifications.

Westinghouse AP1000 Response to AP600 RAI 251.23

The material specification information including ultimate tensile strength and yield strength provided in AP1000 DCD Table 5.4-2 is based on material testing by the material supplier. The composition of the alloys, including the limits on the constituent elements, is also based on the experience of the material supplier. The production of the uranium flywheel is controlled to minimize the formation of voids or other defects. The heat treatment process is controlled to provide the required material properties. See the AP1000 response to AP600 RAI 251.22 for a discussion of the heat treatment. Quality assurance testing of the material verifies that the material supplied conforms to the material specification. Ultrasonic and liquid penetrant inspections are performed on the uranium flywheel to verify the absence of unacceptable defects. See the AP1000 responses to AP600 RAIs 251.9 and 251.13 for a discussion of the ultrasonic and liquid penetrant inspections.

Design Control Document (DCD) Revision:

See the response to AP1000 RAI 440.040. (These changes were incorporated into DCD Revision 3.)

From DCD Revision 3 page 1.6-12:

Table 1.6-1 (Sheet 11 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
5.2	WCAP-8324-A	Control of Delta Ferrite in Austenitic Stainless Steel Weldments, June 1975
	WCAP-8693	Delta Ferrite in Production Austenitic Stainless Steel Weldments, January 1976
5.3	WCAP-15557	Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology, August 2000
5.4	**WCAP-15994-P (P)** **WCAP-15994-NP**	**Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel**
6.2	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods, March 1973
	WCAP-8264-P-A (P) WCAP-8312-A	Westinghouse Mass and Energy Release Data for Containment Design, June 1975 (P), August 1975 (Non-P)
	WCAP-10325 (P)	Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version, May 1983
	WCAP-8822 (P) WCAP-8860	Mass and Energy Releases Following A Steam Line Rupture, September 1976
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, June 1984

(P) Denotes Document is Proprietary

WCAP-14234	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, June 1997
WCAP-15846 (P) WCAP-15862	WGOTHIC Application to AP1000, Revision 0, April 2002

From DCD Revision 3 page 5.4-6:

5.4.1.3.6.3 Flywheel Integrity

The canned-motor reactor coolant pump in the AP1000 complies with the requirement of General Design Criterion (GDC) Number 4. That Criterion states that components important to safety be protected against the effects of missiles.

The flywheel assemblies are located within and surrounded by the heavy walls of the motor end closure, casing, thermal barrier flange, stator shell, or main flange. In the event of a postulated worst-case flywheel assembly failure, the surrounding structure can, by a large margin, contain the energy of the fragments without causing a rupture of the pressure boundary. The analysis **in Reference 10** of the capacity of the housing to contain the fragments of the flywheel is done using the energy absorption equations of Hagg and Sankey (Reference 2).

From DCD Revision 3 page 5.4-76:

5.4.16 References

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

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

3. ASTM-A-609-91, Standard Specification for Longitudinal Beam Ultrasonic Inspection of Carbon and Low-alloy Steel Castings.

4. ASTM-E-165-95, Practice for Liquid Penetrant Inspection Method.

5. ANSI/ANS-5.1-1994, "Decay Heat Power in Light Water Reactors."

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

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

8. QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.

9. ANSI B16.34-1996, Valves - Flanged and Buttwelding End.

10. **WCAP-15994-P (Proprietary) Revision 1, and WCAP-15994-NP (Non-Proprietary) Revision 1, "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel", March 2003.**

Changes to WCAP-15994:

The changes shown in the attached marked-up pages will be incorporated into Revision 1 of WCAP-15994.

PRA Revision:

None

TABLE OF CONTENTS

- Uranium insert normal operation inner radius hoop stress

$$\sigma_{\theta\theta}(R_{OI})_U = \frac{0.688\frac{lb}{in^3}\left(188.5\frac{rad}{sec}\right)^2}{4\times 386.4\frac{in}{sec^2}}\left[(3+v)(14.50in)^2+(1-v)(8.00in)^2\right]=11685\,psi$$

- Uranium insert design condition inner radius hoop stress

$$\sigma_{\theta\theta}(R_{OI})_U = \frac{0.688\frac{lb}{in^3}\left(1.25\times 188.5\frac{rad}{sec}\right)^2}{4\times 386.4\frac{in}{sec^2}}\left[(3+v)(14.50in)^2+(1-v)(8.00in)^2\right]=18255\,psi$$

The uranium insert is Depleted Uranium Alloy U-2Mo, and the yield stress for this material is 55,000 psi. Per the design specification, the $1/3S_y$ is 18,330 psi and $2/3S_y$ is 36,670 psi. For normal operation at a rotational speed of 1800 rpm, the maximum primary hoop stress is 11,685 psi, which is less than 18,330 psi. Additionally, for a design rotational speed of 2250 rpm, the maximum primary hoop stress is 18,255 psi, which is less than 36,670 psi. Since the rotational stresses for the uranium flywheel are less than the prescribed allowable stresses, the requirements of the design specification are satisfied.

5.1.4 Concentric Ring Elastic Hoop Stresses

In Table 5-2, the hoop stresses at the inner and outer diameter of each of the concentric rings in the model are presented for assembly conditions, 1800 rpm of rotation, operating pressure, and a uniform temperature of 165°F, as well as for assembly conditions and 125-percent overspeed at 70°F. From Table 5-2, it is noted that the hoop stresses of 18.71 ksi and 21.34 ksi in the jacket are less than the yield stresses for Alloy 690, which are 32.5 ksi at 165°F and 35 ksi at 70°F.

INSERT A →

Insert A

5.1.5 Flywheel Enclosure Welds

Since the flywheel enclosure is not considered to be a "reactor coolant pressure boundary", the stress analyses of the enclosure welds has been deferred until the detailed design phase of the AP1000 reactor coolant pump. However, the AP600 and AP1000 flywheel enclosure designs are very similar and the radial thickness of the two enclosures is approximately the same. The stresses calculated in the AP1000 flywheel enclosure shell components at both normal operating and design speeds are similar to those in the AP600 flywheel enclosure. The maximum radial displacement of the AP1000 flywheel enclosure welds is also very similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds have been shown to meet the ASME Code limits during operation at both normal operating and design speeds (Reference 2). Therefore, because of the similarities in enclosure configuration, operating conditions, enclosure shell stresses, radial thickness, and maximum weld area radial displacement, it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during operation at both normal and design speeds.

Table 5-2 AP1000 Motor Cavity Flywheel Hoop Stresses

	Shaft		Inner Enclosure		Uranium		Jacket	
	Inner Radius S_{tSI} (psi)	Outer Radius S_{tSO} (psi)	Inner Radius S_{tII} (psi)	Outer Radius S_{tIO} (psi)	Inner Radius S_{tUI} (psi)	Outer Radius S_{tUO} (psi)	Inner Radius S_{tJI} (psi)	Outer Radius S_{tJO} (psi)
Steady-State Operation								
Assembly(1)	-13483	-13483	-5634	-7826	20338	9338	16996	16705
2335 psi	-2500	-2500	-2566	-2547	-2027	-2198	-2809	-2801
1800 rpm	5014	4842	4971	4700	3713	1978	2781	2704
165°F	2857	2857	-1961	-616	-2920	-1379	1745	1715
Total	-8111	-8284	-5191	-6289	19105	7740	18713	18324
125-Percent Overspeed at 70°F								
Assembly(1)	-13483	-13483	-5634	-7826	20338	9338	16996	16705
0 psi	0	0	0	0	0	0	0	0
2250 rpm	7835	7565	7767	7344	5802	3091	4346	4225
70°F	0	0	0	0	0	0	0	0
Total	-5648	-5918	2133	-481	26140	12429	21342	20931

1. Axial Advance, Δ = [0.115 inches](a,c), Insert/Inner Enclosure Radial Shrink σ_{SUI} = [0.010 inches](a,c), Jacket/Insert Radial Shrink σ_{SUJ} = [0.006 inches](a,c)

5.1.~~5~~6 Fracture Mechanics of Uranium Insert

An estimate of the critical flaw sizes in the uranium insert is made using the approach from section 6.4 of WCAP-13734 (Reference 2). For the present calculations, the sudden rupture of the uranium insert is governed by the critical Mode I (tensile) fracture toughness of the material, namely $K_{IC} = 50$ ksi. in$^{1/2}$. The hoop stress distribution across the uranium insert and the critical crack sizes are presented in Table 5-3 for steady-state operation and assembly plus 125-percent overspeed. These crack sizes are estimated using Version 3.0 of NASCRAC (NASA Crack Analysis Code by Failure Analysis Associates, Inc., of Palo Alto, California). For these estimates, Case 205 represents a full-length axial crack on the inner diameter of a hollow cylinder. Additionally, Case 704 is a semi-elliptical axial surface flaw in a cylinder, and for this case, flaws with aspect ratios of 1:1 and 3:1 are considered. From Table 5-3, the minimum flaw size is [1.075 inches](a,c) for assembly + 125-percent overspeed. These results can be used to support fracture toughness and inspection requirements for the uranium alloy material.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 252.002 (Revision 1 Response)

Question:

Paragraph 5.2.3.2.2 on page 5.2-11 in the 2nd paragraph discusses safe ends. What is the purpose of these safe ends? If the purpose of the safe ends is to protect the austenitic stainless steel from sensitization, then an A-8 weld, which is austenitic stainless steel, will become sensitized when the component is postweld heat treated at 1100°F. Please address this concern as part of your response, if applicable. (Section 5.2.3)

Follow-On Comment:

The response is not acceptable since A-8 welds include austenitic stainless steels such as 304 and 316 types which could become sensitized during postweld heat treatment.

Westinghouse Response (Revision 1):

The purpose of the safe ends is to protect austenitic stainless steel from being heat treated during field installation, which may cause sensitization. Based on the experiences of the safe-ends for the current reactors, **the safe-ends are welded to the buttering after** postweld heat treatment of the ~~safe-ends~~**buttering** at the fabrication shop. ~~does not cause a sensitization concern.~~

DCD section 5.2.3.2.2 will be revised to delete reference to A-8 welds.

DCD Table 5.2-1 will be revised to delete type ERNiCr-3 for cladding and buttering from the reactor vessel components materials specification.

Design Control Document (DCD) Revision:

~~None~~

From DCD Revision 3, page 5.2-12:

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

Ferritic low-alloy and carbon steels used in principal pressure-retaining applications have corrosion-resistant cladding on surfaces exposed to the reactor coolant. The corrosion resistance of the cladding material is at least equivalent to the corrosion resistance of Types 304 and 316 austenitic stainless steel alloys or nickel-chromium-iron alloy, martensitic stainless steel, and precipitation-hardened stainless steel. These clad materials may be subjected to the ASME Code-required postweld heat treatment for ferritic base materials.

Ferritic low-alloy and carbon steel nozzles have safe ends of ~~either~~ stainless steel-wrought materials~~,~~ ~~stainless steel weld metal analysis A-8, or~~ **welded to** nickel-chromium-iron alloy-weld metal F-number 43 **buttering**. The ~~latter buttering material requires further~~ safe ending ~~with austenitic stainless steel base material~~ **is welded to the F 43 buttering** after completion of ~~the~~ postweld heat treatment **of the buttering** when the nozzle is larger than a 4-inch nominal inside diameter and/or the wall thickness is greater than 0.531 inch.

From DCD Revision 3 page 5.2-32:

Table 5.2-1 (Sheet 1 of 4)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS SPECIFICATIONS

Component	Material	Class, Grade, or Type
Reactor Vessel Components		
Head plates (other than core region)	SA-533 or SA-508	GR B, CL 1 or CL 3
Shell courses	SA-508	CL 3
Shell, flange, and nozzle forgings	SA-508	CL 3
Nozzle safe ends	SA-182	F316LN
Appurtenances to the control rod drive mechanism (CRDM)	SB-167 or SA-182	TP690 or F304LN, F316LN
Instrumentation tube appurtenances, upper head	SB-167 or SA-182, SA312, SA376	TP690 or F304LN, F316LN
Closure studs	SA-540	GR B23 or GR B24, CL 3
Monitor tubes and vent pipe	SA-312 or SA-376 or SB-166, SB-167	TP304LN, TP316LN or TP690
Cladding and buttering	SFA 5.4, 5.9, 5.11, and 5.14	308L, 309L, ENiCrFe-7, or ERNiCrFe-7~~, or ERNiCr-3~~

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.014 (Response Revision 1)

Question:

Provide quantitative technical analysis to support that the AP1000 can meet the present ATWS Rule that requires an ATWS mitigation system actuation circuitry (AMSAC), which automatically initiate the auxiliary feedwater system and initiate a turbine trip under conditions indicative of an ATWS, and meet the basis of the rule, i.e., the reactor vessel pressure will exceed 3200 pounds-per-square inch (absolute) (psia) for no more than five percent of the cycle time. The analysis should be performed for all applicable non-loss-of-coolant accident (LOCA) transients in order to identify the limiting ATWS case. Discuss the methods used and verify that the methods are acceptable. Also, justify that the assumptions for the applicable ATWS analyses are adequate as they relate to input parameters such as the initial power level, moderator temperature coefficient (MTC), pressurizer safety and relief valves capacity, reactor coolant system (RCS) volume, steam generator pressure, passive residual heat removal system (PRHR) heat transfer capacity and its actuation delay time, and the AMSAC setpont to trip the turbine and initiate the PRHR. Also include a discussion and applicable values of the unfavorable exposure time for the MTC (in accordance with the NUREG-0460 guidance for the newer plant design) assumed in the analyses.

Westinghouse Response:

As discussed in Section 15.8 of the DCD, for Westinghouse plants, the ATWS rule (10 CFR 50.62) requires the installation of ATWS mitigation systems actuation circuitry (AMSAC), which consists of circuitry separate from the reactor protection system, to trip the turbine and initiate decay heat removal.

The basis for the ATWS rule requirements, as outlined in SECY-83-293 (Reference 1), is to reduce the risk of core damage because of ATWS to less than 10^{-5} per reactor year.
The AP1000 includes a diverse actuation system, which provides the AMSAC protection features mandated for Westinghouse plants by 10 CFR 50.62, plus a diverse reactor scram (see Section 7.7 of the DCD). Thus, the AP1000 meets the ATWS rule.

Similar to the AP600, the AP1000 relies on the PRHR in lieu of an auxiliary or emergency feedwater system as its safety-related method of removing decay heat. Westinghouse has requested exemption from the part of the ATWS regulation, 10 CFR 50.62(c)(1), that requires auxiliary or emergency feedwater as an alternate system for decay heat removal during an ATWS event. The staff concluded for AP600 that Westinghouse had met the intent of the ATWS rule by relying on the PRHR system to remove the decay heat, and meets the underlying purpose of the rule. This exemption was reviewed by the NRC during the pre-certification review, and found to be applicable to the AP1000. By separate letter, Westinghouse has requested this exemption for the AP1000.

In the course of the AP600 review, the staff asked Westinghouse to submit an analysis demonstrating that the AP600 ATWS response is within the bounds considered by the staff during its deliberations leading to the ATWS rule. For Westinghouse plants, the complete loss of normal feedwater (LONF) event is the limiting event for the ATWS analysis because it produces the maximum RCS pressure. During the AP600 Design Certification review, Westinghouse was required to demonstrate that the LONF was still the limiting event, considering the differences resulting from using the PRHR heat exchanger as the safety-related decay heat removal system.

In addition, as required by the staff, Westinghouse showed that the unfavorable exposure time (UET), given the cycle design (including the moderator temperature coefficient [MTC]), is less than 5 percent, or equivalently, that the ATWS pressure limit is met for at least 95 percent of the cycle. The UET is the time during the cycle when reactivity feedback is not sufficient to maintain pressure under 3200 psig for a given reactor state.

In Westinghouse letter DCP/NRC 1240, dated January 30, 1998, Westinghouse provided the staff a complete set of analyses to demonstrate the limiting ATWS scenario for the AP600 passive plant. The results of the analyses presented in this report demonstrated that, as was the case for the current Westinghouse PWRs, the LONF event was also the limiting ATWS scenario for the passive AP600. Based on the similarities of the design of the AP600 and AP1000, including their reliance on passive safety features such as the PRHR heat exchanger to mitigate the consequences of the ATWS, Westinghouse performed an ATWS analysis of the LONF event for the AP1000 Probabilistic Risk Assessment. The analysis presented in Appendix A of the PRA demonstrates that the peak AP1000 RCS pressure is less than 3200 psig with a UET of 0. This is a result of the operation of the passive safety systems, as well as the lower MTC associated with the lower core boron concentration of the AP1000. PRA Appendix A provides a discussion of the AP1000 ATWS analysis assumptions.

NRC Additional Comments:

1. Why is the loss of normal feedwater considered the worst case for anticipated transient without scram (ATWS) (as it was for AP600)?

2. Could you provide any quantitative analysis supporting your conclusion regarding the effects resulting from the removal of the three conservatisms for the first cycle unfavorable exposure time (UET) evaluation?

3. On Page A-25 the comment is made that during an ATWS, the maximum allowable pressure may be exceeded. The staff requested supplemental information that would shed light on the extent that the pressure is exceeded.

Westinghouse Additional Response:

1. The analyses reported in Chapter 15 of the DCD show that AP1000 plant behavior following design basis accident is similar to that of the AP600 plant. The protection and safety monitoring system (PMS) and Diverse Actuation System (DAS) logic and the passive safety systems are functionally the same. In addition, the AP1000 plant has been designed following the same design criteria adopted for the AP600. In particular, the sizing of the pressurizer and steam generator safety valves follow the same approach and available inventory in the secondary side provides more or less the same full power seconds. Attachment 440.014-1 provides a detailed analytical justification of the most important contributors to the ATWS events and provides a complete discussion on the AP1000 ATWS initiators to justify the choice of the LNFW as the most limiting ATWS initiator.
   Note that the analyses in Attachment 440.014-1 do not include the DAS logic change discussed in 2.) below but this would not change the conclusion.

2. Based on a series of AP1000 specific sensitivity studies performed to address this RAI response, the diverse actuation system (DAS) actuation logic is revised to improve behavior of the AP1000 plant following ATWS events. The new logic actuates the CMTs and RCP trip on low wide-range steam generator level. It should be noted that RCP trip and CMTs actuation were already implemented following a DAS low pressurizer level signal. Together with the implementation of a new DAS logic, an additional change has been implemented in the plant control system (PLS) such that the plant control system isolates the steam dump system whenever the SG water level drops below the Low SG Water level – wide range setpoint.

   Additional ATWS analyses performed with the new DAS ATWS protection logic demonstrates that the RCS pressure is maintained below the pressure limit of 3200 psia even assuming a moderator temperature coefficient equal to –5 pcm/°F. This moderator temperature coefficient envelops 100% of the AP1000 core life and hence it can be concluded that the UET for the AP1000 is zero. The Sequence of Events table for the limiting ATWS analysis case is included as Table 440.014-1. Figure 440.014-1 through 4 provide the analysis results from this case.
   Based on the revised DAS actuation logic, further analysis of conservatism is not needed because the AP1000 UET is now zero.

   The PRA and DCD will be updated to revise the new DAS actuation logic and revised ATWS analysis results.

3. Based on the revised DAS actuation logic, the RCS pressure limit of 3200 psia is not exceeded.

TABLE 440.014-1

SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER

(MODERATOR TEMPERATURE COEFFICIENT EQUAL TO –5 PCM/°F)

Time (Seconds)	Event
0-4	Loss of feedwater
50	PRZ safety valves open (s)
62.1	Turbine bypass valves fully open
64.1	Turbine trip (TxT) (s)
66.1	RCP trip (s)
n.c.	STS actuation (s)
74.1	PRHR on (s)
74.1	CMT on (s)
93	PRZ overfills (s)
141	Maximum RCS pressure of 2818 psia
230	PRZ safety valves close

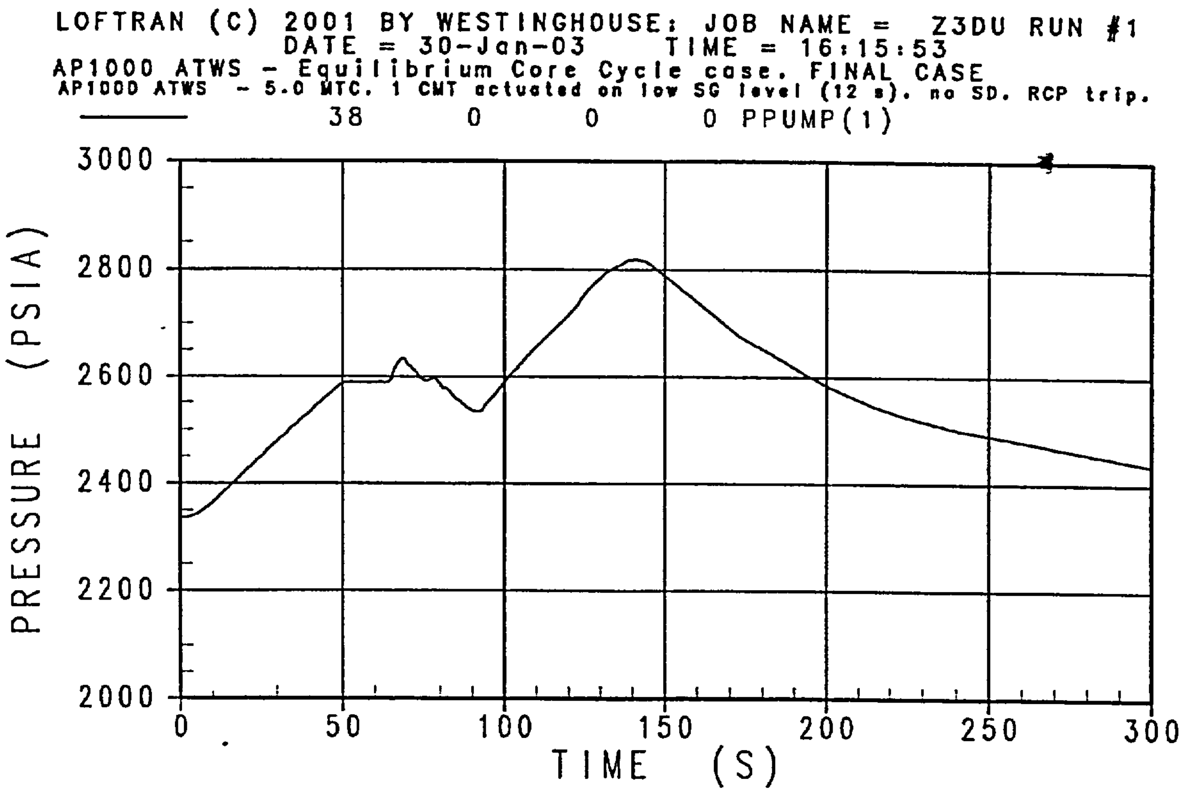


Figure 440.014-1 ATWS Loss of Normal Feedwater, MTC equal to –5 pcm/°F, RCS Pressure (psia) vs. Time

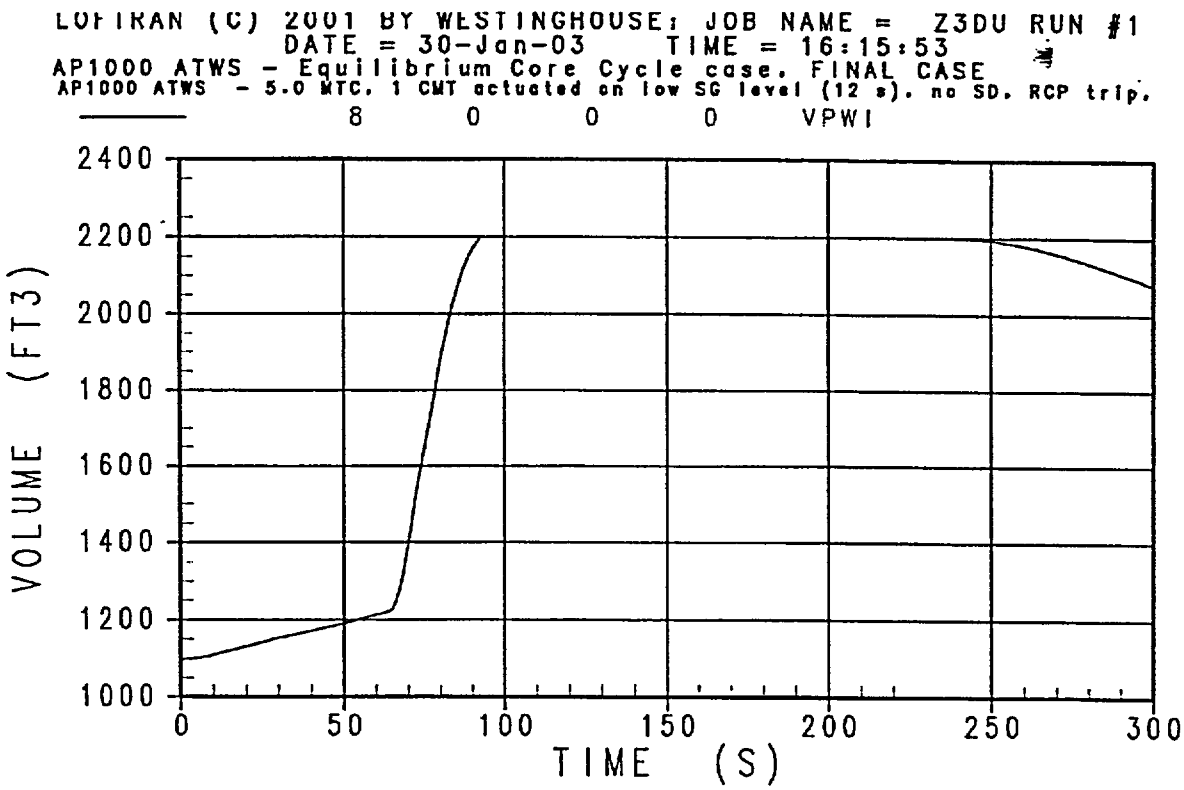


Figure 440.014-2 ATWS Loss of Normal Feedwater, MTC equal to –5 pcm/°F, Pressurizer Water Volume (ft^3) vs. Time

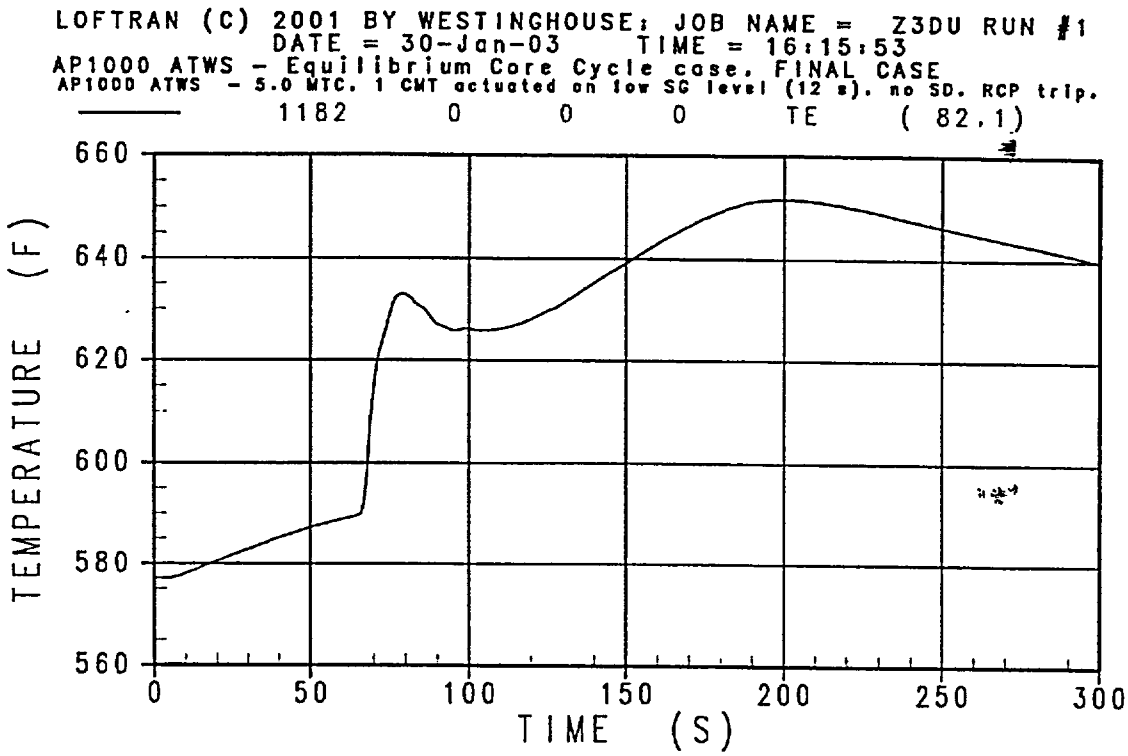


Figure 440.014-3 ATWS Loss of Normal Feedwater, MTC equal to –5 pcm/°F, Core Average Temperature (°F) vs. Time

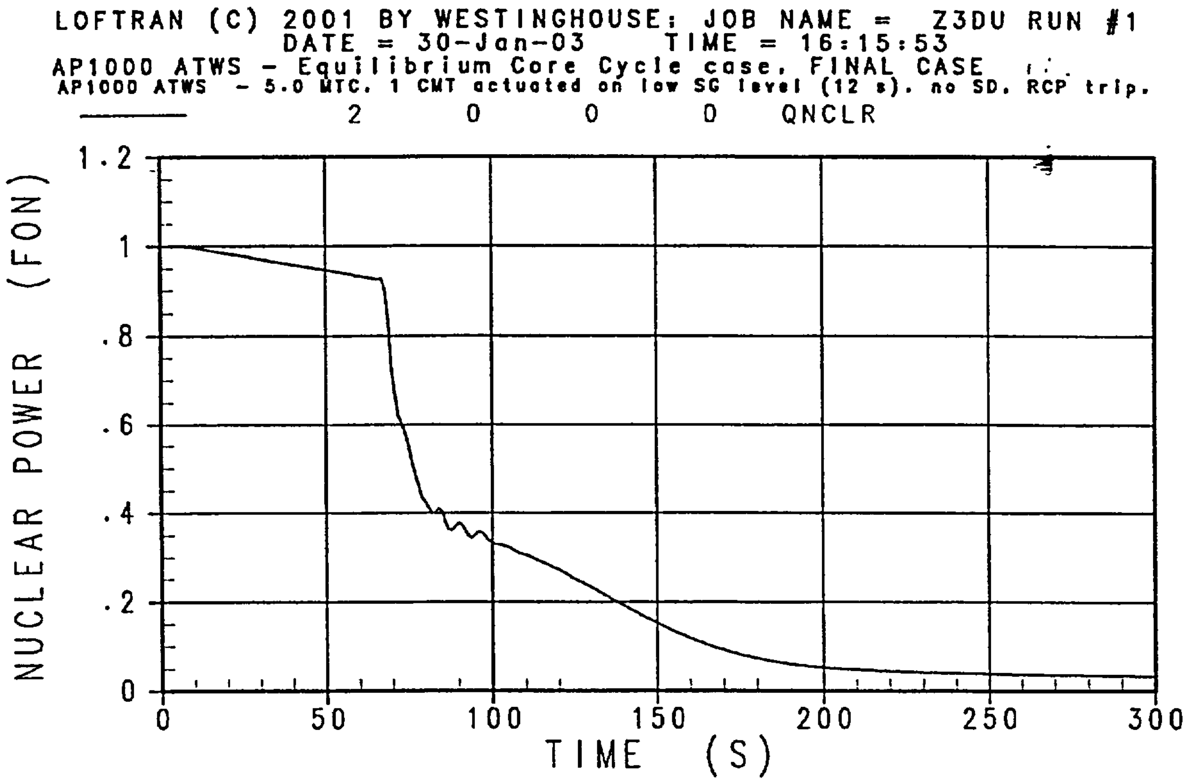


Figure 440.014-4 ATWS Loss of Normal Feedwater, MTC equal to –5 pcm/°F, Nuclear Power vs. Time

Design Control Document (DCD) Revision:

DCD Section 7.7.1.9.1 is revised as follows:

7.7.1.9 Steam Dump Control System

The AP1000 is designed to sustain a 100-percent load rejection, or a turbine trip from 100-percent power, without generating a reactor trip, requiring atmospheric steam relief, or actuating a pressurizer or steam generator safety valve. The automatic steam dump control system, in conjunction with other control systems, is provided to accommodate this abnormal load rejection and to reduce the effects of the transient imposed on the reactor coolant system. By bypassing main steam to the condenser, an artificial load is maintained on the primary system. This artificial load makes up the difference between the reactor power and the turbine load for load rejections and turbine trips. It also removes latent and decay heat following a reactor trip.

The steam dump system is sized to pass 40 percent of nominal steam flow. This capacity, in conjunction with the performance of the reactor power control system, is sufficient to handle reactor trips from any power level, turbine trips from 50-percent power or less, and load rejections equivalent to a step load decrease of 50 percent or less of rated load. For turbine trips initiated above 50-percent power, or load rejections greater than the equivalent of a 50-percent step, the steam dump operates in conjunction with the rapid power reduction system described in subsection 7.7.1.10 to meet the performance described in the previous paragraph.

The steam dump control system has two main modes of operation:

- The Tavg mode uses the difference between measured auctioneered loop Tavg and a reference temperature derived from turbine first-stage impulse pressure, to generate a steam dump demand signal. This mode is largely used for at-power transients requiring steam dump, such as load rejections and turbine trips (where the load rejectionTavg mode is used) and reactor trips (where the plant tripTavg mode is used). The load rejection controller is discussed in subsection 7.7.1.9.1. The plant trip controller is discussed in subsection 7.7.1.9.2.

- The pressure mode uses the difference between measured steam header pressure and a pressure setpoint to generate a steam dump demand signal. This mode is used for low-power conditions (up through turbine synchronization) and for plantcooldown. It is described in subsection 7.7.1.9.3.

Process variable input signals to the steam dump control system are fed from protection channels via isolation devices and the signal selector function. Each input (Tavg, turbine load, steam header pressure, **wide-range steam generator water level**) is obtained from multiple transmitters of the same parameter. The signal selector rejects anysignal which is bad in comparison with the remaining transmitter outputs and allows only valid measurements to be used by the control system. This makes the steam dump system tolerant of single transmitter failures or input signal failures and eliminates interaction between the control and the protection system.

To prevent actuation of steam dump on small load perturbations, an independent load rejection sensing circuit is provided. This circuit senses the rate of decrease in the turbine load as detected by the turbine impulse chamber pressure. It unblocks the dump valves when the rate of a load rejection exceeds a preset value corresponding to a 10-percent step load decrease or a sustained ramp load decrease of greater than 5 percent per minute.

The steam dump system valves also receive a signal to close on a low wide-rangesteam generator water level signal. Isolating steam dump on low wide range water level improves the plant performance to anticipated transients without reactor scram events modeled in the AP1000 Probabilistic Risk Assessment.

DCD Section 7.7.1.11 is revised as follows:

7.7.1.11 Diverse Actuation System

The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection system. This backup is included to support the aggressive AP1000 risk goals by reducing the probability of a severe accident which potentially results from the unlikely coincidence of postulated transients and postulated common mode failure in the protection and control systems.

The protection and safety monitoring system is designed to prevent common mode failures. However, in the low probability case where a common mode failure does occur, the diverse actuation system provides diverse protection. The specific functions performed by the diverse actuation system are selected based on the PRA evaluation. The diverse actuation system functional requirements are based on an assessment of the protection system instrumentation common mode failure probabilities combined with the event probability.

The functional logic for the diverse actuation system is shown in Figure 7.2-1, sheets 19 and 20.

Automatic Actuation Function

The automatic actuation signals provided by the diverse actuation system are generated in a functionally diverse manner from the protection system actuation signals. The common-mode failure of sensors of a similar design is also considered in the selection of these functions.

The automatic actuation function is accomplished by redundant microprocessor-based subsystems. Input signals are received from the sensors by an input signal conditioning block, which consists of one or more electronic modules. This block converts the signals to standardized levels, provides a barrier against electromagnetic and radio frequency interference, and presents the resulting signal to the input signal conversion block. The conversion block continuously performs analog to digital signal conversions and stores the value for use by the signal processing block.

The signal processing block polls the various inputs under the control of a software-based algorithm, evaluates the input signals against stored setpoints, executes the programmed logic when thresholds are exceeded, and issues actuation commands.

The resulting output signals are passed to the output signal conversion block, whose function is to convert microprocessor logic states to parallel, low-level dc signals. These signals are passed to the output signal conditioning block. This block provides high-level signals capable of switching the traditional power plant loads, such as breakers and motor controls. It also provides a barrier against electromagnetic and radio frequency interference.

Diversity is achieved by the use of a different architecture, different hardware implementations and different software from that of the protection and safety monitoring system.

The diverse design uses standard input modules designed for use with small industrial computer systems. It also uses a microprocessor board different from those used in the protection system.

Software diversity is achieved by running different operating systems and programming in different languages.

The diverse automatic actuations are:

- Trip rods via the motor generator set, trip turbine,~~and~~ initiate the passive residual heat removal, **actuate core makeup tanks, and trip the reactor coolant pumps** on low wide range steam generator water level
- Open the passive heat removal discharge isolation valves and close the in-containment refueling water storage tank gutter isolation valves on high hot leg temperature
- Trip rods via the motor generator set, trip turbine, actuate the core makeup tanks, and trip the reactor coolant pumps on low pressurizer water level
- Isolate selected containment penetrations and start passive containment cooling water flow on high containment temperature

The selection of setpoints and time responses determine that the automatic functions do not actuate unless the protection and safety monitoring system has failed to actuate to control plant conditions. Capability is provided for testing and calibrating the channels of the diverse actuation system.

DCD Figure 7.2.1 is revised as shown

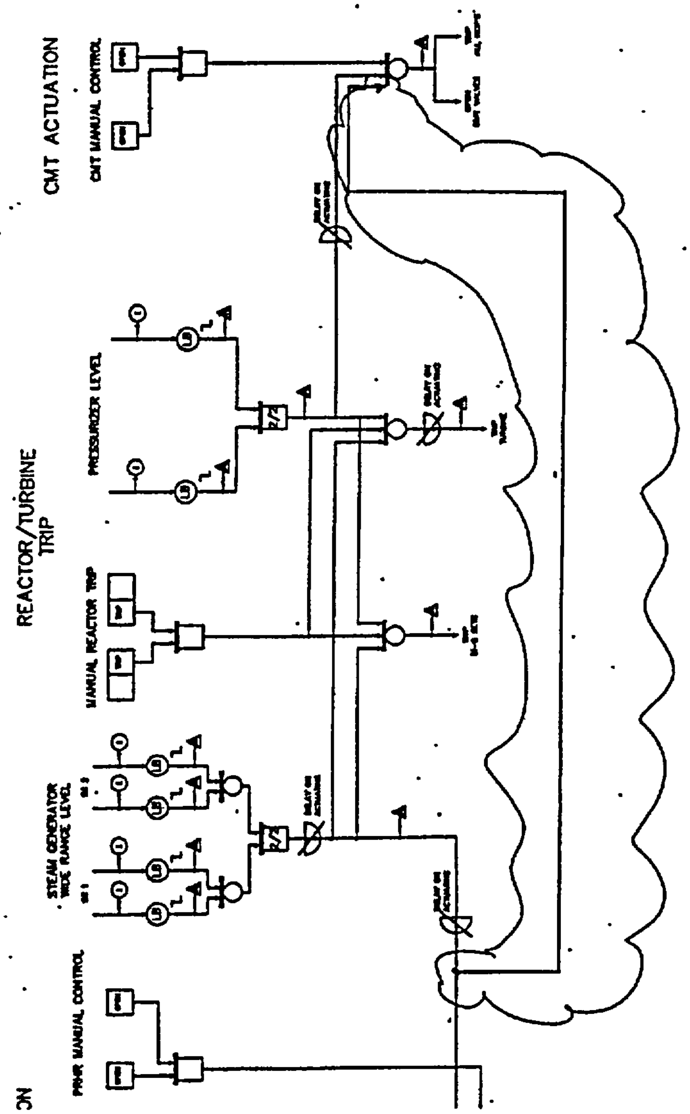


Figure 7.2-1 (Sheet 19 of 20)

Functional Diagram
Diverse Actuation System Logic Automatic Actuations

PRA Revision:

The PRA will be revised to include the revised ATWS analysis results presented in Table 440.014-1 and Figures 440.014-1 through 4.

Revision of Section A4.1

A4.1 ATWS Background

Failure of the reactor trip function could result from several causes:

- Reactor trip signal from the protection and monitoring system (PMS) fails
- Reactor trip breakers fail to open
- Rod cluster control assemblies (RCCAs) fail to fall into the core after power to the gripper coils is removed

For AP1000, if the reactor trip function fails, the diverse actuation system (DAS) would provide a backup method for tripping the reactor. Failure of DAS reactor trip could result from failure of the DAS signal, or failure of the RCCA motor-generator (MG) sets to trip.

There is no credible mechanism for mechanical binding of multiple RCCAs once power is removed from the gripper coils (except possibly as a result of a seismic event, which the internal events PRA and models are not intended to address). Further, even with all of the RCCAs stuck out of the core, AP1000 core characteristics and plant features are available to mitigate the event consequences and avoid an overpressure in excess of the ASME service level C limit.

ATWS analysis was performed for the AP600 plant (Reference A-20). This analysis demonstrated that the AP600 plant could successfully ride out an ATWS event without inserting the control rods, considering that:

- Loss main feedwater is the most limiting initiating event
- PRHR HX provides an adequate heat sink
- The core reactivity feedback is sufficient to limit the peak RCS pressure to less than 3200 psig for more than 95 percent of full power core life

This analysis showed that the AP600 response to ATWS is comparable to existing Westinghouse PWRs.

The AP1000 employs a low-boron core. One of the benefits of such a core design is that the total reactivity feedback properties of the core, including moderator temperature coefficient, are more negative throughout core life than in conventional cores. As a result, as shown by the following plant analysis, the

AP1000 has a zero unfavorable exposure time (UET) for equilibrium core cycles. For about 40 percent of the first core cycle the allowable maximum RCS pressure may be exceeded. As a bounding assumption, this would only result in a UET of 1.5 percent, assuming a plant life of 40 years. ~~If~~ It would be even less for the plant design life of 60 years.

~~The following ATWS T/H analysis has three significant conservatisms. One is that the reactivity feedback during an ATWS transient as the reactor temperature increases is underestimated. The second is that the SG heat transfer remains very high until the SG dries out and then it suddenly drops to zero. Studies done for other plants have indicated that these two effects would significantly reduce the peak RCS pressure such that ATWS would be acceptable during most if not all of the first core cycle and the AP1000 UET would be essentially zero. The third is that the pressurizer safety valve capacity was assumed to be just equal to the system requirement for the valve; the valve provided will have a greater rated capacity.~~ **The AP1000 also includes automatic CMT actuation on the same DAS signal that starts the PRHR on low wide-range steam generator water level. If CMT operation is credited, the UET is zero even for the first core cycle. Additional ATWS analysis is provided in Section A4.2.3 that demonstrates this performance.**

Revision of Section A4.2

A4.2 ATWS Analysis

ATWS analyses for the Westinghouse PWR plants have historically indicated Loss of Normal Feedwater as the most limiting ATWS event.

Specific analyses, performed for the AP600 plant, confirmed that this conclusion is also applicable to the passive plant designs.

Main reasons for this result are the following:

- **Component sizing approach**
  - **PRZ and SG safety valve sizing**
  - **Pressurizer sizing**
- **Core and fuel characteristics**
  - **Fuel pitch**
  - **Moderator to fuel volume ratio**
  - **Moderator coefficient**
  - **Doppler coefficient**
- **Steam generator design**
  - **Vertical U tube steam generator**
  - **Significant secondary side water inventory**

The same conclusions can also be drawn for the AP1000 that compares closely to the AP600 plant.

In particular, the following considerations apply:

- **PRZ and SG safety valves are sized to be able to relief the steam resulting from a full power mismatch. This means that, even considering the worst loss of load transient, PRZ pressure cannot exceed Condition II acceptance criteria (110% of the design pressure).**
- **The pressurizer in the AP600 and AP1000 are about 50% larger than those in traditional PWRs with the same power rating. Hence, a larger coolant swelling (i.e., larger temperature changes) must occur to overfill the pressurizer. This, in turn, results in a larger negative reactivity insertion from the moderator reactivity feedback at the time of overfilling.**
- **Steam generators are essentially sized according to the same rules. Water inventory in the steam generators is sufficient for about 80–90 full power seconds both in the AP1000 and AP600 and, in addition, following a loss of normal feedwater event, there are roughly the same full power seconds from normal steam generator water level to the reactor trip setpoint on low steam generator level.**
- **Kinetics parameters are close enough since, apart for the different fuel length, the same fuel is used for the two plants. In addition, the AP1000 is characterized by a Low Boron Core Design that provides significant benefits in terms of Moderator Reactivity Coefficient.**

~~Analysis has been performed for the AP1000 plant to verify that the peak RCS pressure is less than the ASME emergency stress limits, which occurs at greater than 3200 psia.~~ As was done for the AP600 ATWS analysis (Reference A-20), the LOFTRAN computer code is used to perform these analysis. The AP600 ATWS analysis (Reference A-20) determined that the most limiting initiating event for an ATWS is a complete loss of normal feedwater.

Analysis has been performed for the AP1000 plant to verify that the peak RCS pressure is less than the ASME emergency stress limits, which occurs at greater than 3200 psia. In these analysis, the control rods are not inserted, even though DAS automatically de-energizes the motor generator set power. All of the mitigating system actions are modeled as being actuated by the DAS. DAS uses a low wide range SG level signal to actuate the following:

- Automatic trip of the turbine
- **Automatic trip of the reactor coolant pumps**
- **Automatic trip of the CMTs**
- Automatic start of PRHR HX

Both of the pressurizer safety valves are assumed open when the pressure exceeds their setpoint.

Addition of *Section A4.2.3*

A4.2.3 ATWS Analysis with PRHR Operation

The core MTC is –5.0 pcm/°F, which occurs at the beginning of the first core cycle. At any other time in the first core cycle or in an equilibrium core cycle, the MTC will be more negative.

This case assumes operation of the PRHR heat exchanger and CMTs; the PHRH heat exchanger is assumed to be available. Only one CMT is assumed to be operable.

The sequence of events is as follows:

ATWS with PRHR Operable	
Time (sec)	Event
0 to 4	All feedwater flow to SGs is lost
50	Pzr SV open
62.1	Turbine bypass fully open
64.1	Turbine is tripped on DAS signal
66.1	RCP Trip on DAS signal
74.1	PRHR HX is actuated on DAS signal
74.1	CMTs are actuated on DAS signal
141	Max RCS pressure reached
230	Pzr SV re-close

The results for this analysis are shown in Figures A4.2.3-1 through A4.2.3-4.

As seen in Figure A4.2.3-1, the peak RCS pressure is 2818 psia. This provides margin to the pressure limit of 3200 psig.

Addition of Figures A4.2.3-1 through A4.2.1-4

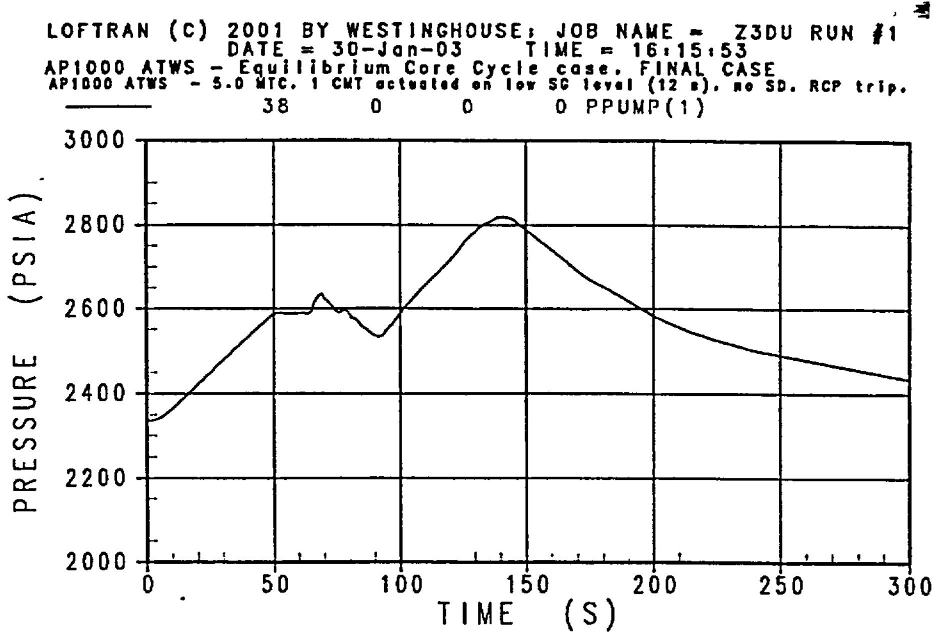
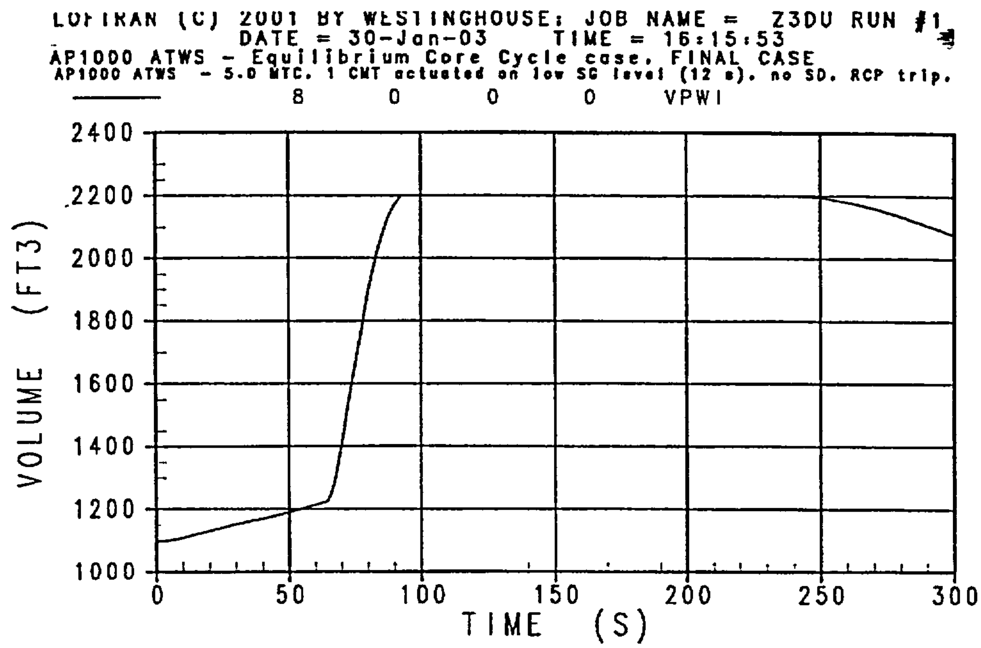


Figure A4.2.3-1

ATWS PRHR Operable – RCS Pressure



Note: The pressurizer volume includes the volume of the surge line.

Figure A4.2.3-2

ATWS PRHR Operable – PRZ Volume

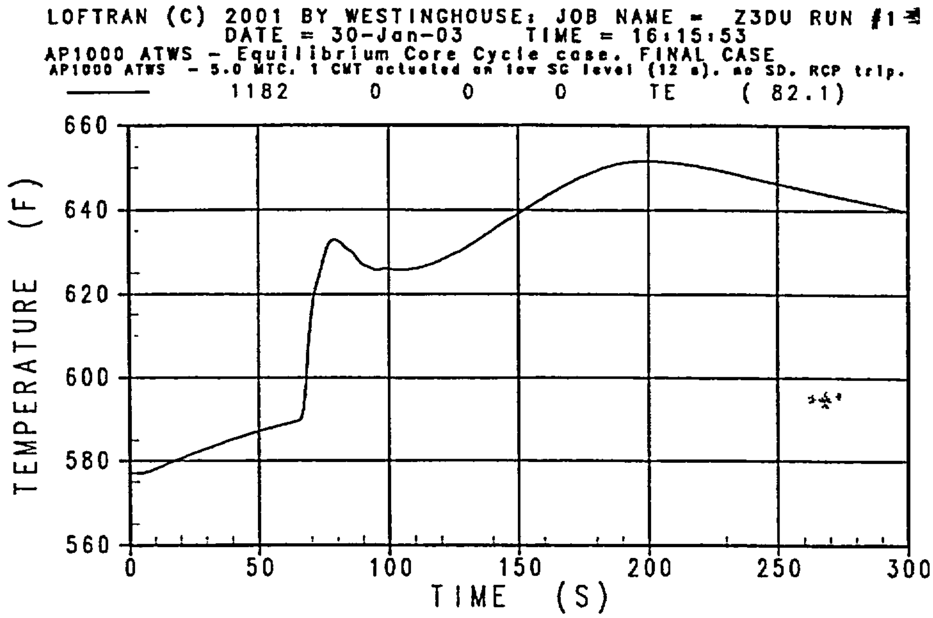


Figure A4.2.3-3

ATWS PRHR Operable – Core Average Temperature

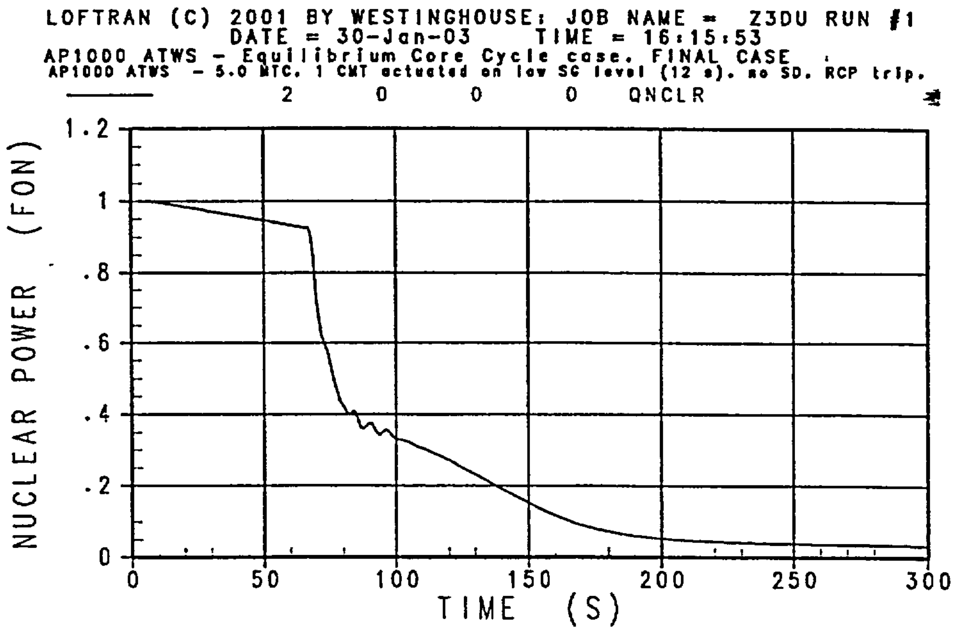


Figure A4.2.3-4

ATWS PRHR Operable – Nuclear Power

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

ATTACHMENT 440.014 – 1

JUSTIFICATION OF THE USE OF LOSS OF NORMAL FEEDWATER AS AP1000 LIMITING ATWS INITIATOR

1 Introduction

ATWS analyses for the Westinghouse PWR plants have historically indicated Loss of Normal Feedwater as the most limiting ATWS event.

Specific analyses, performed for the AP600 plant, confirmed that this conclusion is also applicable to the passive plant designs.

Main reasons for this result are the following:

- Component sizing approach
  - PRZ and SG safety valve sizing
  - Pressurizer sizing
- Core and fuel characteristics
  - Fuel pitch
  - Moderator to fuel volume ratio
  - Moderator coefficient
  - Doppler coefficient
- Steam generator design (U tubes steam generator)

The same conclusions can also be drawn for the AP1000 that compares closely to the AP600 plant.

In particular, the following considerations apply:

- PRZ and SG safety valves are sized to be able to relief the steam resulting from a full power mismatch. This means that, even considering the worst loss of load transient, PRZ pressure cannot exceed Condition II acceptance criteria (110% of the design pressure).
- The pressurizer in the AP600 and AP1000 are about 50% larger than those in traditional PWRs with the same power rating. Hence, a larger coolant swelling (i.e., larger temperature changes) must occur to overfill the pressurizer. This, in turn, results in a larger negative reactivity insertion from the moderator reactivity feedback at the time of overfilling.

- Steam generators are essentially sized according to the same rules. Water inventory in the steam generators is sufficient for about 80–90 full power seconds both in the AP1000 and AP600 and, in addition, following a loss of normal feedwater event, there are roughly the same full power seconds from normal steam generator water level to the reactor trip setpoint on low steam generator level.

- Kinetics parameters are close enough since, apart for the different fuel length, the same fuel is used for the two plants. In addition, the AP1000 is characterized by a Low Boron Core Design that provides significant benefits in terms of Moderator Reactivity Coefficient.

2 Assessment of AP1000 ATWS Initiators

In the following paragraph, for each Condition 2 accident (Anticipated Transient) presented in the DCD Chapter 15, a brief justification will be provided to demonstrate that the consequences of the ATWS event initiated by the considered accident are bounded by those related to the loss of normal feedwater assumed as initiating event.

The initiating events will be discussed following the DCD Chapter 15 sequence.

In addition, analyses will be provided for the anticipated transients precursors that contribute the most to the ATWS overall frequency.

The selection of the transients to be considered has been made on a probabilistic basis and following the same approach defined in the AP600 ATWS Analysis Report (Attachment #2 to Westinghouse letter DCP/NRC1240 dated January 30, 1998).

3 Definition of the Initiating Events to be Analyzed

Table 1 provides a breakdown of the frequency of the various ATWS failure modes. Table 2 reports the ATWS precursors together with their frequency for the AP1000. It would be possible, combining the frequency from Table 2 with the Failure Modes frequencies from Table 1, to obtain a table reporting the frequencies of occurrence for each initiating event and the associated failure mode frequencies.

TABLE 1
ATWS BREAKDOWN BY FAILURE MODE

Failure Mode	Failure on Demand	Frequency	Percent
PMS Reactor Trip Failure	8.03×10^{-5}	1.34×10^{-4}	89
Reactor Trip Breaker Failure	8.1×10^{-6}	1.35×10^{-5}	9
Mechanical Failure	1.8×10^{-6}	$3. \times 10^{-6}$	2

TABLE 2				
Event Category	Frequency	Contributors		
		Category	Description	Frequency
ATWS Precursor Without Main Feedwater	4.81E-01	17	Loss of main feedwater to both steam generators	2.99E-1
		18	Loss of condenser	1.12E-1
		19	Loss of compressed air	3.48E-2
		Consequential ATWS	It is the results of the ATWS precursors with MFW event and the failure of the main feedwater system	3.51E-2
ATWS Precursor With SI	1.48E-2	3	Medium LOCA	4.36E-04
		4	Core makeup tank line break	9.31E-05
		5	Safety injection line break	2.12E-04
		6	Small LOCA	5.00E-04
		7	RCS leakage	6.20E-03
		8	PRHR tube rupture	1.34E-04
		9	Steam generator tube rupture	3.88E-03
		21	Main steam line break downstream of MSIVs	5.96E-04
		22	Main steam line break upstream of MSIVs	3.72E-04
		23	Main steam line stuck open valve	2.39E-03
ATWS Precursor With Main Feedwater	1.17E+00	12	Transients with MFW (turbine trip only)	8.14E-01
		13	Loss of RCS flow	1.80E-02
		14	Loss of main feedwater to one steam generator	1.92E-01
		15	Core power excursion	4.50E-03
		16	Loss of component cooling water/service water	1.44E-01

Nevertheless, it is easy to see that Categories 12, 13, 14, 16, 17, and 18, represent more than 95% of the initiating events. Hence, also from this point of view, the AP1000 is very similar to the AP600.

The classes to be considered more in detail in the AP1000 ATWS assessment can be further reduced based on the following consideration:

1. Loss of Condenser Vacuum and other events resulting in a turbine trip (Categories 18 and 16) are enveloped by the turbine trip analysis. In fact, the Loss of Condenser Vacuum resulting in a turbine trip is equivalent to a Turbine Trip Event without availability of the steam dump system.

2. Category 14 events are essentially bounded by Loss of Normal Feedwater Event.

3. Feedwater System Malfunction that results in a decrease in FW temperature is a mild transient and it has been analyzed for inclusion in the Chapter 15 of the DCD assuming no protective actions. The analysis shows that all the acceptance criteria are met.

4. Inadvertent operation of the PRHR has been analyzed for inclusion in Chapter 15 of the DCD assuming no protective actions. All safety criteria are met.

5. ATWS precursors with SI (i.e., Condition 2 Events resulting in a S signal, including the Inadvertent Operation of the PXS during power operation/CMT spurious actuation) result in a fast boration of the primary system with the consequent power decrease. CMT boration provides the addition of more than 500pcm of boron to the RCS coolant and hence assures the core shutdown.

Based on the above and considering the results of the AP600 analysis, it has been decided to analyze the following transients:

1. Turbine trip without feedwater system operable with turbine bypass system operable

2. Turbine trip with feedwater system operable with turbine bypass system operable

3. Turbine trip without feedwater system operable without turbine bypass system operable (This analysis would also cover the loss of condenser vacuum transients)

4. Loss of normal feedwater event with turbine bypass system operable

5. Loss of normal feedwater event, without spray system, without turbine bypass system operable

6. Loss of normal feedwater event with turbine bypass system operable, more realistic SG UA

7. Complete loss of forced coolant flow with MFW system operable with turbine bypass system operable

8. Complete loss of forced coolant flow with MFW system operable without turbine bypass system operable

9. Complete loss of forced coolant flow induced by the loss of ac power (mechanical CMF), MFW system not operable, no steam dump system operable, turbine trip at the initiation of the transient

4 Review of the ATWS Initiators

4.1 Increase in Heat Removal from Secondary System

4.1.1 Feedwater System Malfunction Resulting in a Decrease in Feedwater Temperature

The analysis of this accident is presented in the DCD assuming no PMS actuation. The increase in heat removal results in a decrease in RCS average temperature. Moderator temperature reactivity coefficient and/or Reactor Control system insert the reactivity required to match the secondary side load requests. The plant reaches a new steady state condition at higher power level. Also assuming no protective action all safety criteria are met. Should a PMS setpoint be reached, resulting in a turbine trip and should a mechanical failure or a Reactor Trip Breaker CMF prevent reactor trip, the resulting transient would be similar but less limiting than the turbine trip.

Eventually, based on the system availability (e.g., Main Feedwater, Steam Dump), DAS actuation would provide, if needed, the mitigative actions to maintain RCS Pressure below the acceptance criteria. This is the same as AP600.

4.1.2 Feedwater System Malfunction Resulting in an Increase in FW Flow

This analysis is performed assuming maximum moderator reactivity feedback. The increase in FW flow causes an increase in heat transfer from the secondary system and the moderator temperature reactivity coefficient and/or Reactor Control system insert the reactivity required to match the secondary side load requests. Reactor core reaches a new equilibrium power, few percent above the nominal.

The level in the steam generator increases until a reactor trip setpoint is reached. Should the ATWS event be caused by a mechanical common failure or a trip breakers common mode failure, the PMS would isolate the feedwater and trip the turbine. From this time on, the event would proceed as a Turbine Trip ATWS, but will be less severe since more water would be available in the steam generators.

If the ATWS would be caused by a PMS common mode failure, the operator, once they diagnose the event, can use DAS to trip the reactor and the turbine. In this case, automatic turbine trip is also expected on Turbine Protection logic that would trip the turbine in case of high steam moisture.

All the acceptance criteria for this initiating event are met with margin. In fact, it should be noted that, before any protective action (by DAS or PMS), temperature and pressure in the RCS are lower than nominal and power level is only slightly higher than nominal.

4.1.3 Excessive Increase in Secondary Steam Flow

Also in this case the analysis for the Chapter 15 of the DCD is performed assuming no PMS actuations. The reactor core power increases (due to moderator feedback or power control system operation) to match secondary request. All the acceptance criteria for condition 2 are met, even assuming no reactor trip (the analysis presented in the DCD is equivalent to an ATWS with CMF of the PMS). Should, however, a reactor trip setpoint be reached, and a mechanical or RTB failure avoid the reactor trip, the ATWS event would develop similarly to a turbine trip ATWS event starting from a lower temperature and pressure. The availability of the feedwater would depend on the assumed PMS failure. In any case, the consequences of this event are bounded by those resulting from the Turbine trip ATWS event without main feedwater.

4.1.4 Inadvertent Opening of an SG PORV or SV

This event results in a transient similar to the excessive increase in secondary steam flow. The core power increases to match the secondary system heat transfer. The load increase caused by the opening of a SG safety valve is less than the 10% increase assumed for the Excessive load increase. Without any PMS or DAS actuation, the plant reaches a new equilibrium at a lower RCS pressure and temperature and at a higher core power level.

As described in the AP600 ATWS Analysis Report, an ATWS event due to mechanical CMF results in several signals from the PMS. Even if rods do not insert due to the assumed mechanical CMF, PMS calls for emergency boration on a Low-T_{cold} or Low Steam Line Pressure signal. Following boration of the RCS via CMT injection, core power progressively decreases.

If the ATWS is caused by a failure of the PMS, a DAS setpoint may be reached on Low Pressurizer Level Signal or on a Low SG Level signal (if feedwater system does not compensate the increased steam flow). DAS actuates the reactor trip, the turbine trip and PRHR. If the DAS signal is generated on the Low PRZ Level also RCPs are tripped and CMTs are actuated to provide boration and shutdown the reactor.

4.1.5 Inadvertent PRHR Operation

As the previous events, also this one results in an additional heat load. Chapter 15 DCD analyses are performed assuming that no trip would occur. If a PMS trip signal is reached, but reactor trip does not occur due to mechanical or RTB common mode failures, the transient is similar but less limiting than the turbine trip. In fact, following the PMS trip signal, a turbine trip would occur and even assuming that the RCCA do not drop in the core due to mechanical CMF (that represent only 2% of the ATWS causes), the transient would develop similar to a turbine trip event, with the PRHR already actuated and the full secondary side water inventory available.

4.2 Decrease in Heat Removal from the Secondary System

4.2.1 Loss of Electrical Load

Following the loss of electrical load, the turbine control system trips the turbine. From here on the transient is similar to a turbine trip event.

4.2.2 Turbine Trip Events

Following the turbine trip, RCS pressure and temperature quickly rise and a negative reactivity insertion occurs. RCS temperature increase is dictated by the availability of the steam dump system. However, even assuming that feedwater system is not available, the transient is much milder than the Loss of Normal Feedwater, since a large secondary side inventory is available at the time of turbine trip. The RCS pressurization and heat up are slower than in the loss of normal feedwater case, since more time is required to dryout the steam generators, even assuming that the steam dump operates.

Three different cases have been analyzed as follows:

1. Turbine trip without feedwater system operable with turbine bypass system operable
2. Turbine trip with feedwater system operable with turbine bypass system operable
3. Turbine trip without feedwater system operable without turbine bypass system operable (This analysis would also cover the loss of condenser vacuum transients.)

The results confirm that the consequences of the turbine trip event are bounded by those related to the loss of normal feedwater. The results are illustrated in the following.

Case 1) Turbine trip without feedwater system operable with turbine bypass system operable

The event is characterized by a sudden decrease in steam flow. This results in a power mismatch between primary and secondary system and hence in a fast RCS temperature increase. Power level starts decreasing soon while there is still a large secondary side inventory. Nevertheless, due to the steam dump operation, the SG level eventually decreases down to the Low Low SG WR DAS setpoint. It should be noted that on the DAS setpoint a diverse reactor trip should actuate unless the reactor trip fails due to mechanical CMF.

Table 3 reports the time sequence of the events. Figures 1 through 7 show the behavior of the main plant parameters, including the reactor coolant pump outlet pressure (Figure 2).

TABLE 3

SEQUENCE OF EVENTS FOR TURBINE TRIP CASE 1
WITHOUT FEEDWATER – WITH TURBINE BYPASS

Time (Seconds)	Event
4.	Turbine trip – steam flow to steam generator terminates – main feedwater is assumed to be lost
8.	Turbine bypass valves fully open
13.	PRZ safety valves open
30.	Steam generator safety valves open
90.8	DAS low steam generator level WR setpoint
Not Credited	Reactor tripped on DAS signal above
112.8	PRHR actuation on DAS signal above
113.	PMS low steam line pressure setpoint reached (no actuation credited)
132.	PRZ overfills
136.	HL temperature > DAS high HL temperature setpoint (650°F)
141.	Maximum RCS pressure (at RCP outlet) – 2943.5 psia
> 200	PRZ safety valves close

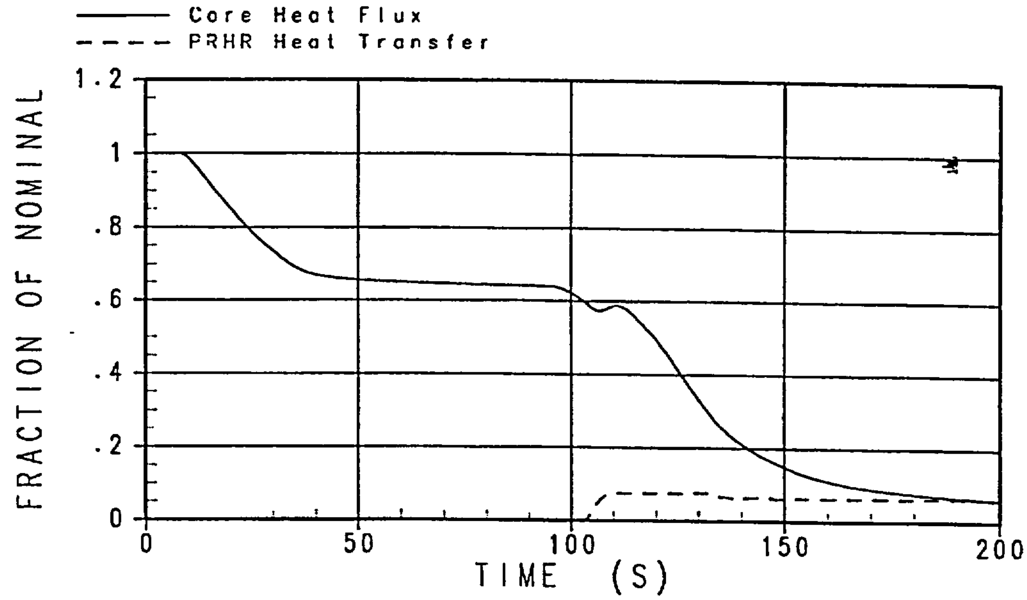


Figure 1 ATWS Turbine Trip – No Feedwater, Steam Dump Operable Core Heat Flux and PRHR Heat Flux (f.o.n) vs. Time

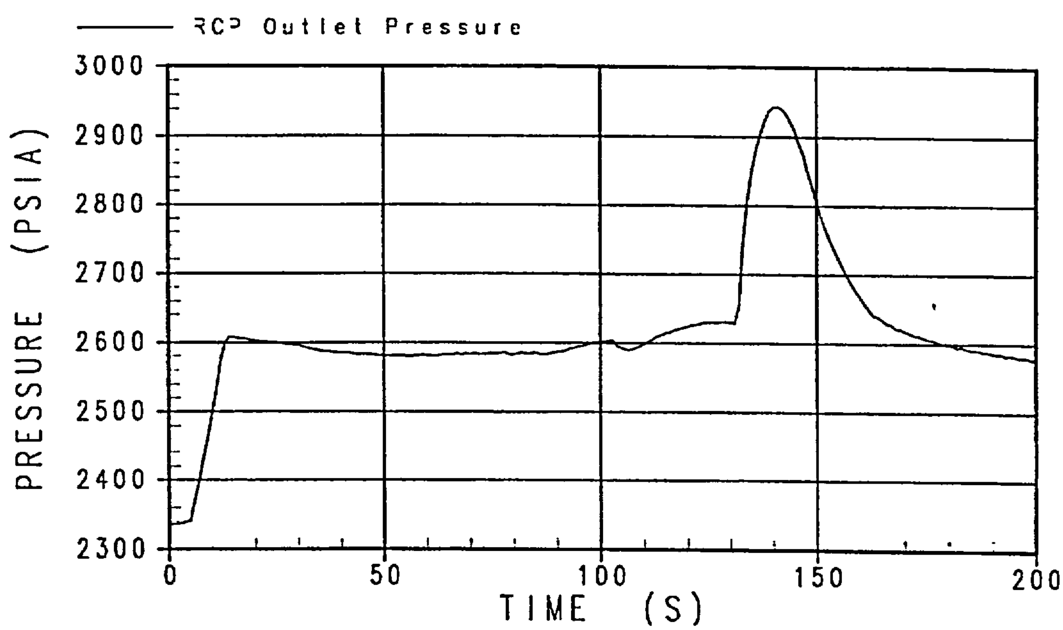


Figure 2 ATWS Turbine Trip – No Feedwater, Steam Dump Operable Reactor Coolant Pump Outlet Pressure (psia) vs. Time

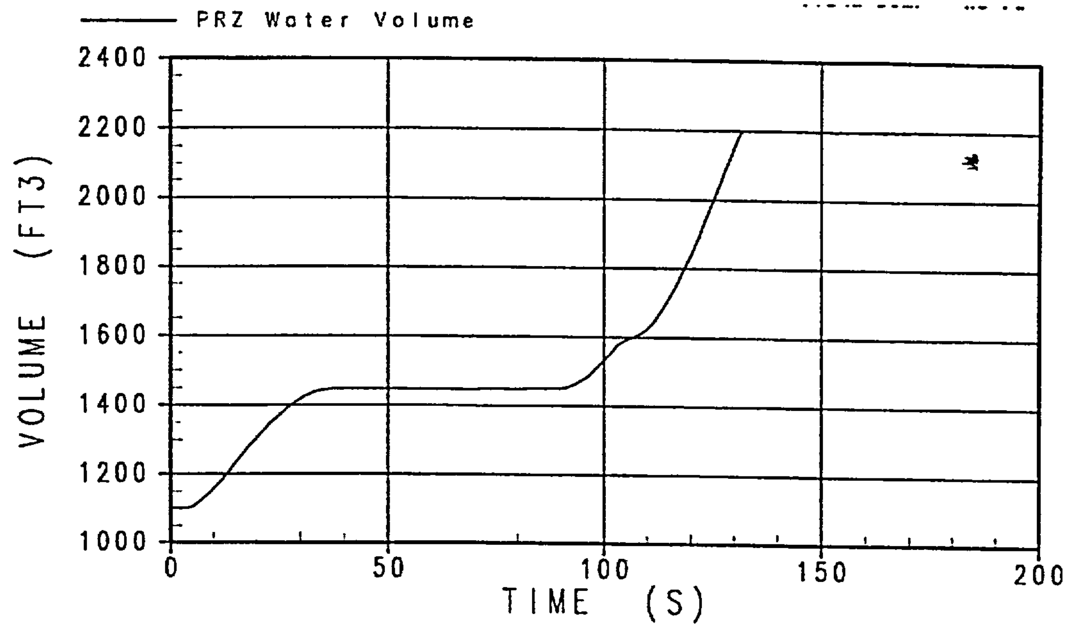


Figure 3 ATWS Turbine Trip – No Feedwater, Steam Dump Operable Pressurizer Water Volume (ft^3) vs. Time

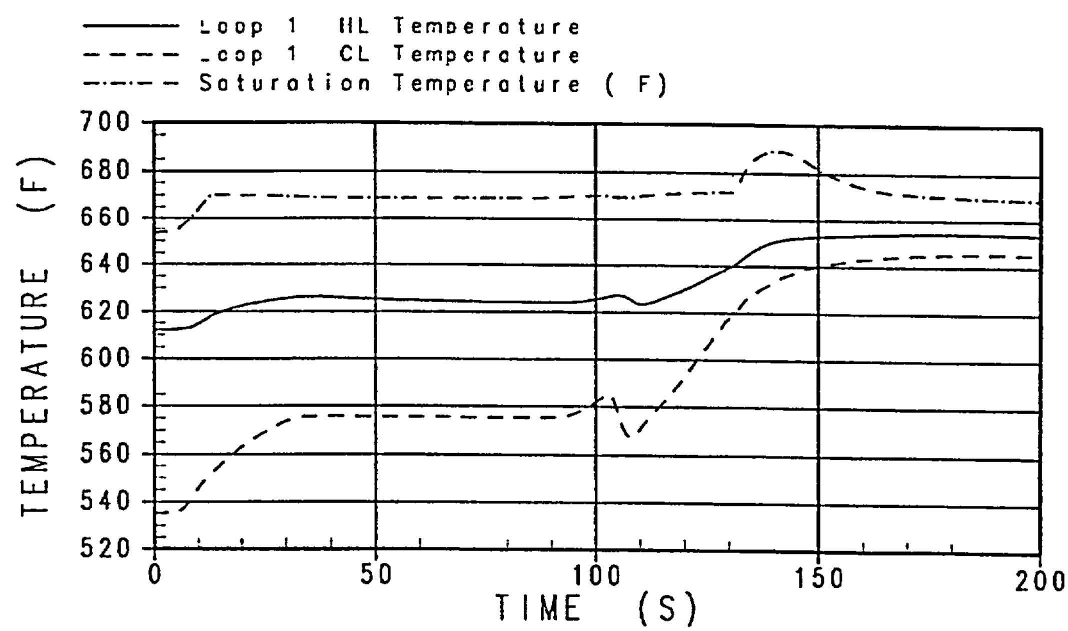


Figure 4 ATWS Turbine Trip – No Feedwater, Steam Dump Operable Loop 1 Coolant Temperatures (°F) vs. Time

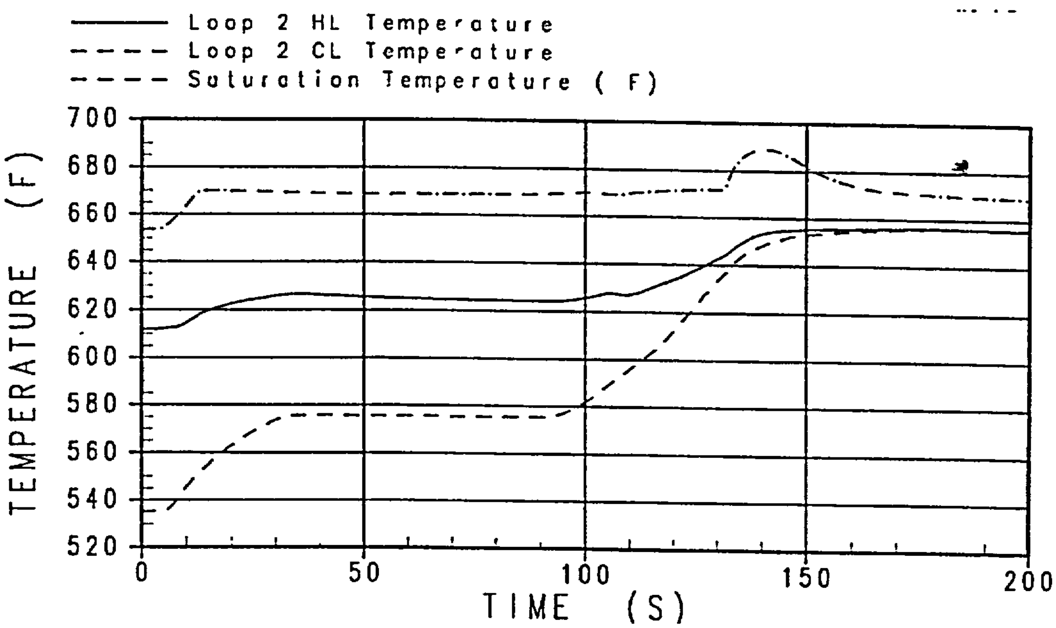


Figure 5 **ATWS Turbine Trip – No Feedwater, Steam Dump Operable Loop 2 Coolant Temperatures (°F) vs. Time**

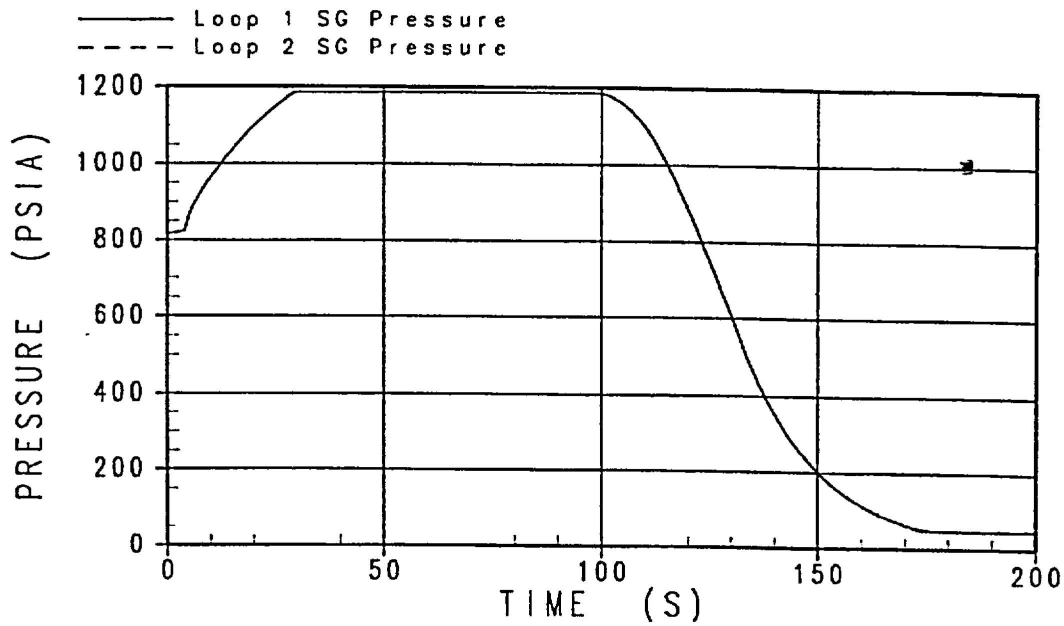


Figure 6 **ATWS Turbine Trip – No Feedwater, Steam Dump Operable Steam Generator Pressure (psia) vs. Time**

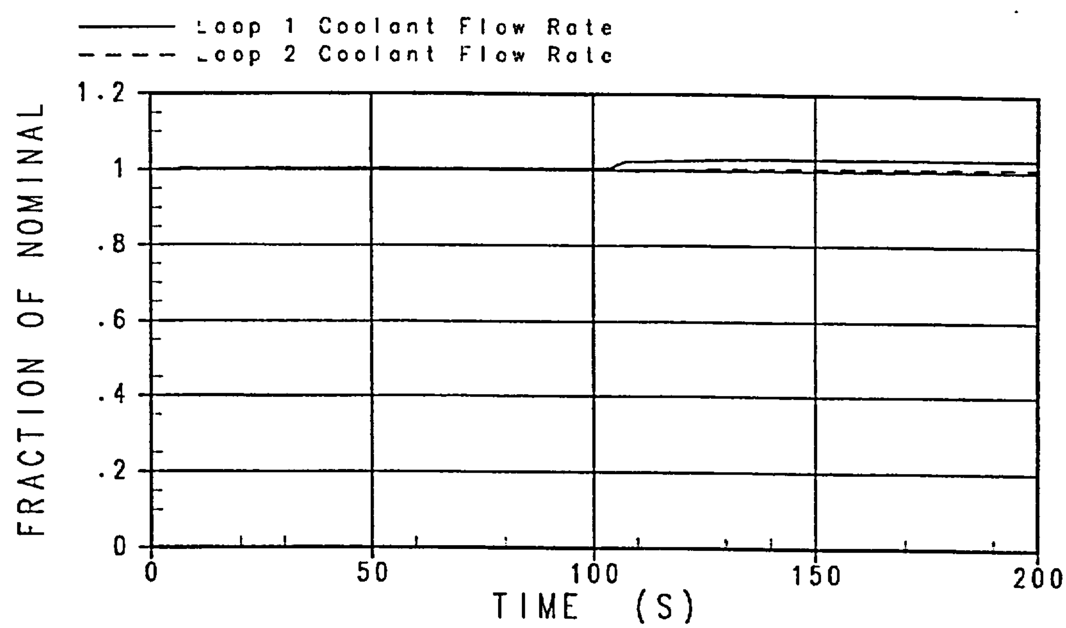


Figure 7 **ATWS Turbine Trip – No Feedwater, Steam Dump Operable Coolant Loop Volumetric Flow Rate (f.o.n) vs. Time**

Case 2) Turbine trip with feedwater system operable with turbine bypass system operable

This case is representative of Category 12 initiators, that represent the most important contributor to ATWS with a frequency of 1.22E-04 ev/y.

The sequence of events is reported in Table 4 and Figures 8 through 12 show the behavior of the main plant parameters.

It should be noted that this transient is characterized by the fact that core power reaches an equilibrium with the steam generators that release steam through the steam dump and the SG safety valves. The equilibrium is reached at about 64% of the nominal power.

Maximum pressure at the RCP outlet is 2605 psia at 15 seconds in the transient.

It should be noted that no DAS setpoint are reached in this case. Operator is expected to manually trip the reactor or initiate the boration to shutdown the core.

TABLE 4
SEQUENCE OF EVENTS FOR TURBINE TRIP CASE 2
WITH FEEDWATER – WITH TURBINE BYPASS

Time (Seconds)	Event
4.	Turbine trip – steam flow to steam generator terminates
8.	Turbine bypass valves fully open
14.	PRZ safety valves open
15.	Maximum RCS pressure (at RCP outlet) – 2605 psia
38.	Steam generator safety valves open
-----	DAS low steam generator level WR setpoint
Not Credited	Reactor tripped on DAS signal above
----	PRZ overfills
----	HL temperature > DAS high HL temperature setpoint (650°F)
~100	Core power and steam generator reach equilibrium

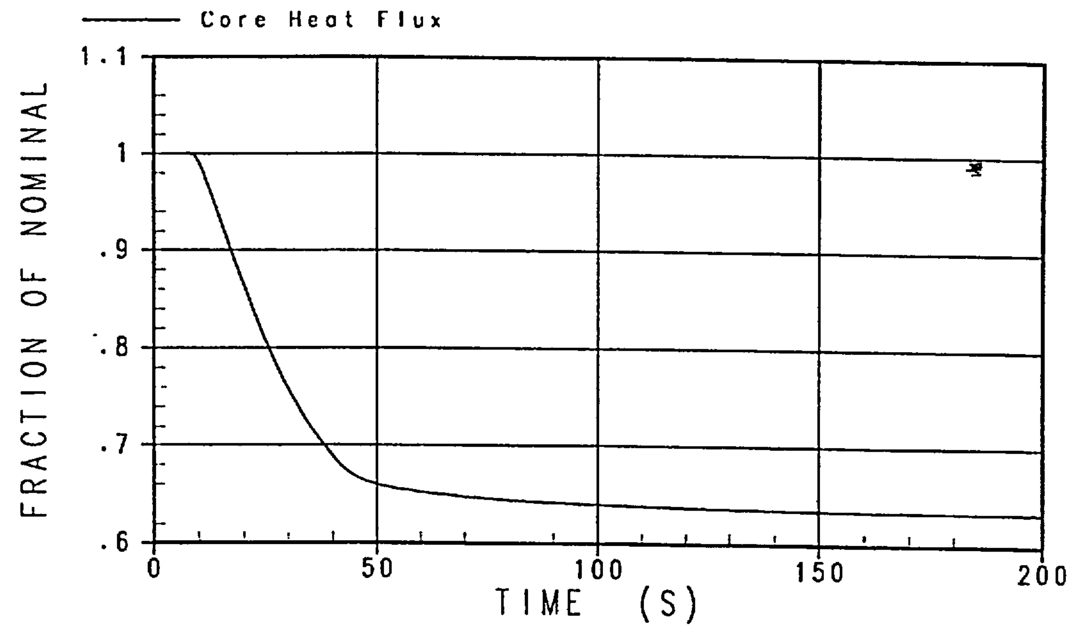


Figure 8 ATWS Turbine Trip – Feedwater Available, Steam Dump Operable Core Heat Flux (f.o.n) vs. Time

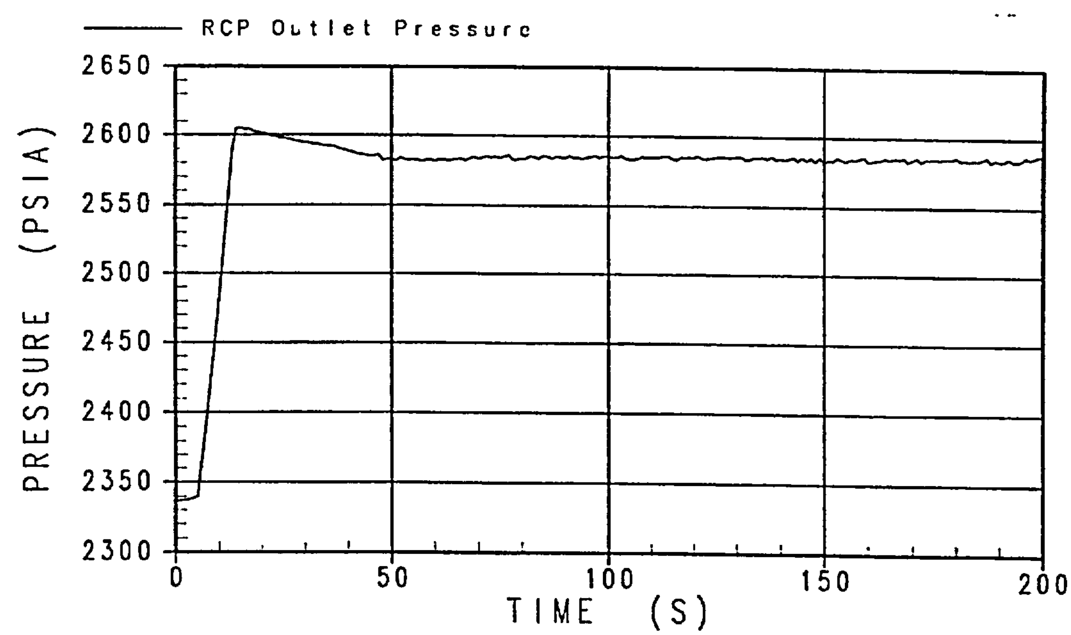


Figure 9 ATWS Turbine Trip – Feedwater Available, Steam Dump Operable Reactor Coolant Pump Outlet Pressure (psia) vs. Time

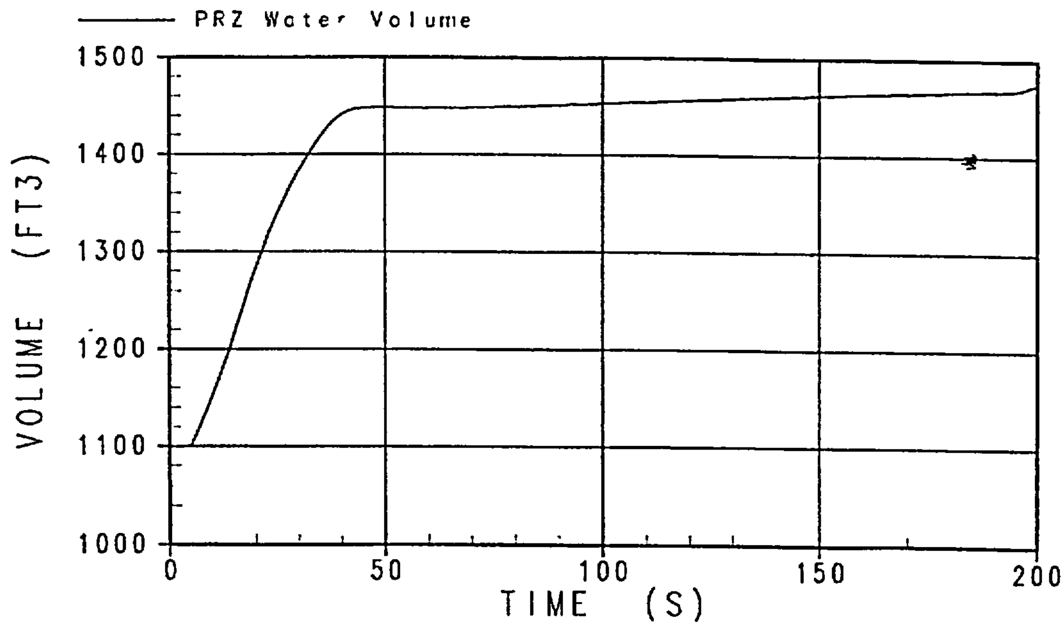


Figure 10 **ATWS Turbine Trip – Feedwater Available, Steam Dump Operable Pressurizer Water Volume (ft^3) vs. Time**

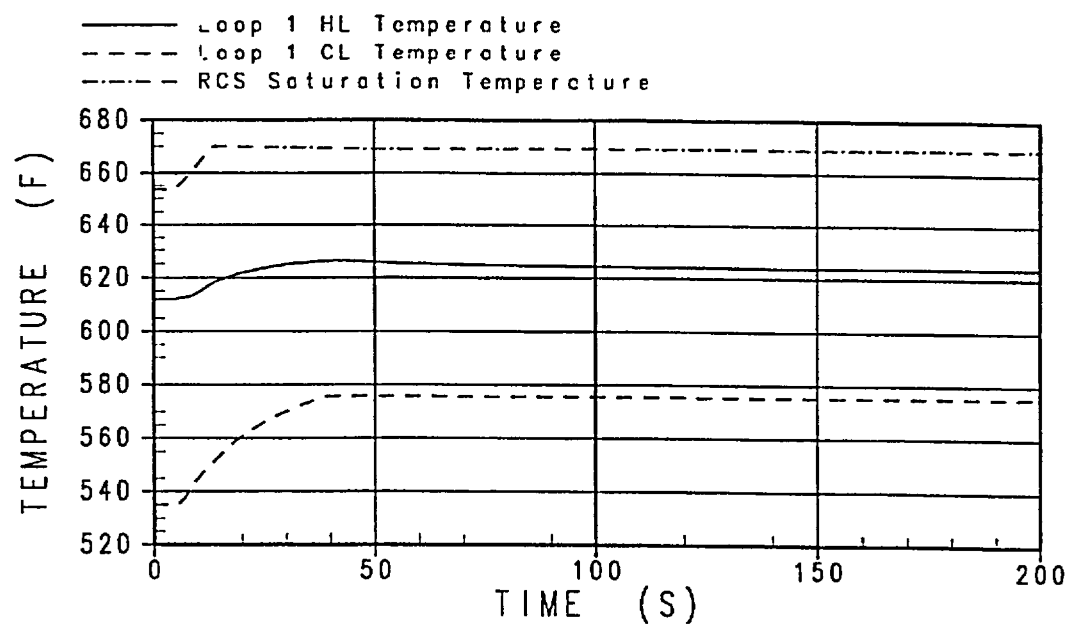


Figure 11 **ATWS Turbine Trip – Feedwater Available, Steam Dump Operable Loop 1 Coolant Temperatures (°F) vs. Time**

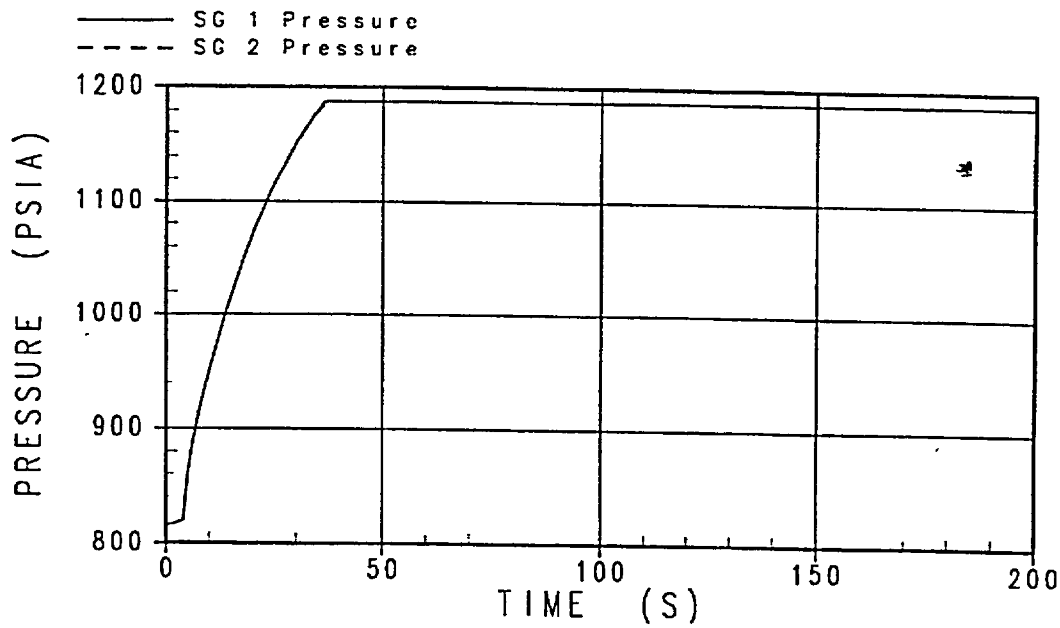


Figure 12 ATWS Turbine Trip – Feedwater Available, Steam Dump Operable SG 1 and SG 2 Pressure (psia) vs. Time

Case 3) Turbine trip without feedwater system operable without turbine bypass system operable

This analysis also covers the loss of condenser vacuum transients. It differs from Case 1 for the unavailability of the steam dump system. This of course results in an earlier heatup and pressurization of the RCS.

Nevertheless, a quasi-equilibrium condition is maintained for about 100 seconds. During this time frame, the heat transfer from the secondary system occurs by venting steam through the safety valves. However, due to the secondary inventory depletion, after the DAS setpoint on Low SG WR Level is reached, the steam generator UA starts degradating and RCS temperatures start to increase driving with them the pressurizer water level and pressure. This temperature increase also results in a power decrease via moderator temperature coefficient.

RCS peak pressure (RCP Outlet) occurs after pressurizer overfilling and it is limited to less than 2900 psia.

The time sequence of events for this case is reported in Table 5. The behavior of the main plant parameters is illustrated in Figures 13 through 18.

TABLE 5
SEQUENCE OF EVENTS FOR TURBINE TRIP CASE 3
WITHOUT FEEDWATER – WITHOUT TURBINE BYPASS

Time (Seconds)	Event
4.	Turbine trip – steam flow to steam generator terminates – main feedwater is assumed to be lost
11.	PRZ safety valves open
30.	Steam generator safety valves open
93.2	DAS low steam generator level WR setpoint
Not Credited	Reactor tripped on DAS signal above
105.2	PRHR actuation on DAS signal above
136.	PRZ overfills
144.	Maximum RCS pressure (at RCP outlet) – 2832.2 psia
152.	HL temperature > DAS high HL temperature setpoint (650°F)
> 200	PRZ safety valves close

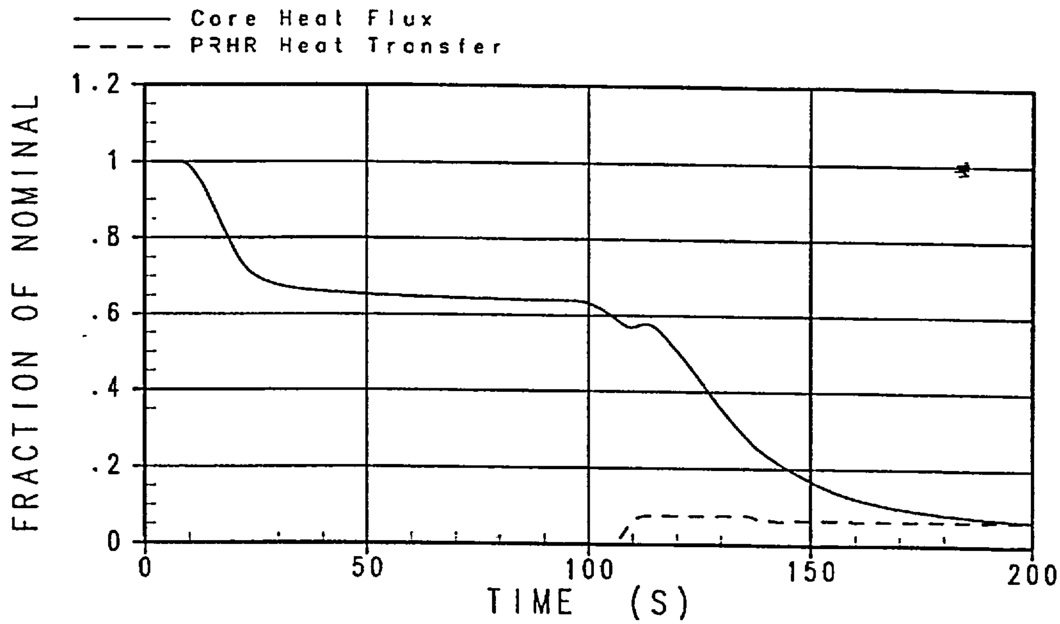


Figure 13 **ATWS Turbine Trip – Feedwater Not Available, Steam Dump Not Operable Core and PRHR Heat Flux (f.o.n) vs. Time**

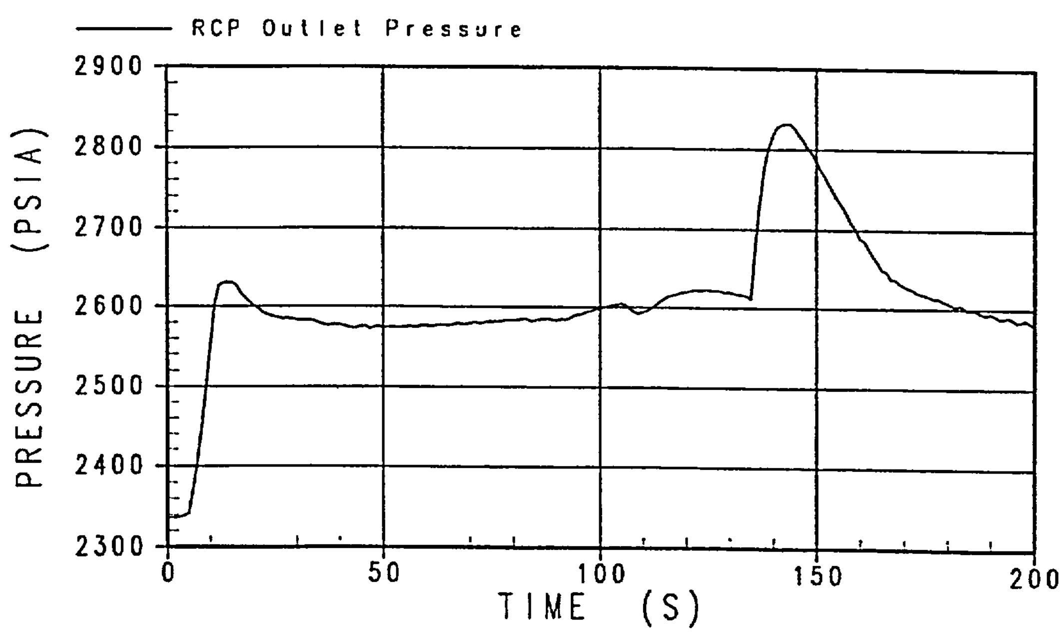


Figure 14 **ATWS Turbine Trip – Feedwater Not Available, Steam Dump Not Operable RCP Outlet Pressure (f.o.n) vs. Time**

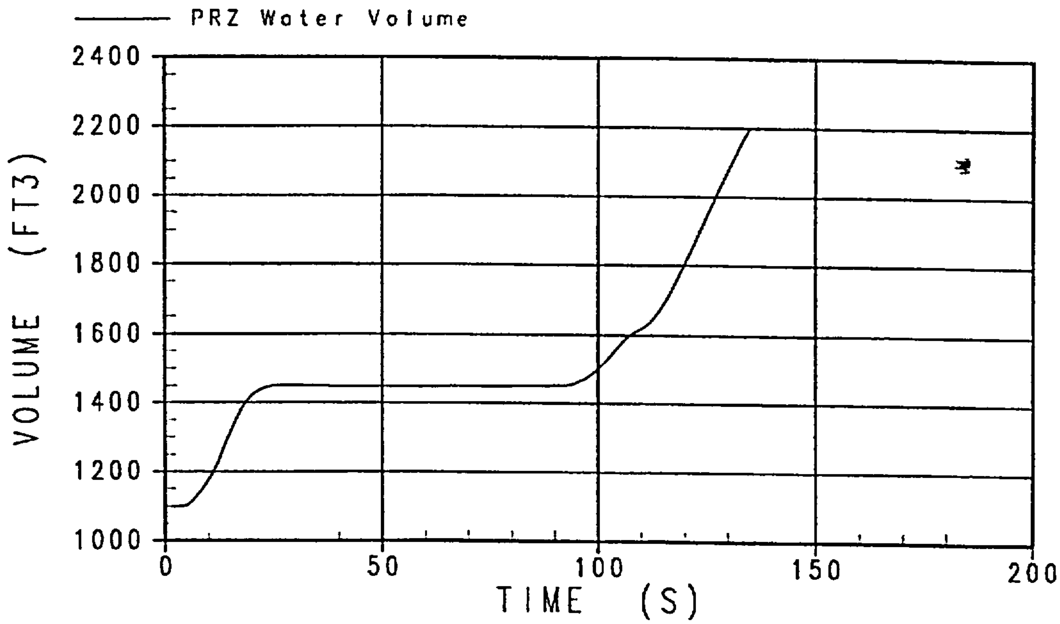


Figure 15 **ATWS Turbine Trip – Feedwater Not Available, Steam Dump Not Operable PRZ Water Volume (f.o.n) vs. Time**

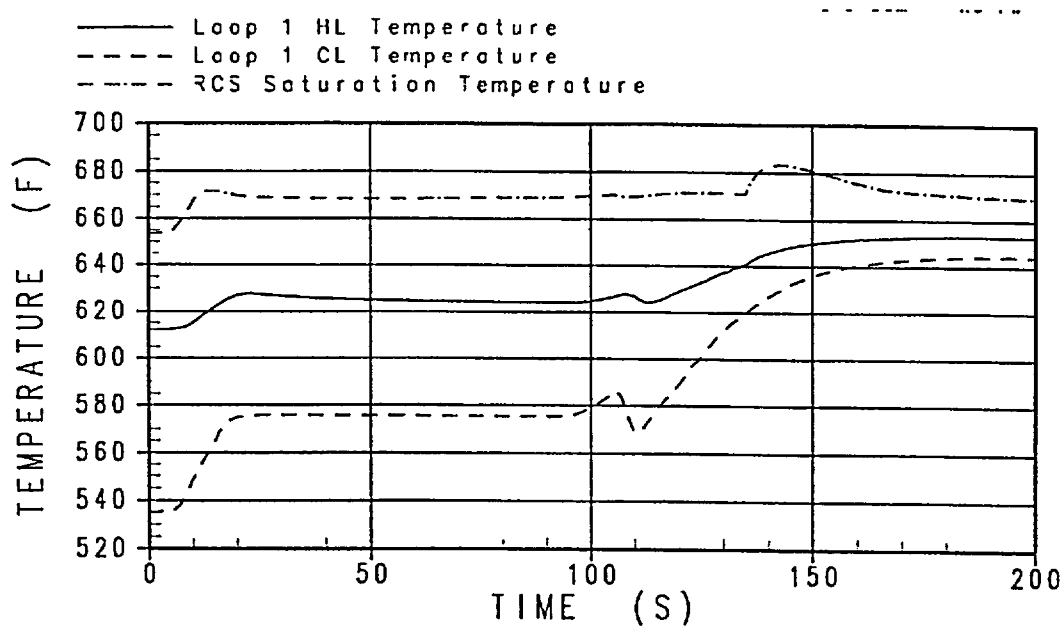


Figure 16 **ATWS Turbine Trip – Feedwater Not Available, Steam Dump Not Operable Loop 1 Coolant Temperatures (°F) vs. Time**

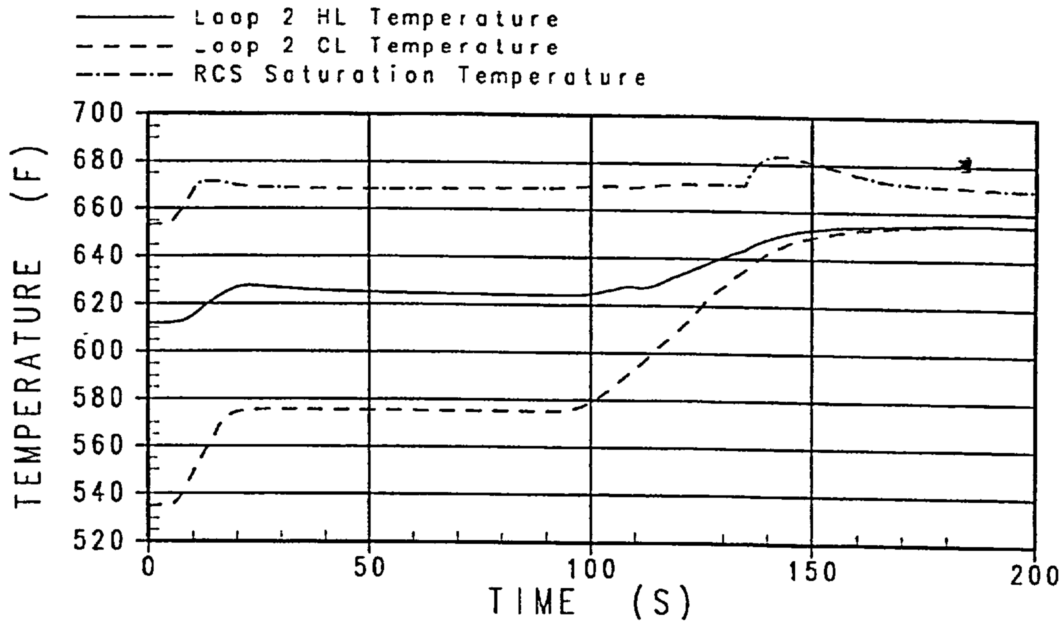


Figure 17 ATWS Turbine Trip – Feedwater Not Available, Steam Dump Not Operable
Loop 2 Coolant Temperatures (°F) vs. Time

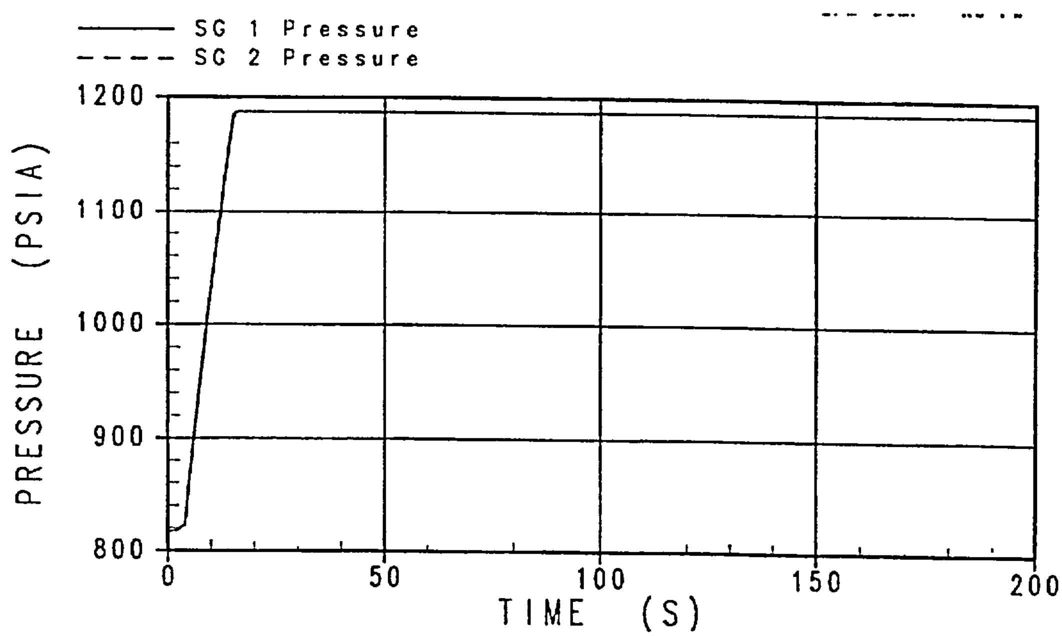


Figure 18 ATWS Turbine Trip – Feedwater Not Available, Steam Dump Not Operable
Steam Generators Pressure (psia) vs. Time

4.2.3 Inadvertent Closure of the MSIV

This transient is the same as the turbine trip, except that steam dump is not operable. This results in an earlier but slower heat up of the RCS with an earlier moderator reactivity feedback. The steam generator inventory is maintained for longer time assuring a milder Steam Generator UA transient since no inventory is lost via steam dump. This event is analyzed in subsection 4.2.2 (see Case 3).

4.2.4 Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

These transients are no worse than the turbine trip. If the condenser is not available, the steam dump is not operable and hence the considerations made for the Inadvertent Closure of the MSIV apply. This event is analyzed in subsection 4.2.2 (see Case 3).

4.2.5 Loss of ac Power

A complete loss of ac power to the station auxiliaries would result from a loss of offsite power combined with a trip of the turbine and generator. The loss of ac power, as postulated, would result in the coastdown of the reactor coolant pumps and a loss of main feedwater. The condenser would be not available and systems using the condenser, such as turbine bypass, would be inoperable. The loss of ac power to the control rod motor generator sets results in a loss of offsite power to the rod drive control mechanisms gripper coils and hence the rods would fall in the core independently of any protection system action. Hence, it is impossible to associate a loss of ac power ATWS to the PMS CMF. Should a mechanical failure avoid the reactor trip, the transient is expected to be much milder than the loss of normal feedwater. In fact, following the reactor coolant pumps coastdown, the core average temperature would increase such that, as effect of the moderator reactivity feedback, core power would be immediately reduced. Heat transfer from the secondary system would be guaranteed by the large initial steam generator inventory. Once the low low SG level is reached, DAS actuates the PRHR assuring the required heat sink. Core power will balance PRHR heat transfer.

The time sequence of events for a loss of forced reactor coolant flow analyzed assuming the unavailability of the feedwater and condenser, reported in subsection 4.3, Table 11, is fully applicable to the loss of ac power. The behavior of the main plant parameters is illustrated in Figures 55 through 60.

4.2.6 Loss of Normal Feedwater

Loss of normal feedwater is the most limiting ATWS initiator transient. Three cases are analyzed in the following to show the plant behavior.

Case 4) Loss of normal feedwater event with turbine bypass system operable

This is the worst ATWS transient for the Westinghouse PWR plants. It is characterized by an initial transient resulting in a loss of secondary coolant while RCS parameters are maintained close to their initial values. Mitigating actions are provided by DAS that is expected to trip the reactor and the turbine. While in the analysis reported hereafter, no reactor trip is credited under the assumption of a mechanical CMF[1]. For the same event concomitant with a PMS or RTB CMF, the DAS actuation would trip the reactor and essentially turn the pressure down immediately.

Under the analyses assumptions, the continuous loss of secondary coolant eventually results in a fast degradation of the steam generator capability. The large mismatch between the core power and the steam generators heat transfer capability results in a large PRZ insurge, that leads soon to the overfilling, while power level is still high. No other anticipated transient can lead to this condition that makes the loss of normal feedwater the worst ATWS event.

The time sequence of events for this case is reported in Table 6. The behavior of the main plant parameters is illustrated in Figures 19 through 27.

TABLE 6
SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER – CASE 4
TURBINE BYPASS OPERABLE

Time (Seconds)	Event
0–4.	Loss of main feedwater (s)
17.	Turbine bypass valves begin modulating
~45.	PMS low narrow range SG water level set point reached (s)
Not Credited	PMS turbine trip
~62.6.	DAS low wide range SG water level set point reached (s)
64.6	Turbine trip (TxT) (s)
——	Reactor trip – not credited due to mechanical CMF
~69.	PRZ safety valves open (s)
~70	Turbine bypass valves fully open
74.6	PRHR actuation on DAS signal
Not Credited	PMS "S" signal on low steam line pressure • Isolate steam line (steam dump) • Actuate CMTs
104.	PRZ overfills (s) – power level = 30% SG UA = 0.2% → QXSG ~ 0%
116	Max RCS pressure = 3135 psia
178	PRZ safety valves close (s)

[1] Mechanical CMF only account for 2% of the ATWS being the remaining 98% due to PMS or RTB CMF.

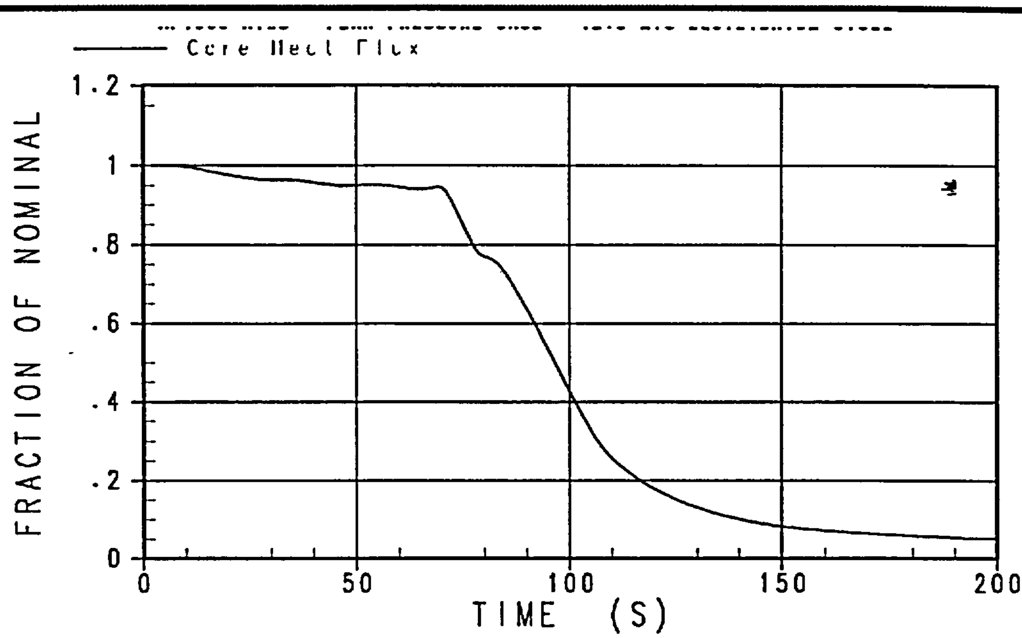


Figure 19 **ATWS Loss of Normal Feedwater – Steam Dump Operable Core Heat Flux (f.o.n) vs. Time**

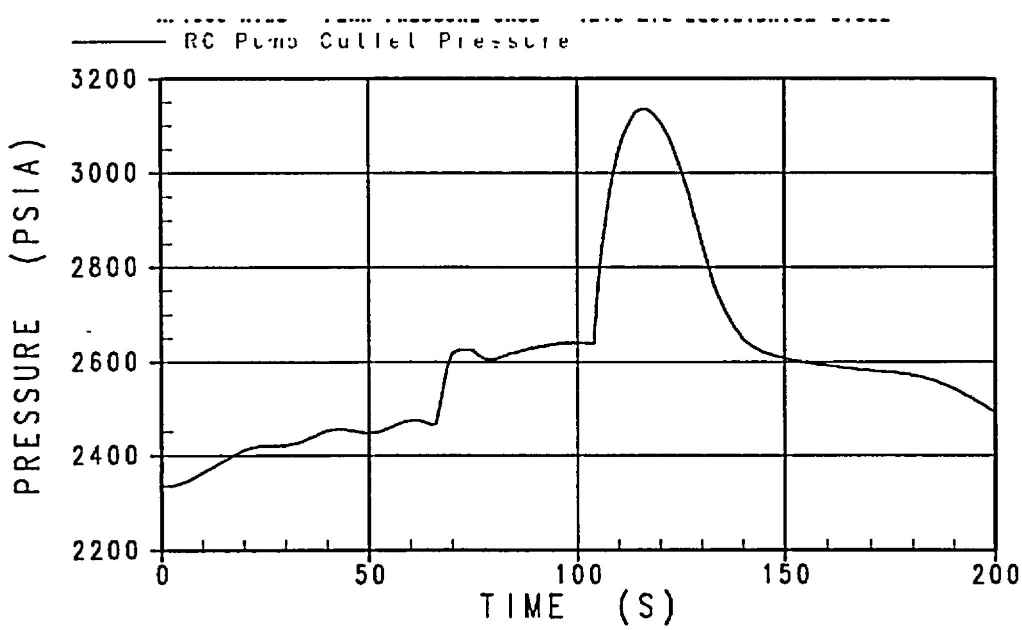


Figure 20 **ATWS Loss of Normal Feedwater – Steam Dump Operable Reactor Coolant Pump Outlet Pressure (psia) vs. Time**

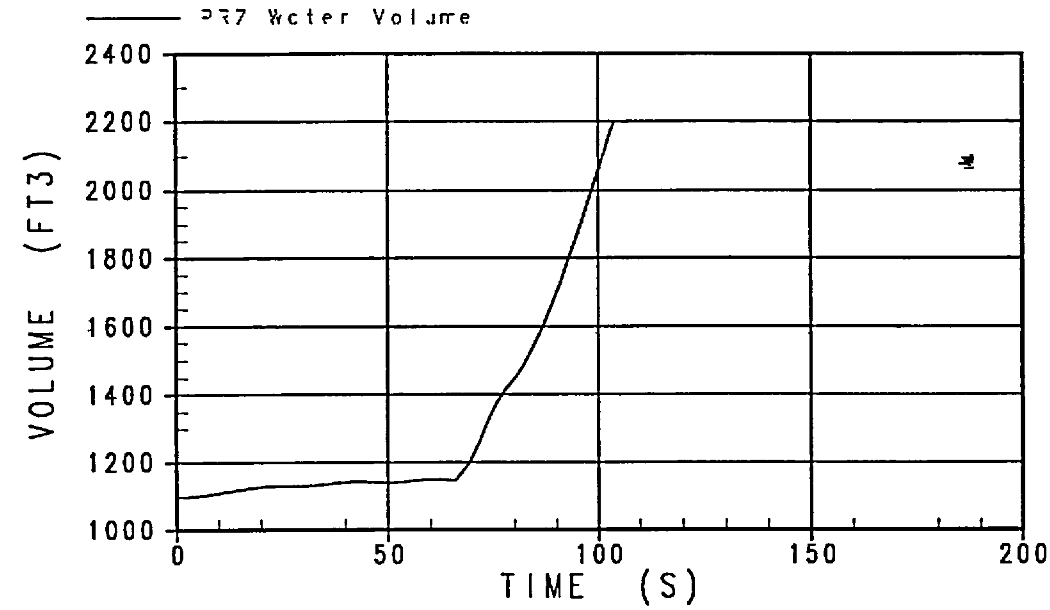


Figure 21 ATWS Loss of Normal Feedwater – Steam Dump Operable Pressurizer Water Volume (ft³) vs. Time

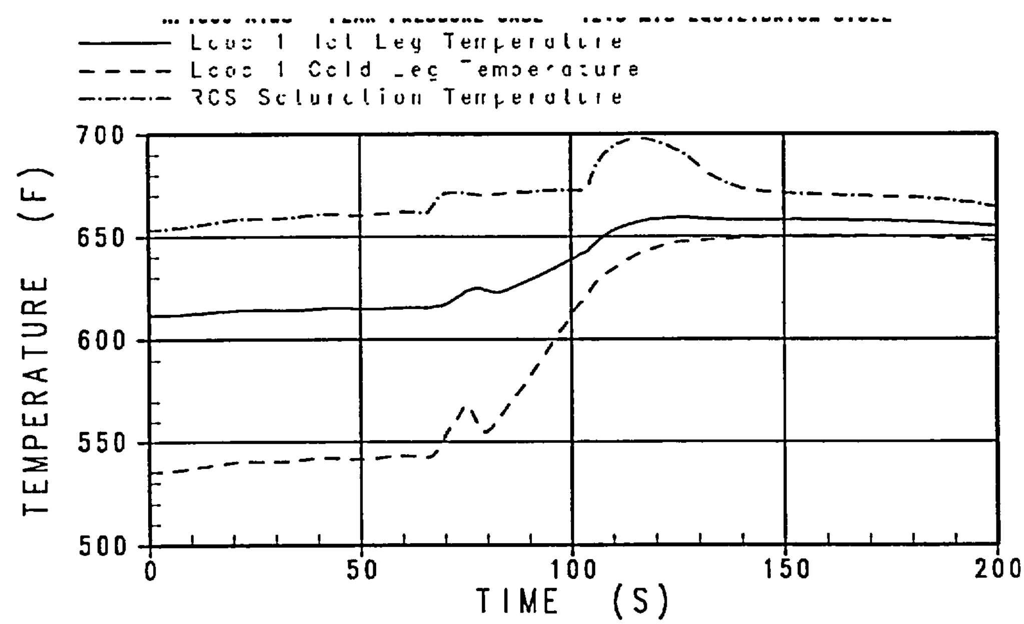


Figure 22 ATWS Loss of Normal Feedwater – Steam Dump Operable Loop 1 Coolant Temperatures (°F) vs. Time

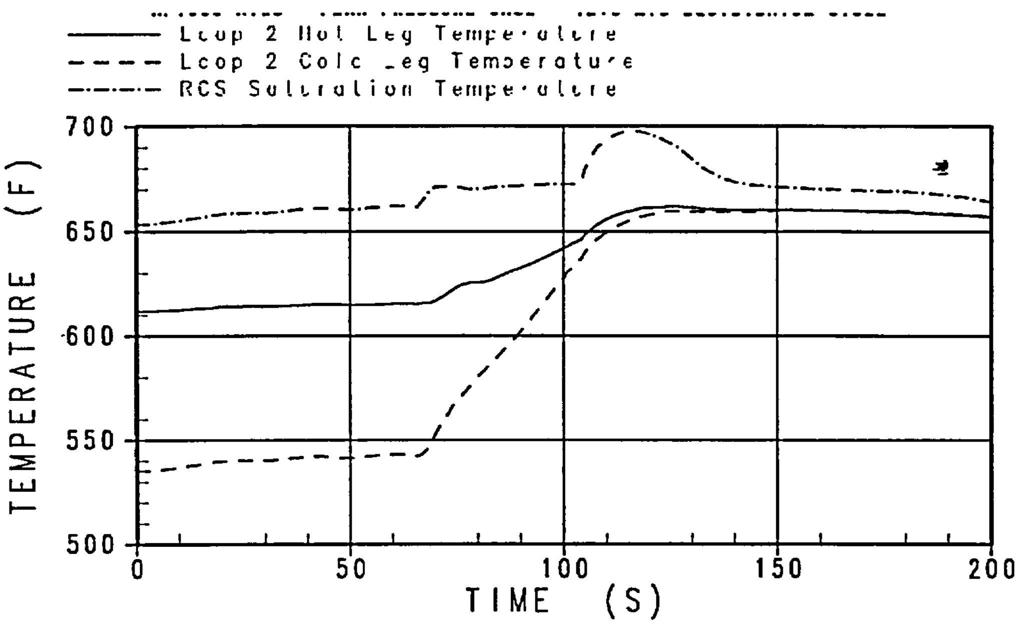


Figure 23 ATWS Loss of Normal Feedwater – Steam Dump Operable Loop 2 Coolant Temperatures (°F) vs. Time

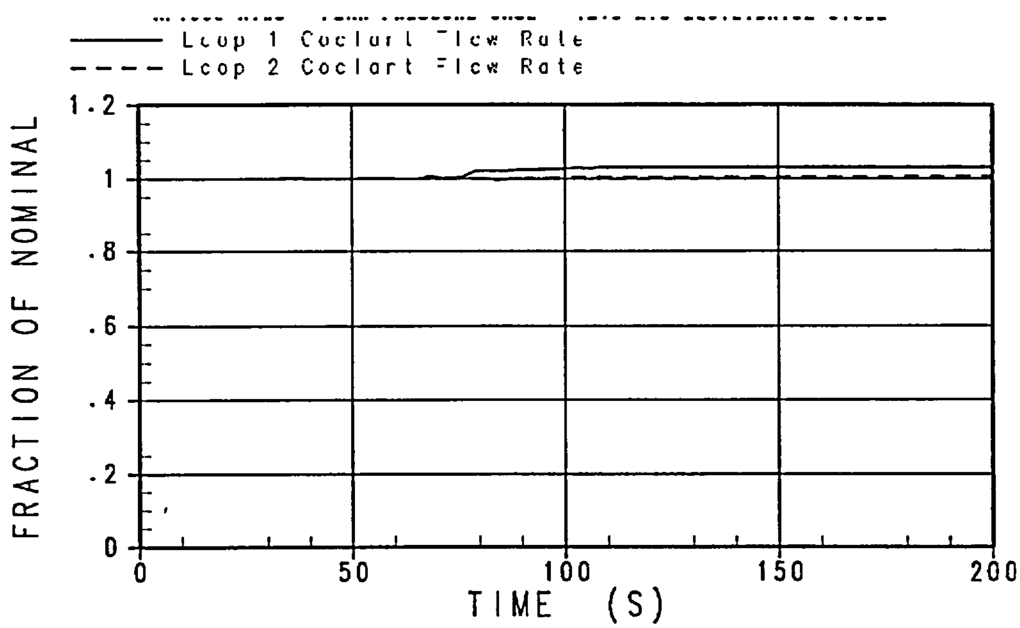


Figure 24 ATWS Loss of Normal Feedwater – Steam Dump Operable Loops Coolant Flow Rates (f.o.n) vs. Time

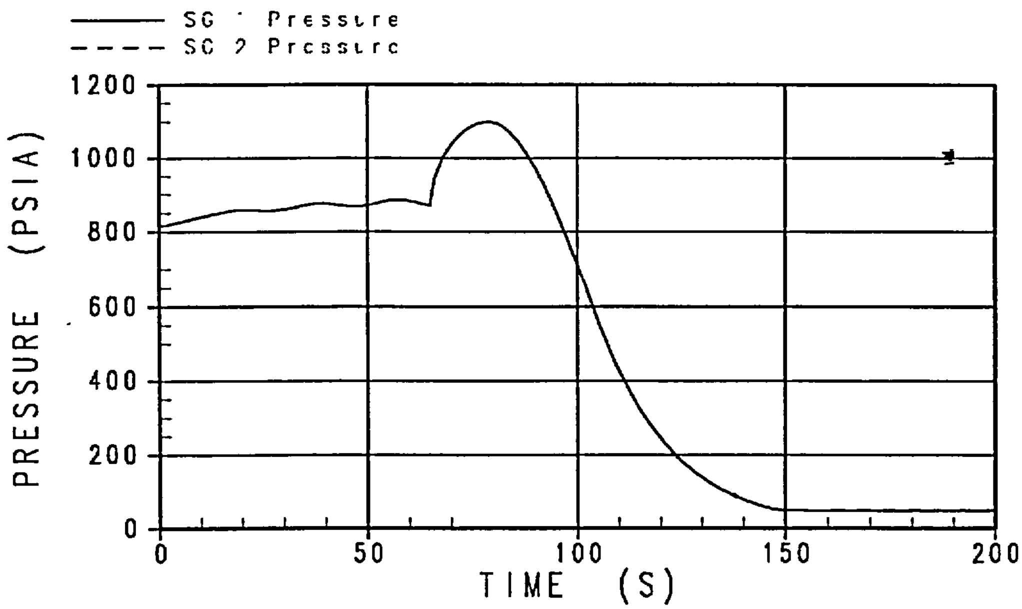


Figure 25 **ATWS Loss of Normal Feedwater – Steam Dump Operable Steam Generators Pressure (psia) vs. Time**

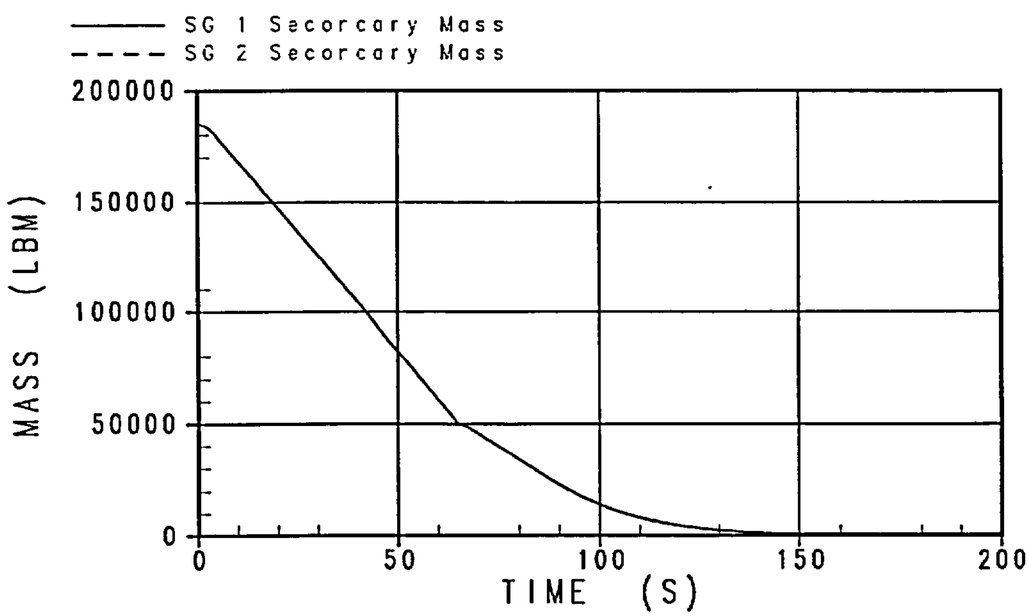


Figure 26 **ATWS Loss of Normal Feedwater – Steam Dump Operable Steam Generators Secondary Side Inventory (lbm) vs. Time**

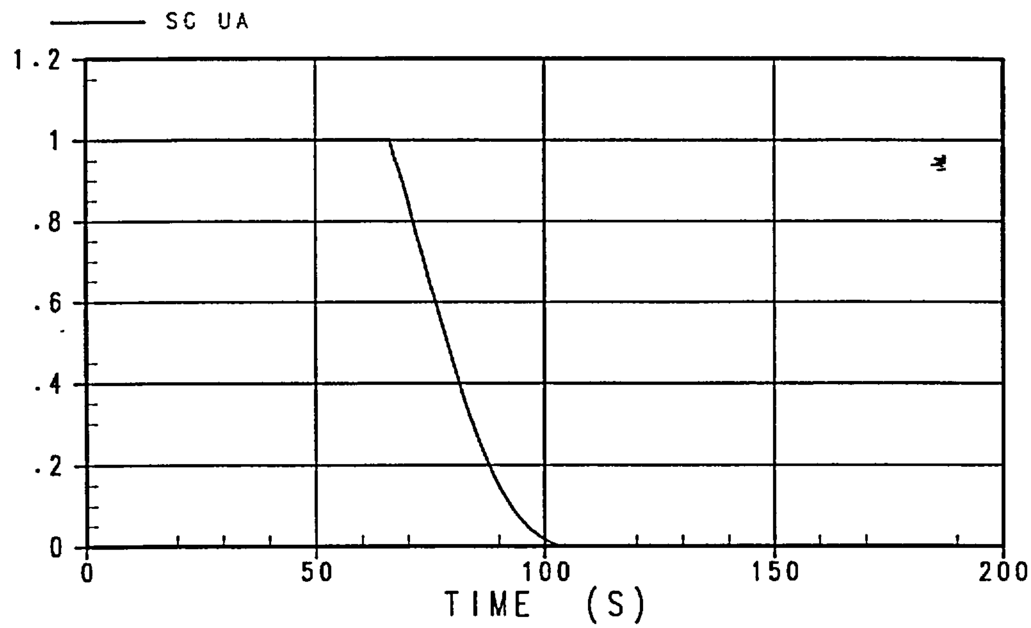


Figure 27 **ATWS Loss of Normal Feedwater – Steam Dump Operable Steam Generators Overall UA (f.o.i) vs. Time**

Case 5) Loss of normal feedwater event without turbine bypass system operable

This case is the same as the previous one, but steam dump system is assumed to be unavailable. This case is less limiting than the previous one since, after turbine trip, steam is relieved through the SG safety valves at a higher pressure than in the previous case. This results in higher RCS temperatures and lower core power levels at the time the PRZ overfills. Moreover, since, in this case, less water is lost from the secondary side, there is still some water left at the time of PRZ overfill and a portion of core power is exchanged with the secondary system, resulting in a less severe power mismatch.

The time sequence of events for this case is reported in Table 7. The behavior of the main plant parameters is illustrated in Figures 27 through 33.

TABLE 7
SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER – CASE 5
TURBINE BYPASS NOT OPERABLE

Time (Seconds)	Event
0–4.	Loss of main feedwater (s)
----	Turbine bypass valves begin modulating
~45.	PMS low narrow range SG water level set point reached (s)
Not Credited	PMS turbine trip
~64.5	DAS low wide range SG water level set point reached (s)
66.5	Turbine trip (TxT) (s)
-----	Reactor trip – not credited due to mechanical CMF
~69.	PRZ safety valves open (s)
-----	Turbine bypass valves fully open
76.5	PRHR actuation on DAS signal
108.	PRZ overfills (s) – power level = 27.7% SG UA = 2.5% → QXSG = 3.9%
118	Max RCS pressure = 2949 psia
185	PRZ safety valves close (s)

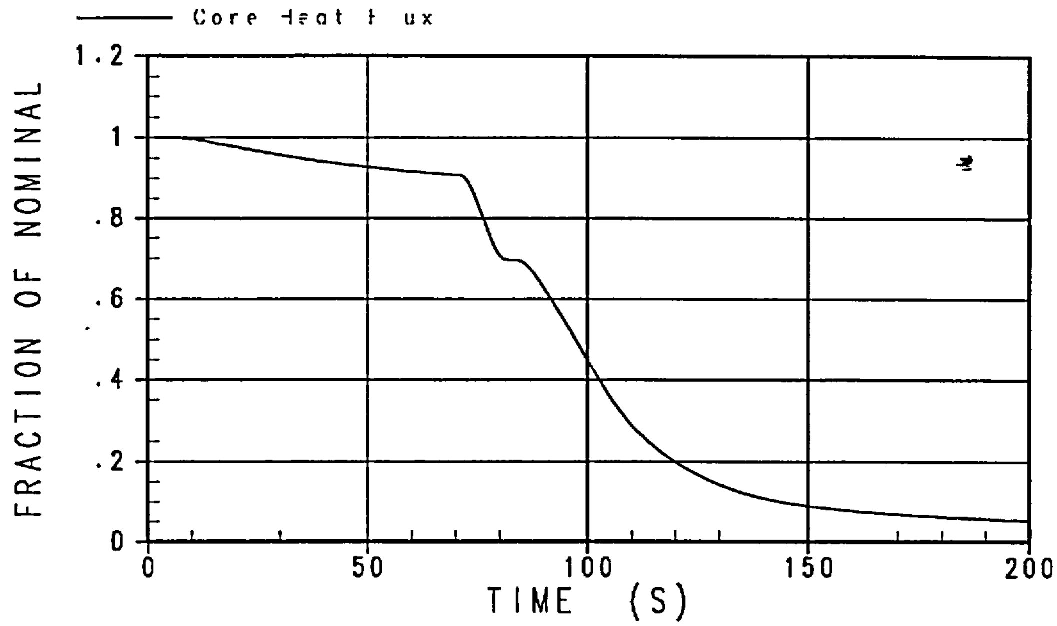


Figure 27 **ATWS Loss of Normal Feedwater – Steam Dump Not Operable Core Heat Flux (f.o.n) vs. Time**

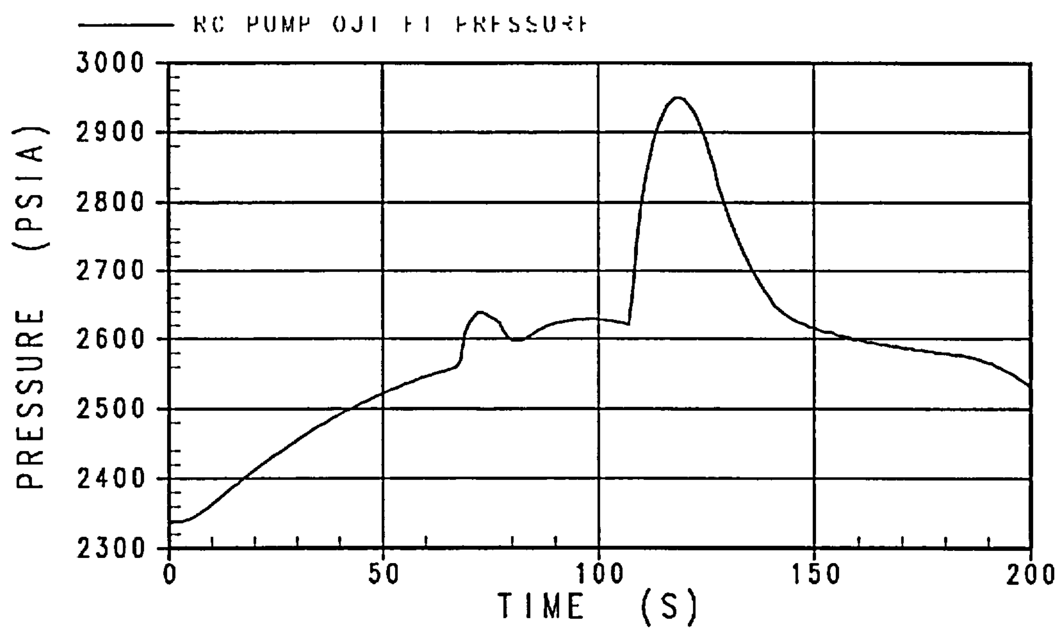


Figure 28 **ATWS Loss of Normal Feedwater – Steam Dump Not Operable Reactor Coolant Pump Outlet Pressure (psia) vs. Time**

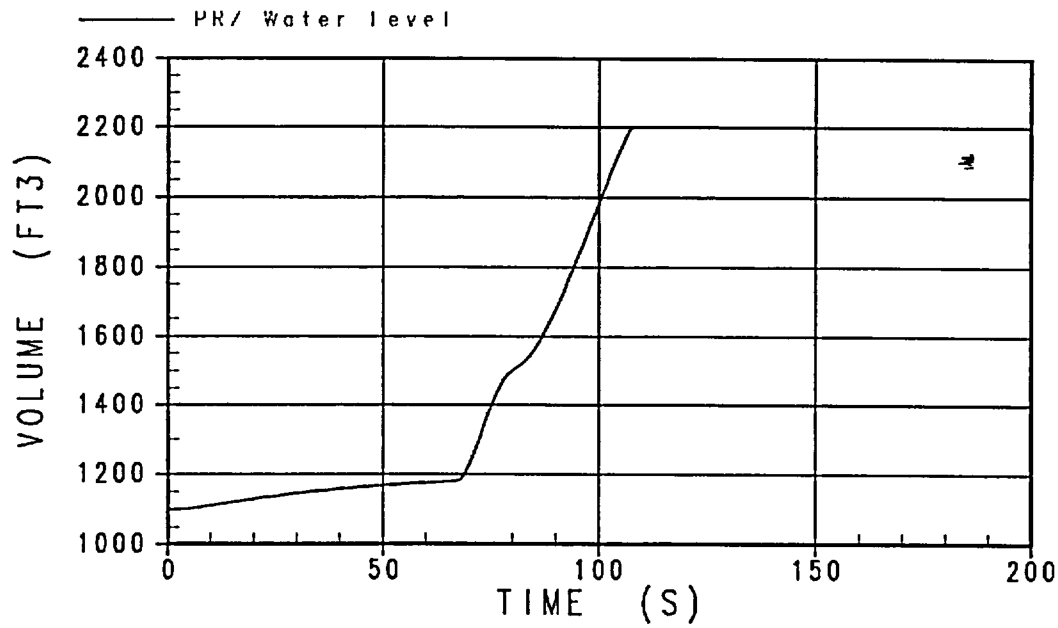


Figure 29 ATWS Loss of Normal Feedwater – Steam Dump Not Operable PRZ Water Volume (ft^3) vs. Time

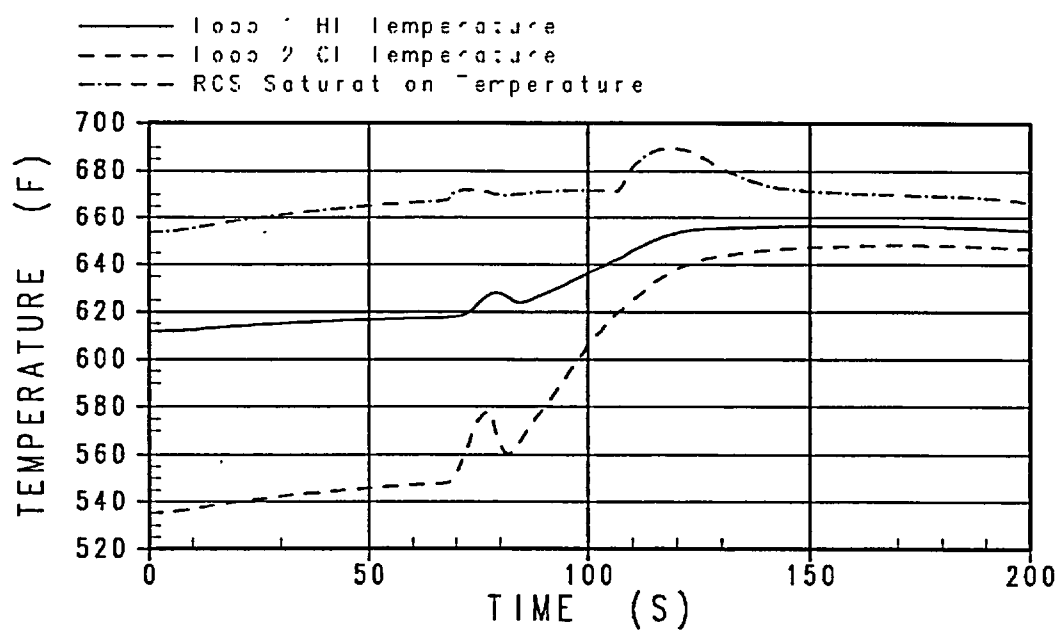


Figure 30 ATWS Loss of Normal Feedwater – Steam Dump Not Operable Loop 1 Coolant Temperatures (°F) vs. Time

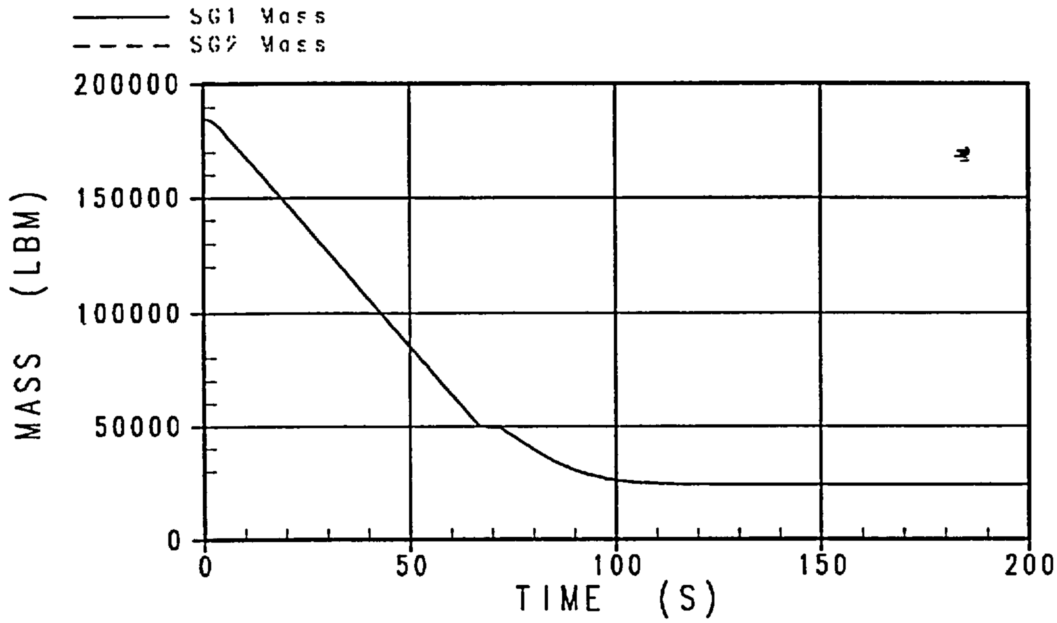


Figure 31 ATWS Loss of Normal Feedwater – Steam Dump Not Operable Steam Generators Mass (lbm) vs. Time

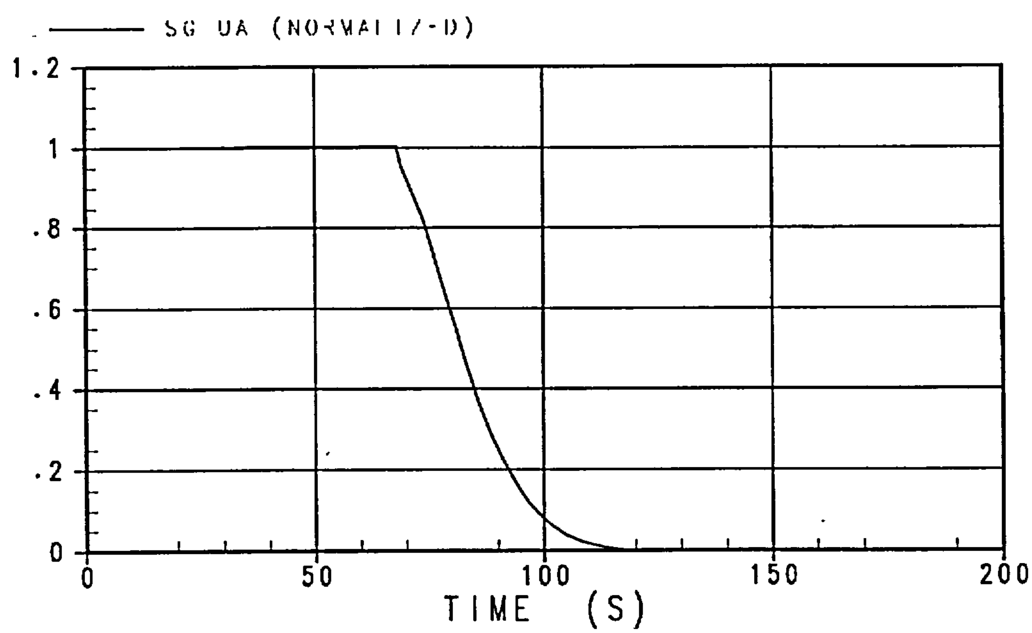


Figure 32 ATWS Loss of Normal Feedwater – Steam Dump Not Operable Normalized Steam Generators UA (ft^3) vs. Time

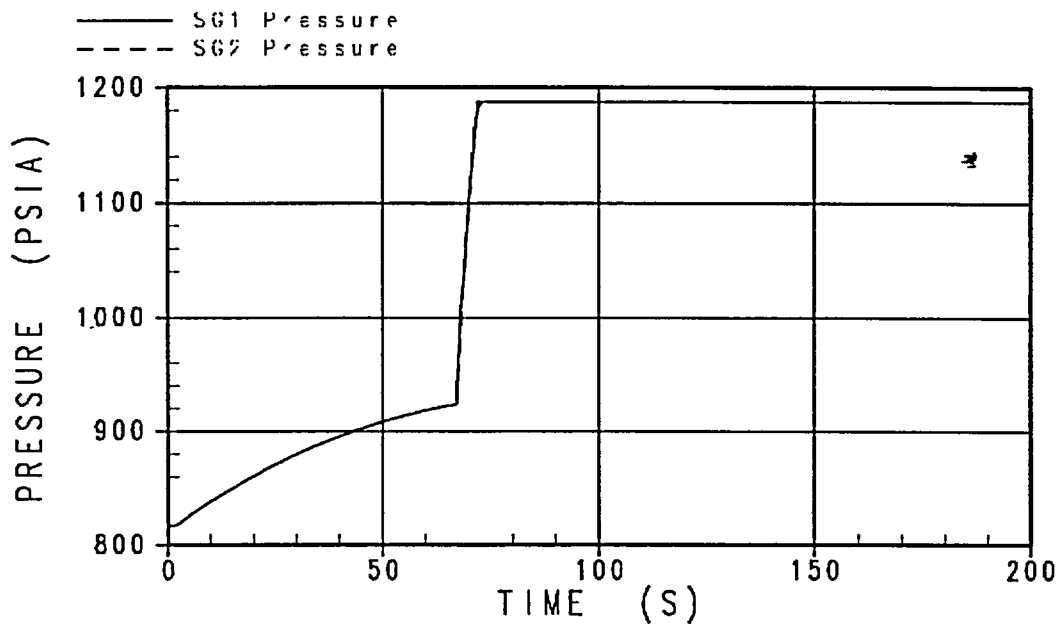


Figure 33 ATWS Loss of Normal Feedwater – Steam Dump Not Operable Steam Generators Pressure (psia) vs. Time

Case 6) Loss of normal feedwater event with turbine bypass system operable, more realistic SG UA

The Steam Generator UA adopted for the analysis of this case has been evaluated using a more realistic steam generator model. The SG UA has been evaluated via the following equation:

$$UA = Wcore/(DT1 - DT2)/\ln(DT1/DT2)$$

Where:

Wcore	=	Core Power (W)
DT1	=	SG Tube Inlet Temperature Difference = $T_{hot} - T_{sat}$ (°F)
DT2	=	SG Tube Outlet Temperature Difference = $T_{cold} - T_{sat}$ (°F)
T_{sat}	=	SG Secondary Side Saturation Temperature (°F)
T_{hot}	=	SG Tube Inlet Temperature (°F)
T_{cold}	=	SG Tube Outlet Temperature (°F)

The UAs calculated with the formula above have then been normalized to the initial value.

A transient calculation of the Δ125 steam generator model has been run with the following boundary conditions:

1. Constant inlet flow rate
2. Constant hot leg temperature
3. Nominal full steam flow up to the time of turbine trip (evaluated at the time of low SG water level – WR)
4. 50% of nominal full steam flow from the time of turbine trip on (to account for steam dump operation)

The outcome can be summarized as follows:

1. Steam Generator UA is almost constant up to about 60,000 lbm
2. Between 60,000 and 55,000 SG UA increases slightly above the initial value
3. UA starts dropping as soon as the turbine is tripped (UA = 0.92 at 50,000 lbm, note that turbine trip setpoint is set at 55,000 lbm)
4. At about 20,000 lbm SG UA is about 7% of the nominal
5. At about 16,500 lbm SG UA drops down to about 3% (only steam in the SG)
6. SG UA eventually decreases to zero (for a steam mass of 5000 lbm)

A conservative approximation of the UA characteristics summarized above are then used in the LOFTRAN model. The time sequence of events for this case is reported in Table 8. The behavior of the main plant parameters is illustrated in Figures 34 through 37.

TABLE 8 SEQUENCE OF EVENTS FOR LOSS OF NORMAL FEEDWATER – CASE 6 TURBINE BYPASS OPERABLE – REALISTIC UA	
Time (Seconds)	**Event**
0–4.	Loss of main feedwater (s)
17.	Turbine bypass valves begin modulating
~45.	PMS low narrow range SG water level set point reached (s)
Not Credited	PMS turbine trip/PMS reactor trip
~62.7	DAS low wide range SG water level set point reached (s)
64.7	Turbine trip (TxT) (s)
------	Reactor trip – not credited due to mechanical CMF
~64.	PRZ safety valves open (s)
~70	Turbine bypass valves fully open
74.7	PRHR actuation on DAS signal
Not Credited	PMS "S" signal on low steam line pressure • Isolate steam line (steam dump) • Actuate CMTs
99.	PRZ overfills (s) – power level = 27% SG UA = 2.1% → QXSG = 8.2%
104.	Max RCS pressure = 2748 psia
> 200	PRZ safety valves close (s)

The following figures show the impact of the realistic UA versus the conservative LOFTRAN model.

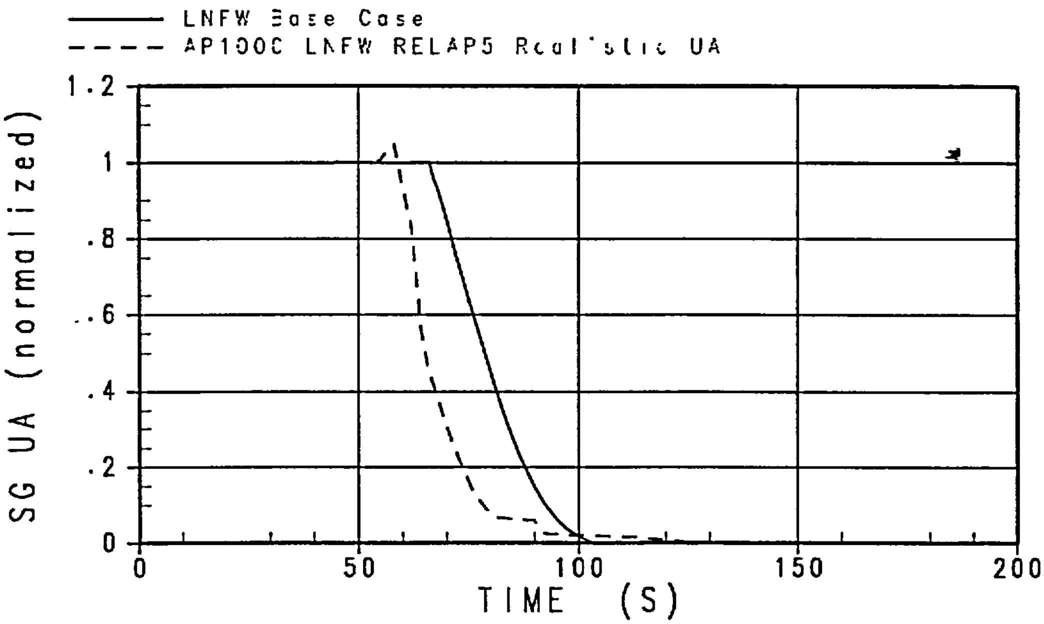


Figure 34 ATWS Loss of Normal Feedwater – Steam Dump Operable – Realistic UA Steam Generator UA (f.o.i) vs. Time (comparison with Worst Case)

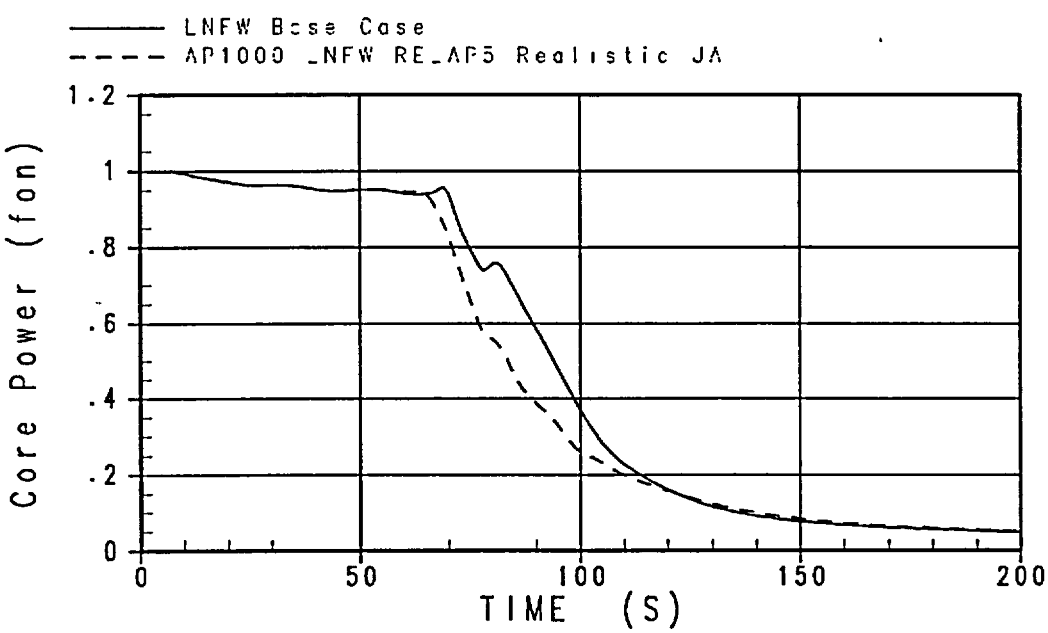


Figure 35 ATWS Loss of Normal Feedwater – Steam Dump Operable – Realistic UA Core Power (f.o.n) vs. Time (comparison with Worst Case)

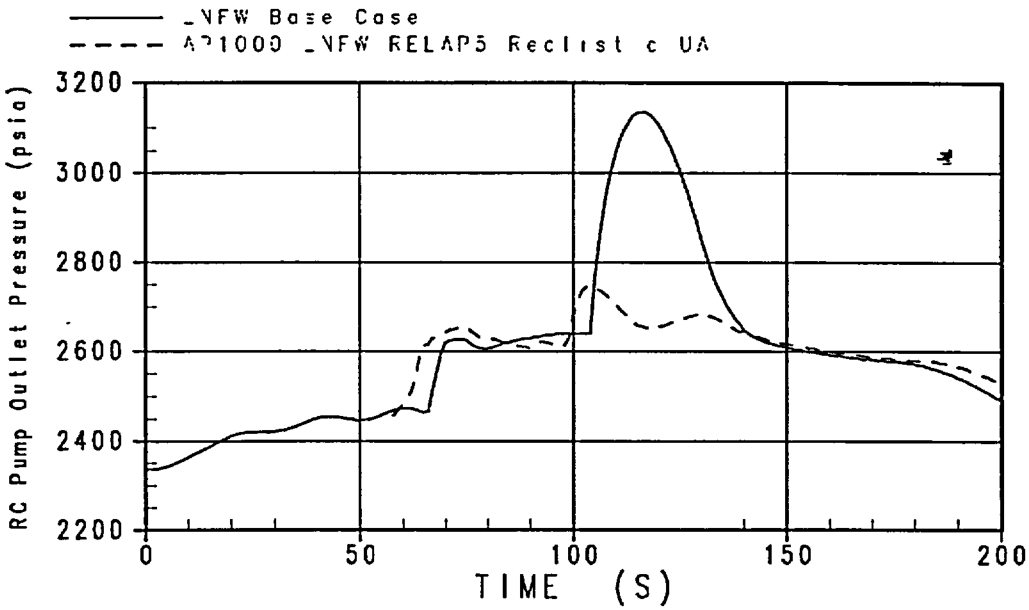


Figure 36 ATWS Loss of Normal Feedwater – Steam Dump Operable – Realistic UA RCP Outlet Pressure (psia) vs. Time (comparison with Worst Case)

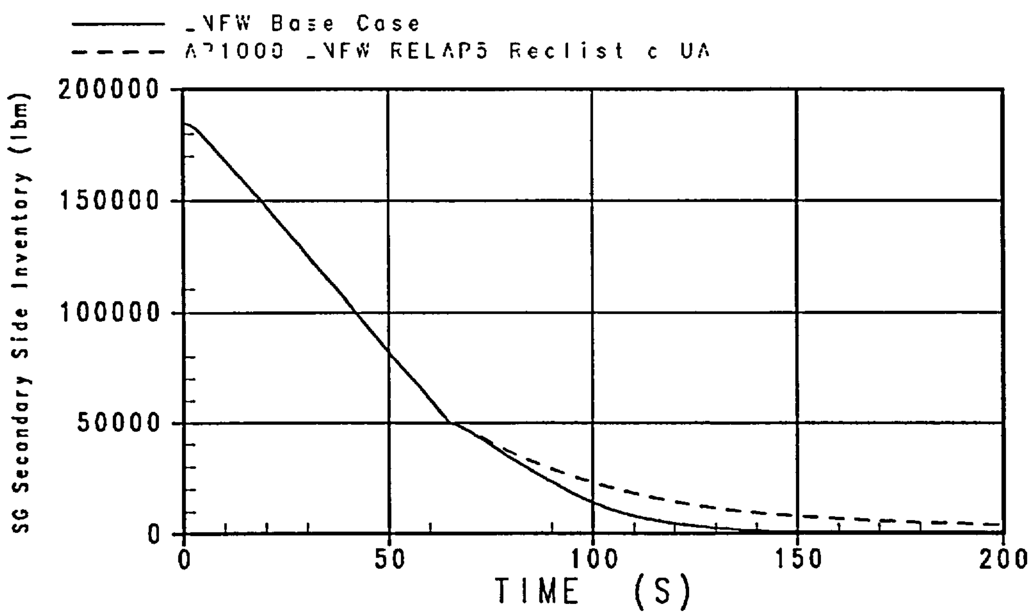


Figure 37 ATWS Loss of Normal Feedwater – Steam Dump Operable – Realistic UA Steam Generator Mass (lbm) vs. Time (comparison with Worst Case)

4.3 Decrease in Reactor Coolant System Flow Rate

4.3.1 Partial Loss of Forced Reactor Coolant Flow

The progression of a partial coolant flow event is similar to the complete loss of flow event. Due to the partial loss of flow, core average moderator temperatures quickly rise and voids are generated in the core. Nuclear power quickly drops while the heat transfer through the secondary side is assured by the initial water inventory. A DAS signal, on High HL temperature, is reached quite early in the transient. The transient is bounded by the complete loss of flow transient (see below).

4.3.2 Complete Loss of Flow Event

The complete loss of flow event is characterized by the coastdown of the reactor coolant pumps. Feedwater system is assumed to maintain steam generator level.

Following the pumps coastdown, hot leg temperature increases and eventually the DAS High HL Temperature setpoint is reached. DAS actuates the reactor trip, turbine trip and also actuates the PRHR. Should a mechanical failure avoid the reactor trip, the core power will reach a power level consistent with PRHR and steam generator heat transfer.

Following turbine trip steam dump may or may not be available and hence a different equilibrium power could be reached. In both case however, the steam generator inventory guarantees, with the PRHR operation, the heat sink required to avoid PRZ peak to exceed the 3200 psia limit. Eventually, operator will manually shut the plant down via DAS or via CMT boration.

Three analyses of Complete Loss of Flow ATWS events are reported in the following. The results of the cases analyzed demonstrate that the acceptance criteria are met with large margin.

Case 7) Complete loss of forced coolant flow with MFW system operable with turbine bypass system operable

The initiating event for this analysis is the trip of all of the RCPs at the time t=4 seconds.

Immediately after the RCPs trip, a number of signals calls for the reactor trip that is assumed to fail. Due to the reduced core flow rate, core outlet temperatures quickly reach saturation and core coolant density drops resulting in a negative reactivity insertion that causes a fast power decrease.

A DAS signal on high hot leg temperature setpoint (650°F) actuates the PRHR, trips the turbine and the reactor. Should the ATWS be the consequence of a PMS failure of Reactor Trip Breaker Failure, the DAS signal would terminate the accident.

However, even assuming that the rods do not enter the core, steam generators, fed by the main feedwater, are still able to extract almost 40% of the nominal power so that the core reaches a new power equilibrium value just above 40%.

Maximum RCS pressure, 2530 psia, is reached at about 31 seconds (at RCP pump outlet).

Assuming that no reactor trip occurs, due to mechanical common mode failures, the operator has sufficient time to shutdown the reactor by injecting borated water in the RCS via CMTs or CVS.

The time sequence of events for this case is reported in Table 9. The behavior of the main plant parameters is illustrated in Figures 38 through 45.

TABLE 9
SEQUENCE OF EVENTS FOR COMPLETE LOSS OF FORCED COOLANT FLOW – CASE 7
MAIN FEEDWATER AVAILABLE – TURBINE BYPASS OPERABLE

Time (Seconds)	Event
4.	Reactor coolant pumps trip
4.4	PMS reactor trip setpoint on low speed (no trip generated)
11.5	Steam dump valves fully open
16.	PRZ safety valves open
-----	Steam generator safety valves open
12.	HL temperature > DAS high HL temperature setpoint (650°F)
Not Credited	Reactor tripped on DAS signal above
14	Turbine trip on signal above
21	PRHR actuation on DAS signal above
----	PRZ overfills
31.	Maximum RCS pressure (at RCP outlet) – 2530 psia
> 500	PRZ safety valves close

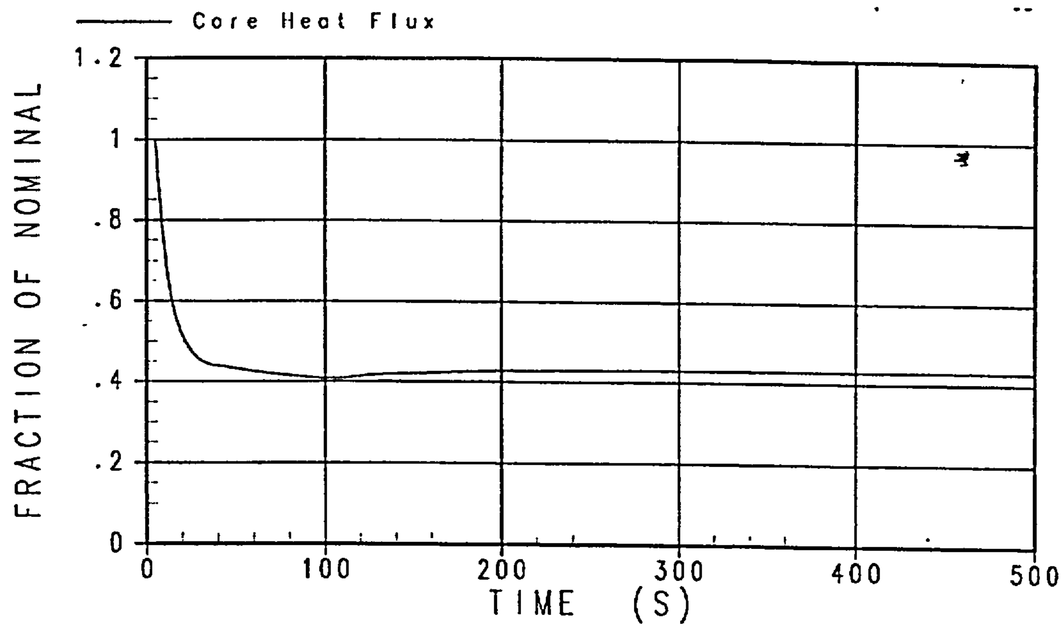


Figure 38 ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable Core Heat Flux (f.o.n) vs. Time

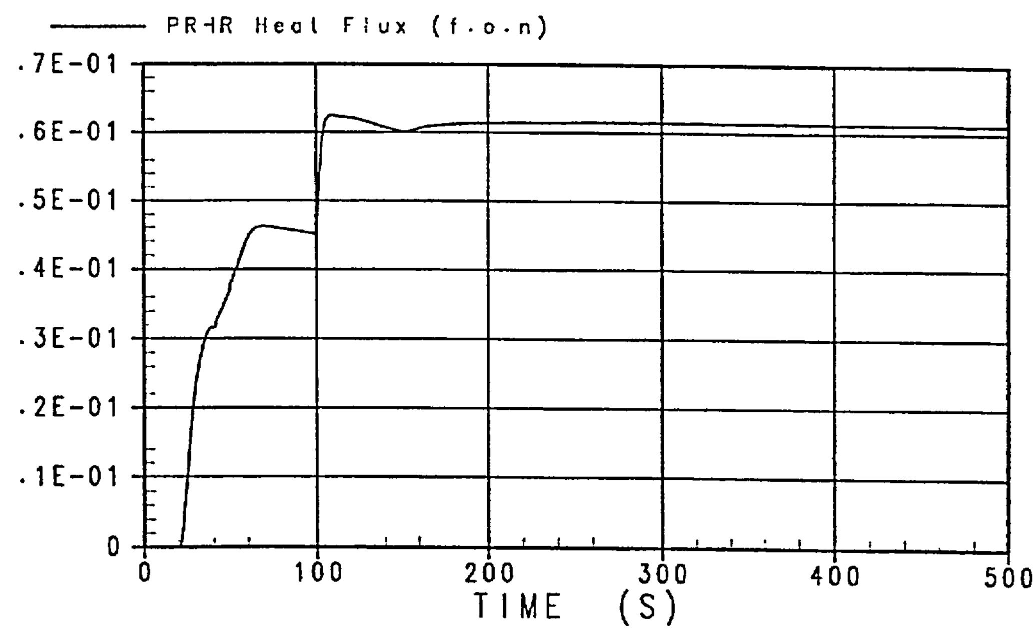


Figure 39 ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable PRHR Heat Flux (f.o.n) vs. Time

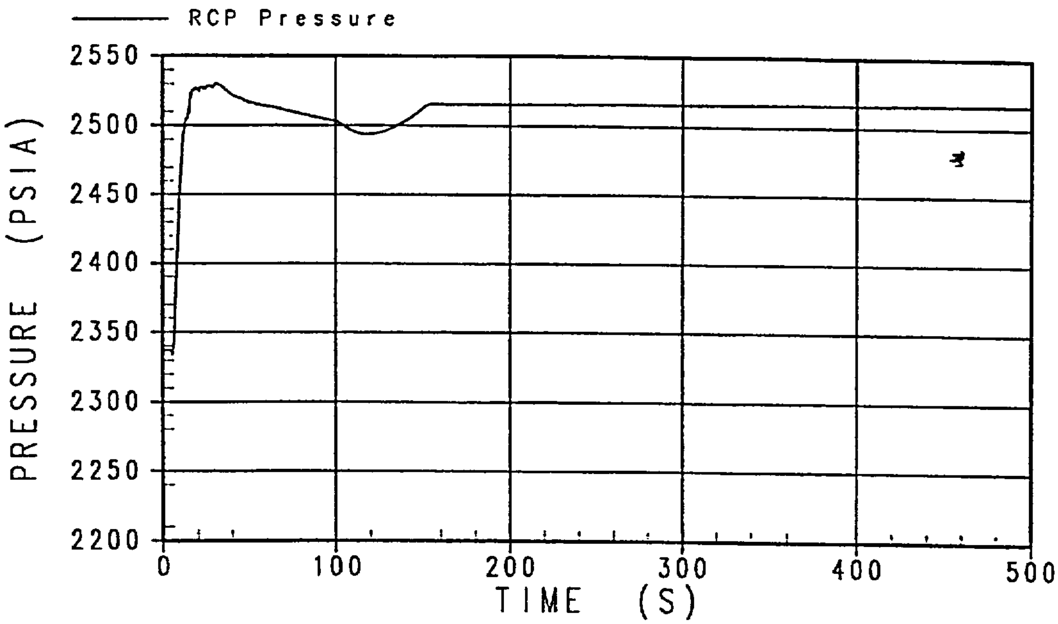


Figure 40 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable RCP Outlet Pressure (psia) vs. Time**

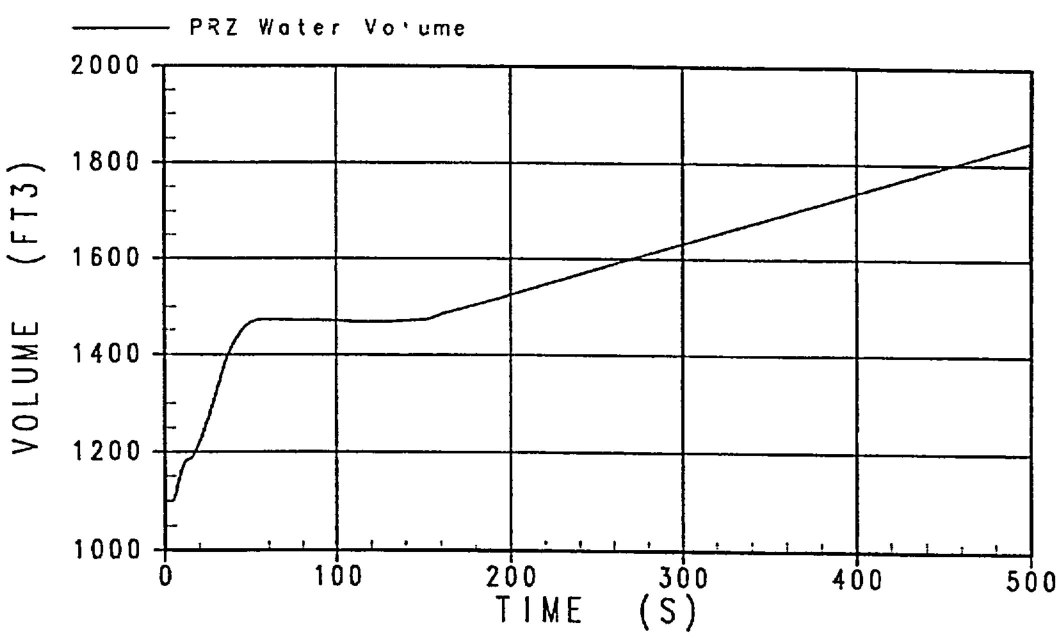


Figure 41 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable PRZ Water Volume (ft^3) vs. Time**

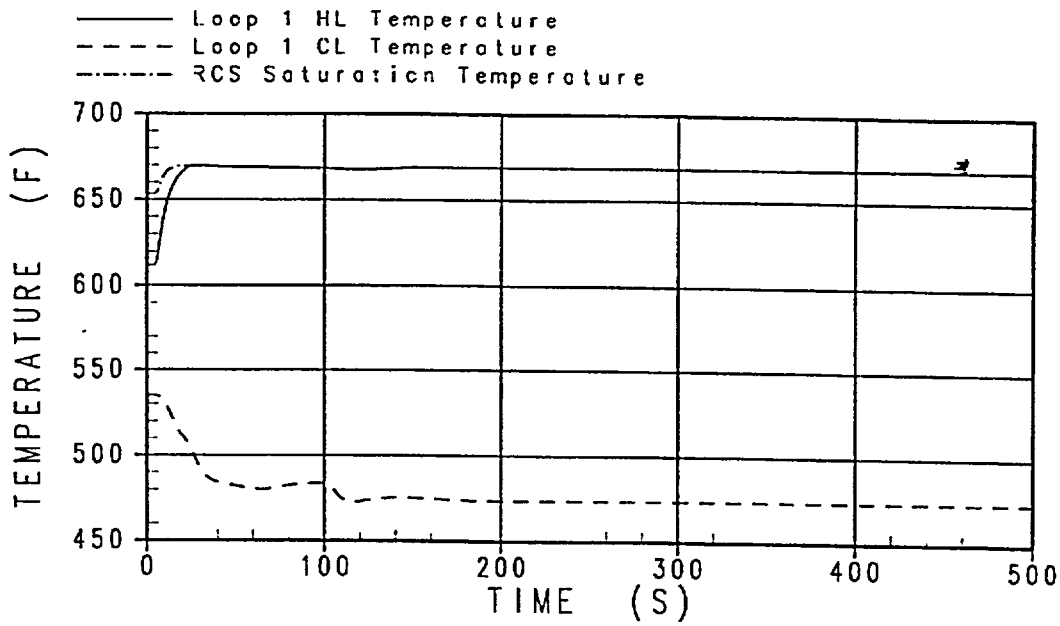


Figure 42 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable Loop 1 Coolant Temperatures (°F) vs. Time**

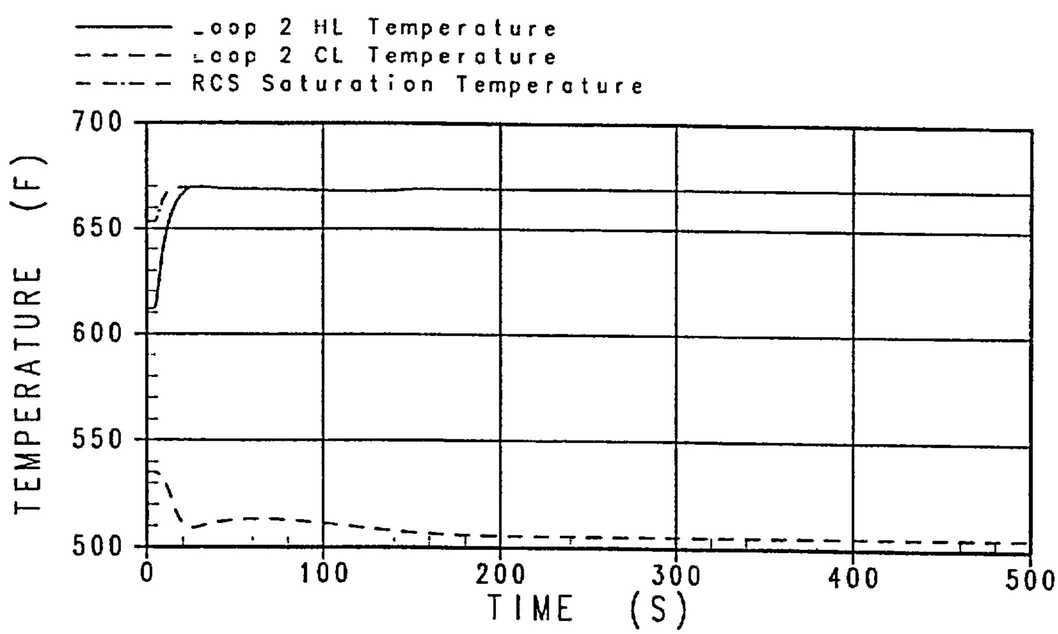


Figure 43 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable Loop 1 Coolant Temperatures (°F) vs. Time**

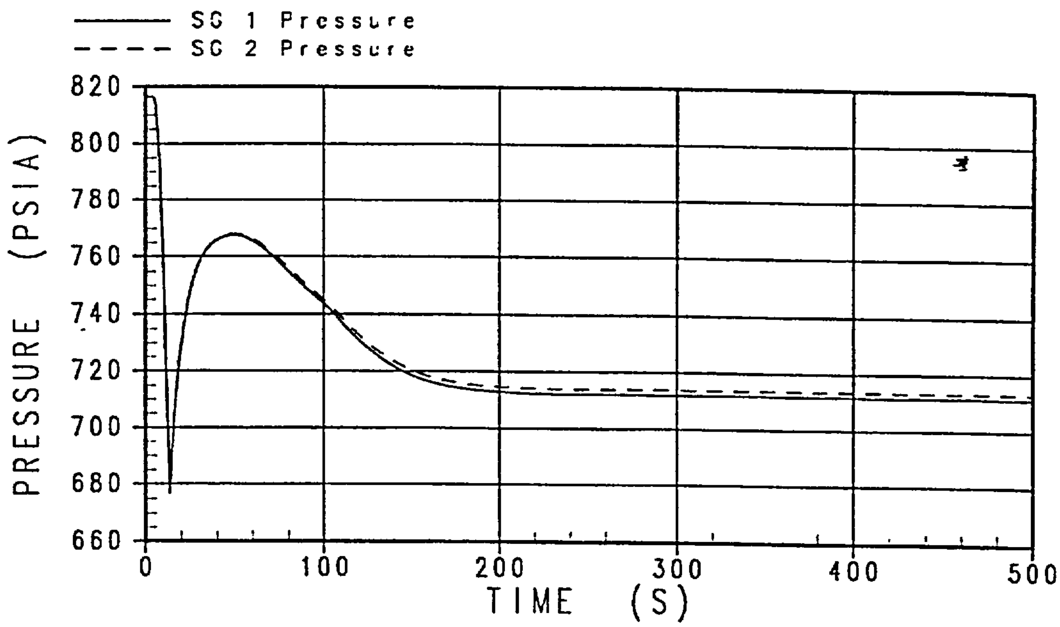


Figure 44 ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable Steam Generators Pressure (psia) vs. Time

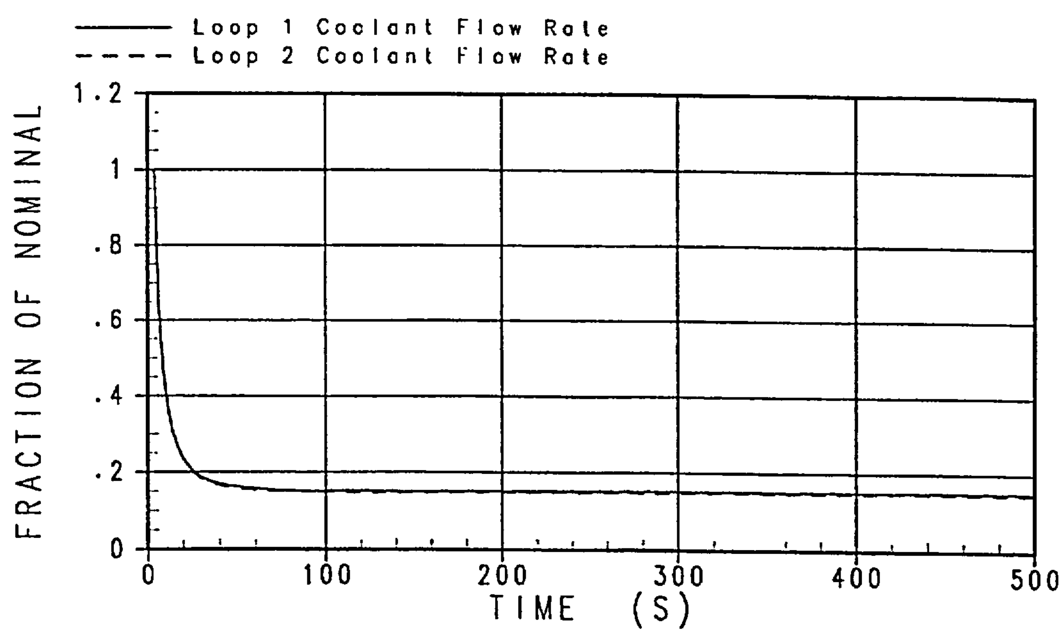


Figure 45 ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Operable Reactor Coolant Loops Flow Rate (f.o.n) vs. Time

Case 8) Complete loss of forced coolant flow with MFW system operable without turbine bypass system operable

This event is similar to the previous one, but turbine bypass is assumed not operable. Essentially, this results in a reduced heat transfer and in higher pressure and temperatures in the secondary side, and hence in a higher temperature of the reactor coolant. In turn, core power, due to the reactivity effects, reaches an equilibrium with the heat flux from the primary system (steam generator plus PRHR).

It should be noted that RCS pressure behavior does not show any particular peak after the first one, that is essentially due to the initiating event and to the subsequent turbine trip. In fact, steam generator heat transfer is kept at sufficient high values since secondary inventory is maintained by the feedwater system.

The time sequence of events for this case is reported in Table 10. The behavior of the main plant parameters is illustrated in Figures 46 through 52.

TABLE 10
SEQUENCE OF EVENTS FOR COMPLETE LOSS OF FORCED COOLANT FLOW – CASE 8
MAIN FEEDWATER AVAILABLE – TURBINE BYPASS NOT OPERABLE

Time (Seconds)	Event
4.	Reactor coolant pumps trip
4.4	PMS reactor trip setpoint on low speed (no trip generated)
-----	Steam dump valves fully open
16.	PRZ safety valves open
46.5	Steam generator safety valves open
12.	HL temperature > DAS high HL temperature setpoint (650°F)
Not Credited	Reactor tripped on DAS signal above
14	Turbine trip on signal above
24	PRHR actuation on DAS signal above
----	PRZ overfills – max PRZ water volume 1833. ft^3 margin to fill = 367. ft^3
32.	Maximum RCS pressure (at RCP outlet) – 2543 psia
> 500	PRZ safety valves close

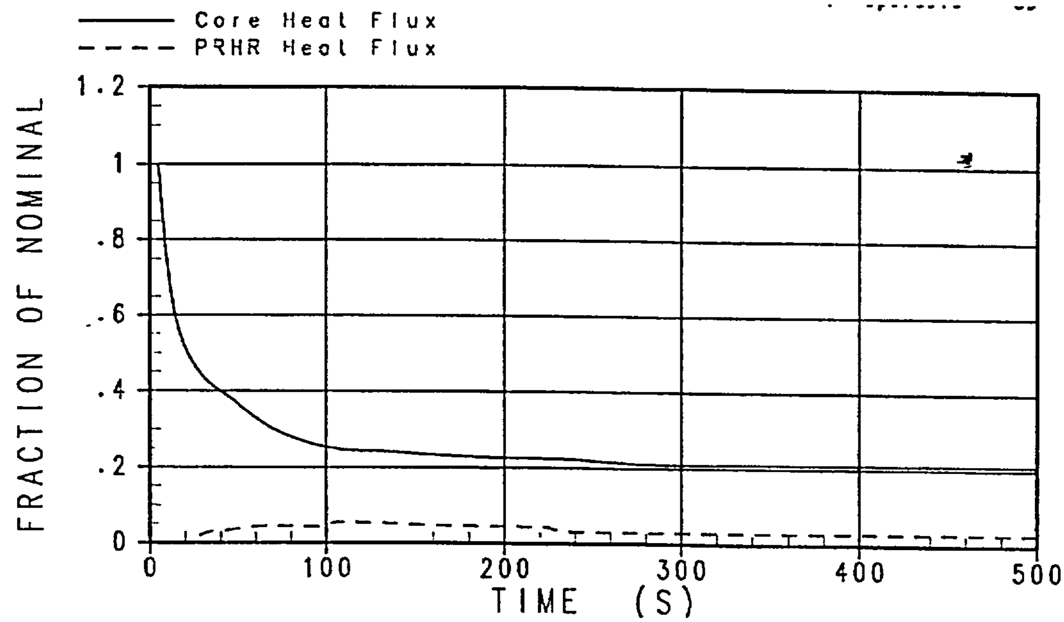


Figure 46 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Unavailable Core and PRHR Heat Flux (f.o.n) vs. Time**

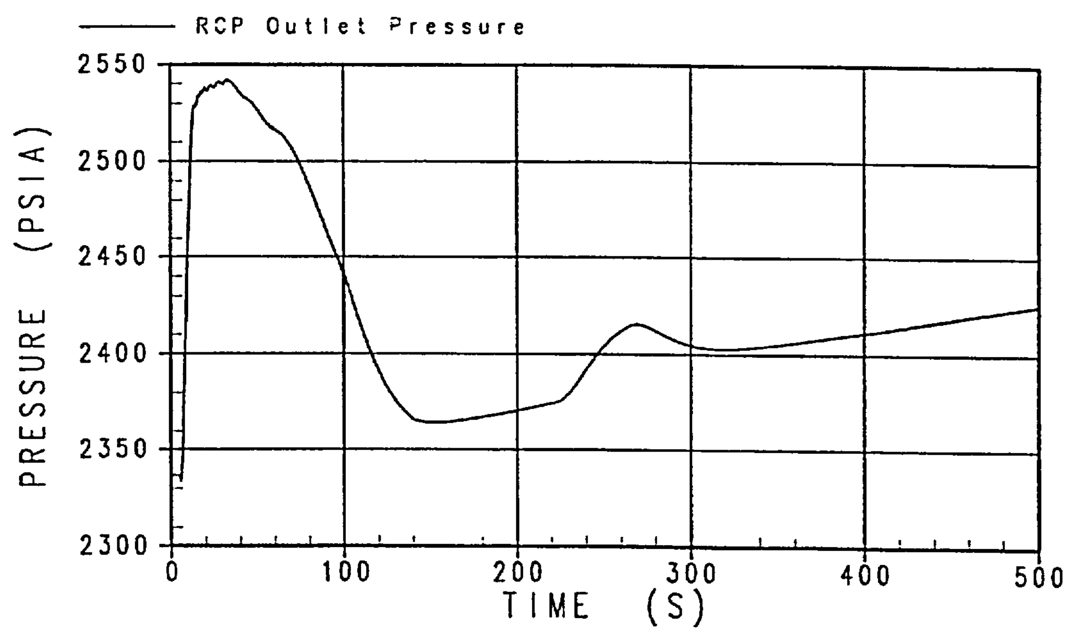


Figure 47 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Unavailable Reactor Coolant Pump Outlet Pressure (psia) vs. Time**

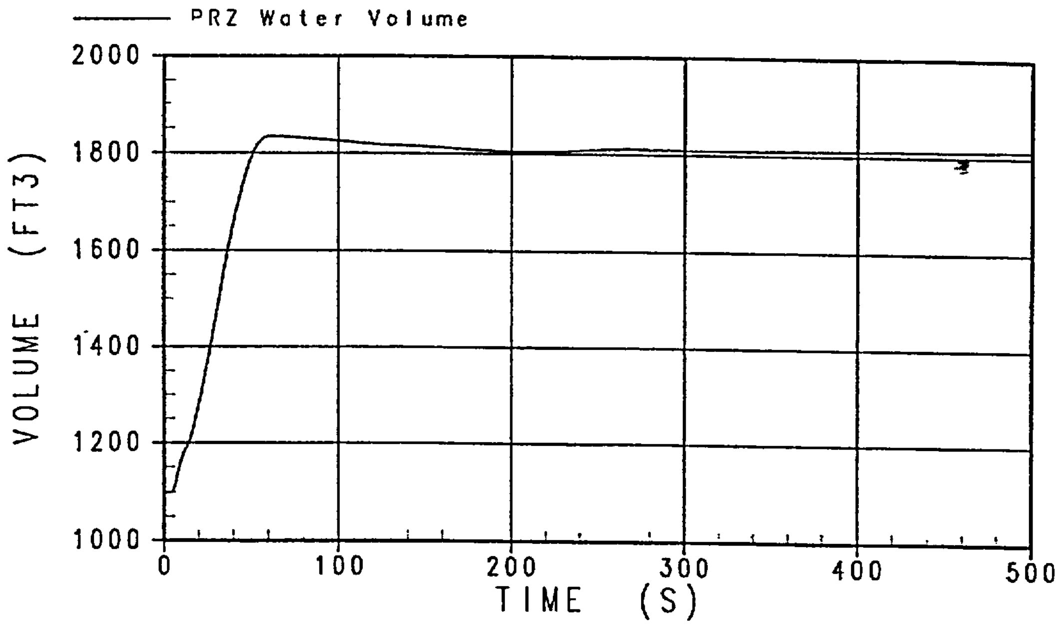


Figure 48 ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Unavailable Pressurizer Water Volume vs. Time

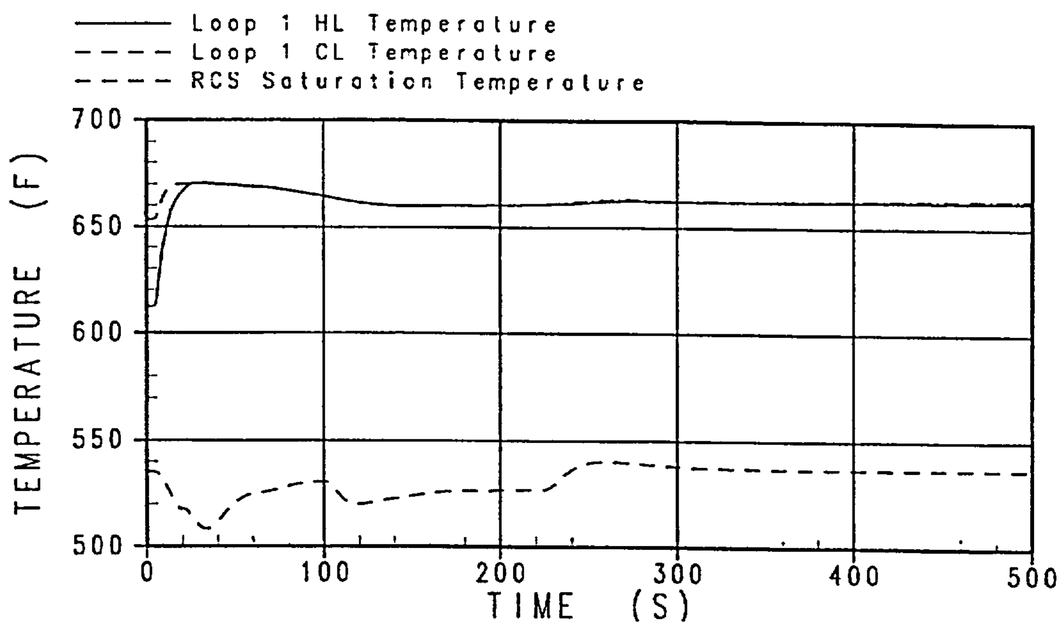


Figure 49 ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Unavailable Loop 1 Coolant Temperatures (°F) vs. Time

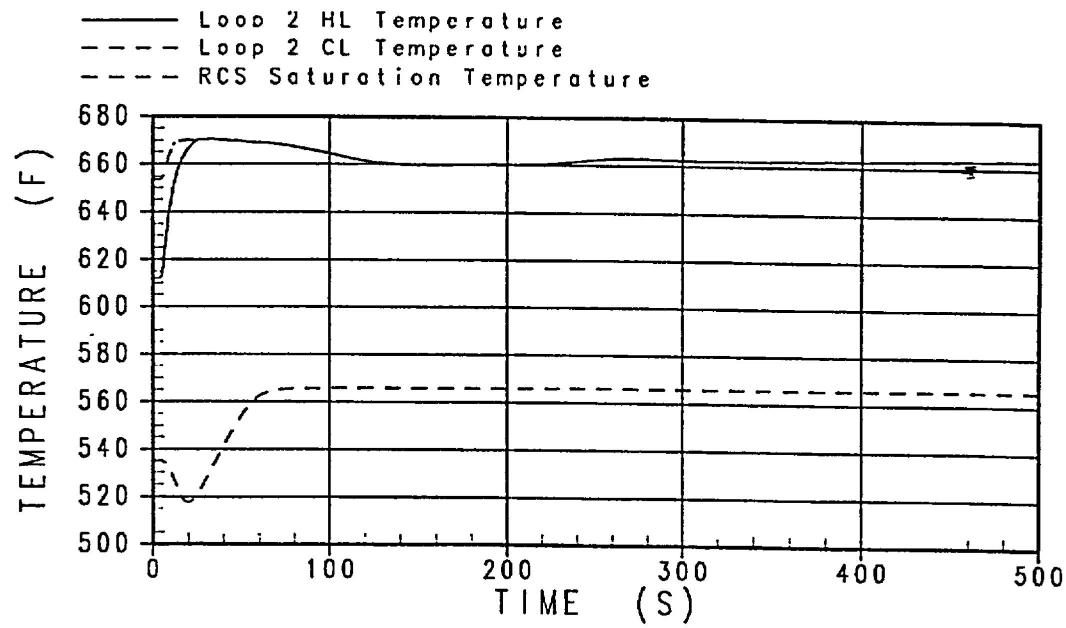


Figure 50 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Unavailable Loop 2 Coolant Temperatures (°F) vs. Time**

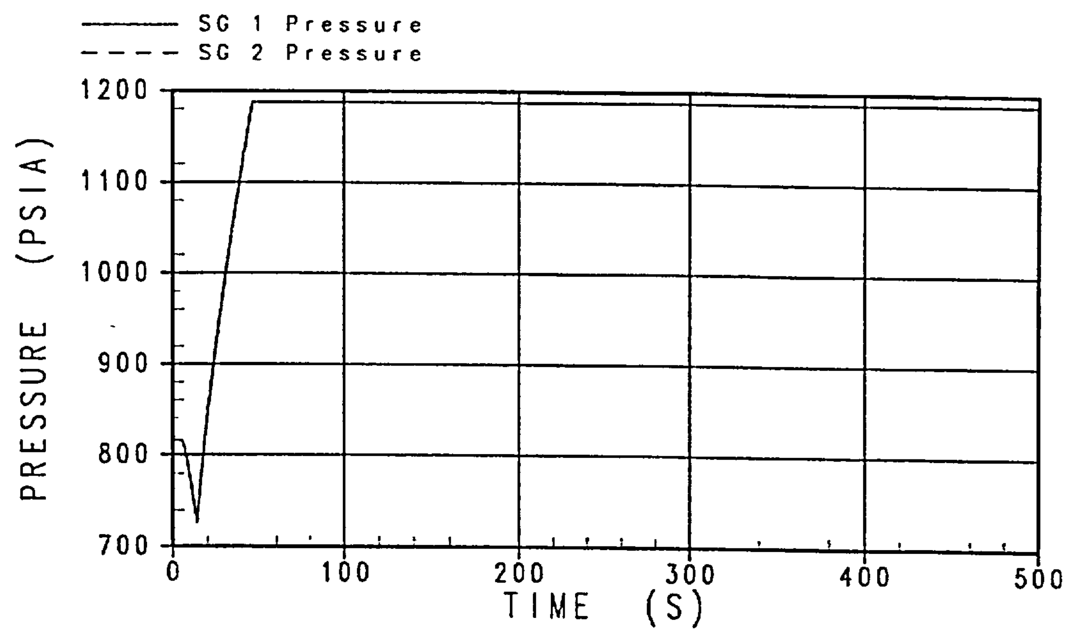


Figure 51 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Unavailable Steam Generators Pressure (psia) vs. Time**

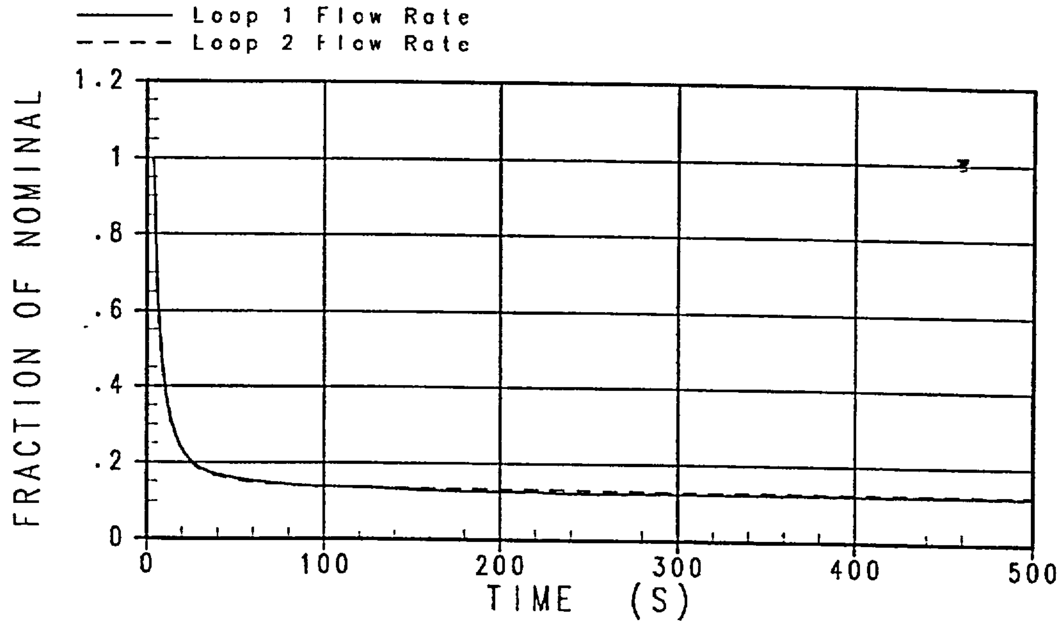


Figure 52 **ATWS Complete Loss of Flow – Feedwater Available, Steam Dump Unavailable Loop Coolant Flow Rate (f.o.n) vs. Time**

Case 9) Complete loss of forced coolant flow induced by the loss of ac power (mechanical CMF)

This event simulates all the ATWS events caused by a Loss of AC Power Event to the Plant Auxiliaries and a concomitant Mechanical Common Cause Failure. It should be noted that the Mechanical CMF is the only concomitant failure that can result in an ATWS starting from a loss of ac power as initiating event. In fact, should a concomitant PMS failure occur following a loss of ac power, the de-energization of the gripper coils would result in an immediate RCCA release and hence in a reactor trip.

The analysis of this event assumes that main feedwater system and steam dump system are not available. In addition, turbine trip is assumed to occur at the initiation of the transient.

With respect to the previous cases, the steam generator capability is further reduced and at the end of the transient, when the secondary inventory has been completely depleted, core power reaches the equilibrium with the PRHR. This, however, requires a more negative reactivity insertion and hence higher coolant temperatures/lower densities. PRZ overfills and RCS pressure shows a peak following the overfill. However, due to the minor power mismatch between core and PRHR at the time the PRZ overfills (PRZ overfills following a mild transient consequent to the depletion of the secondary inventory at about 380 seconds), the pressure peak is lower than the Condition 2 acceptance criteria (i.e., 2750 psia).

As additional conservatism, it should be noted that PRHR is actuated by DAS on Low SG Water Level – WR. Actuation on a DAS High Hot Leg Temperature (650°F) that occurs at about 12 seconds has not been credited. Please, note that PRHR actuation signals are also available, at different times, by the PMS (i.e., Low Steam Generator Water Level (NR) + Delay, Low Steam Generator Water Level – WR, High Hot Leg Temperature concomitant with Low SG Water Level – WR).

The time sequence of events for this case is reported in Table 11. The behavior of the main plant parameters is illustrated in Figures 53 through 60.

TABLE 11 SEQUENCE OF EVENTS FOR COMPLETE LOSS OF FORCED COOLANT FLOW – CASE 9 LOSS OF AC POWER MAIN FEEDWATER NOT AVAILABLE – TURBINE BYPASS NOT OPERABLE – TURBINE TRIP AT T = 4.0 S	
Time (Seconds)	**Event**
4.	Reactor coolant pumps trip
4.	Turbine trip at the beginning of the accident
4.4	PMS reactor trip setpoint on low speed (no trip generated)
~12.	HL temperature > DAS high HL temperature setpoint (650°F)
Not Credited	Steam dump valves fully open
16.	PRZ safety valves open
22.	Steam generator safety valves open
Not Credited	Reactor tripped on DAS signal above
Not Credited	PRHR actuation on DAS signal above
280.7	DAS low SG water level wide range
Not Credited	Reactor tripped on DAS signal above
292.7	PRHR actuation on DAS signal above
383.	PRZ overfills
398.	Maximum RCS pressure (at RCP outlet) – 2682 psia
> 500	PRZ safety valves close

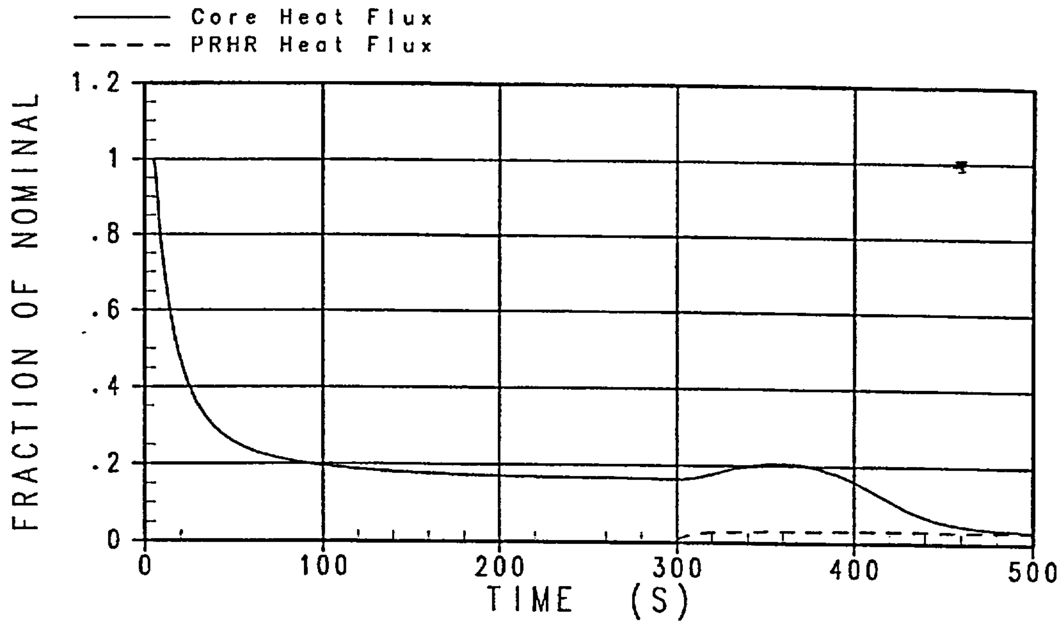


Figure 53 **ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump Core and PRHR Heat Flux (f.o.n) vs. Time**

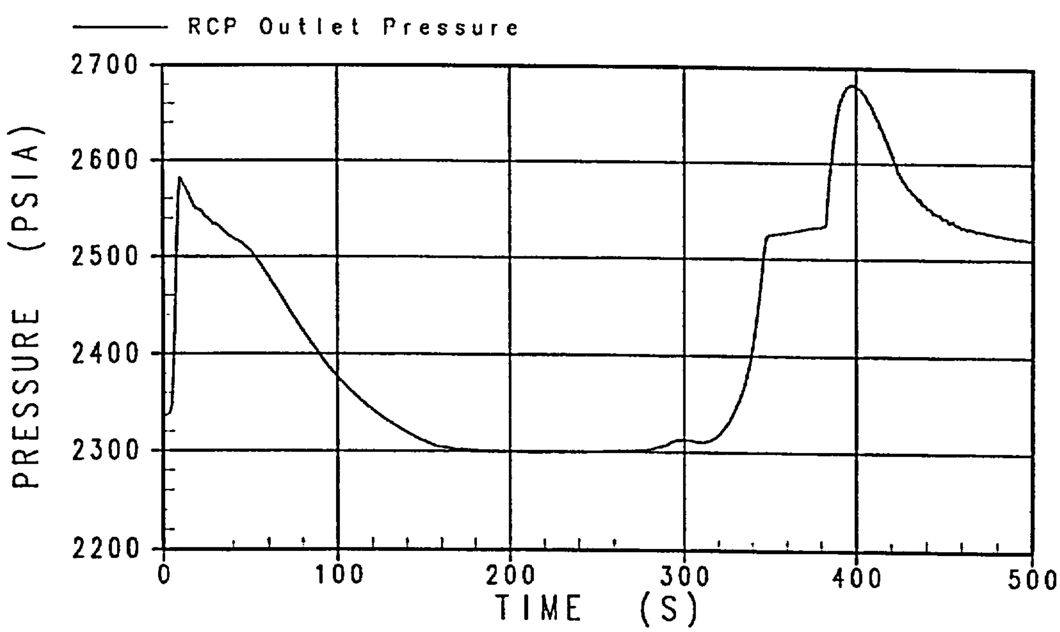


Figure 54 **ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump Reactor Coolant Pump Outlet Pressure (psia) vs. Time**

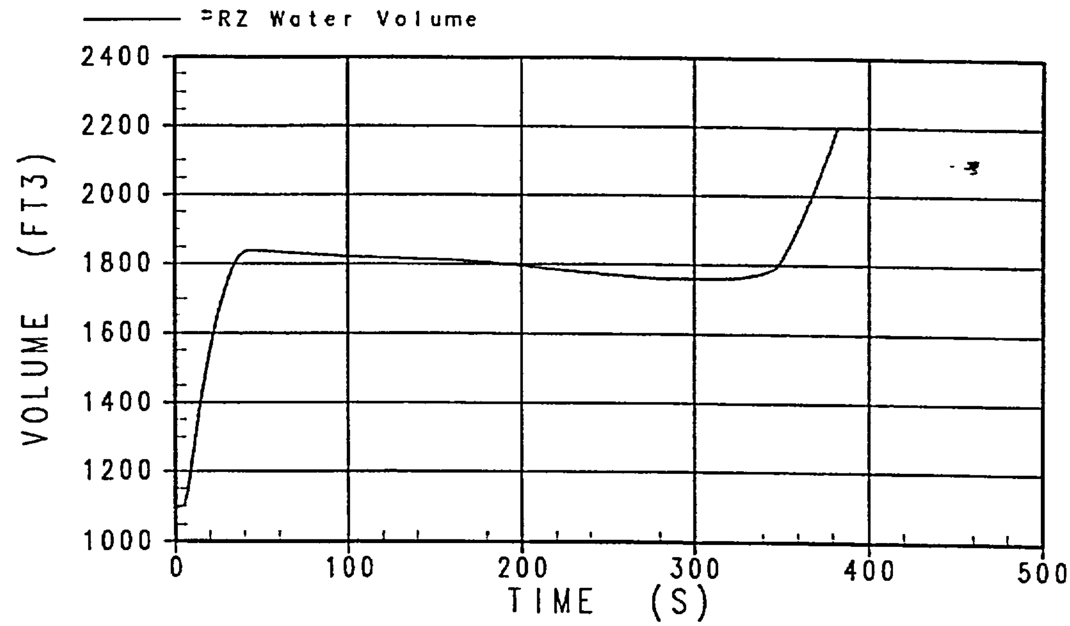


Figure 55 ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump PRZ Water Volume (ft^3) vs. Time

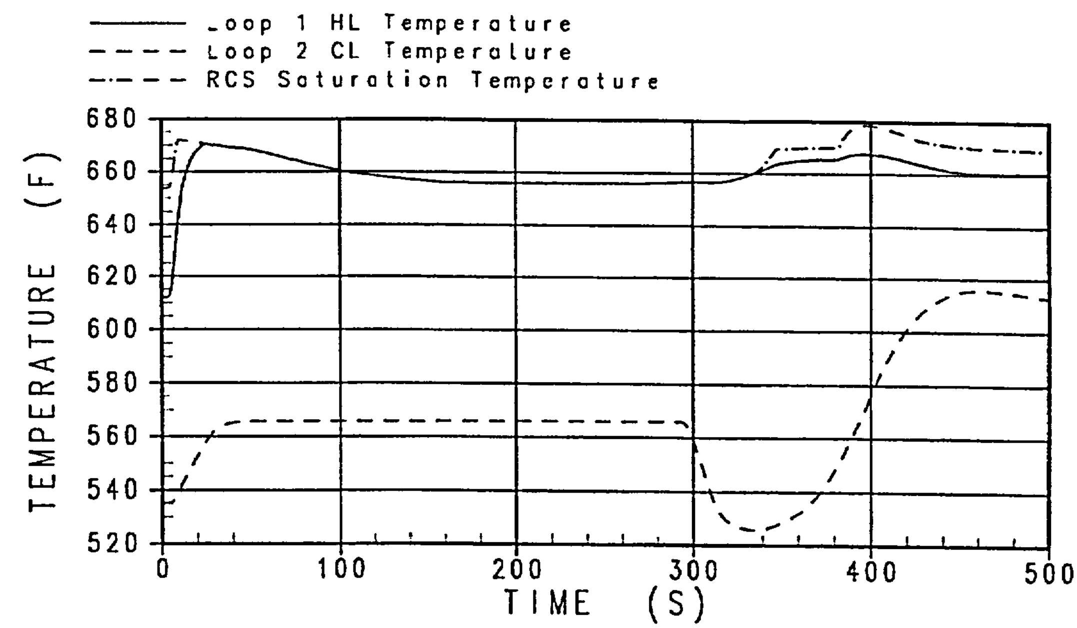


Figure 56 ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump Loop 1 Coolant Temperatures (°F) vs. Time

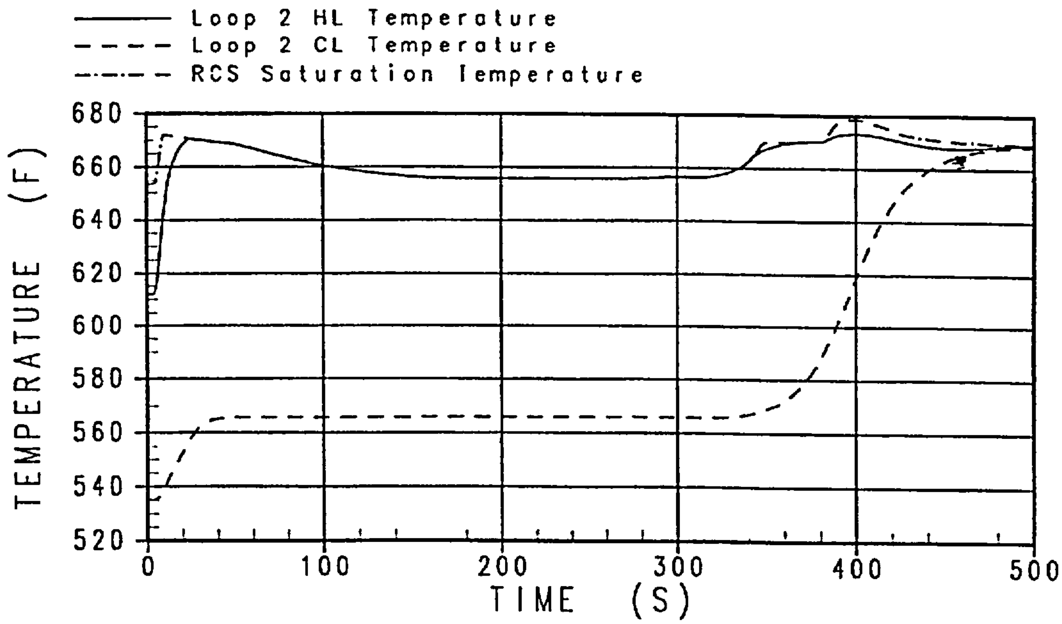


Figure 57 ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump Loop 2 Coolant Temperatures (°F) vs. Time

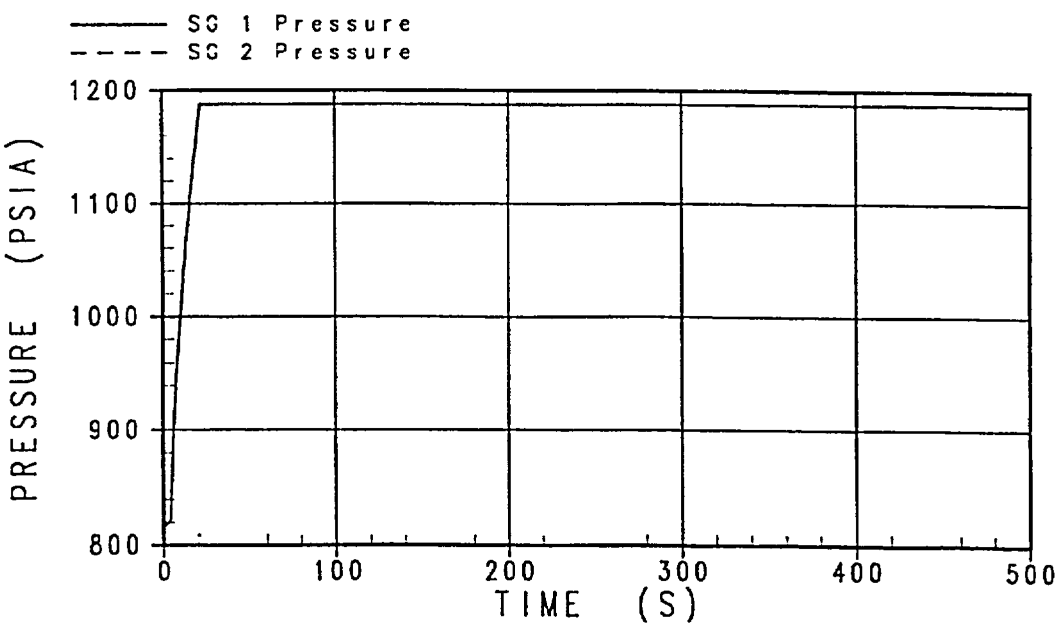


Figure 58 ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump Steam Generators Pressure (psia) vs. Time

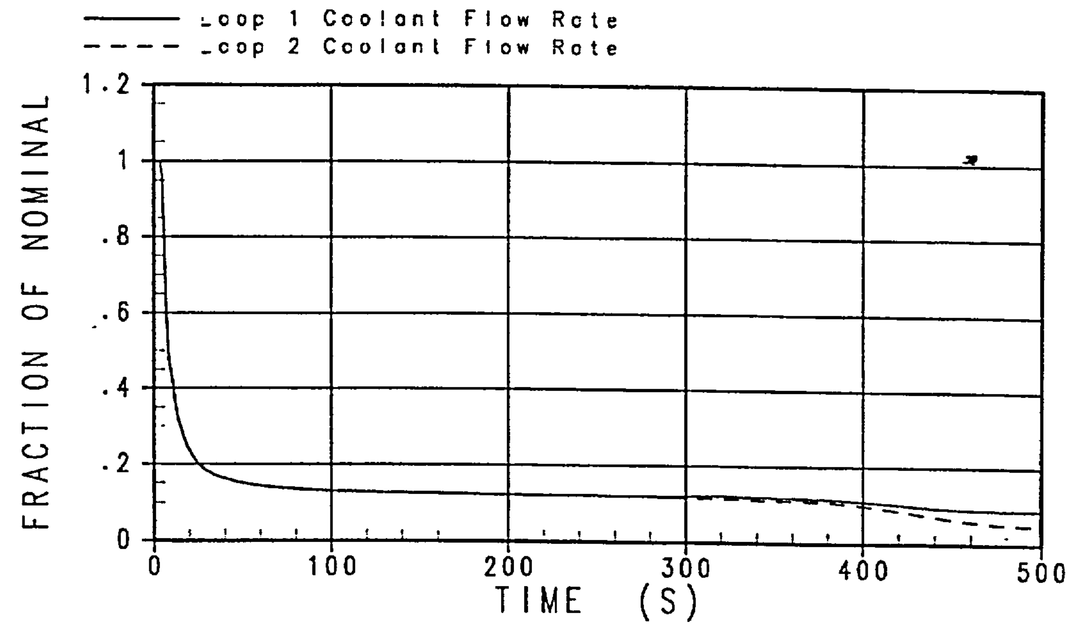


Figure 59 ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump Loops Coolant Flow Rate (f.o.n) vs. Time

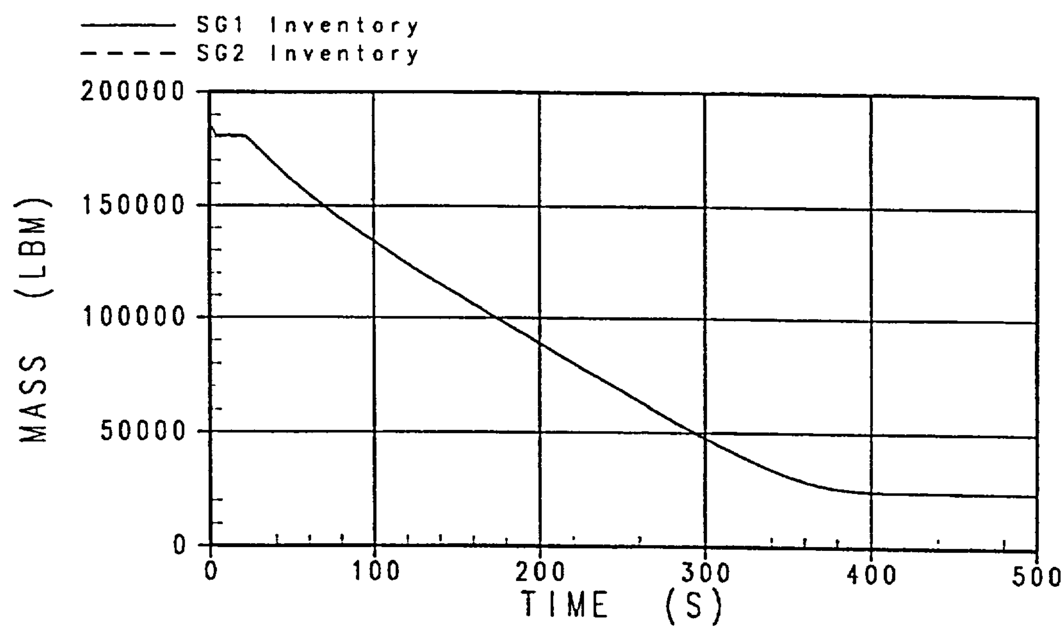


Figure 60 ATWS Complete Loss of Flow – Loss of ac Power – No FW and Steam Dump Steam Generators Inventory (lbm) vs. Time

4.4 Power and Reactivity Distribution Anomalies

4.4.1 Uncontrolled Bank Withdrawal at Power Events

During the initial part of the transient the RCCA bank withdrawal results in an increase in core power and coolant temperature. If PMS is available (assuming the trip failure associated with mechanical failures or trip breakers failures) a reactor trip setpoint would be reached (on a High Nuclear Power, OT/DT, OP/DT or High PRZ Pressure) and a turbine trip would occur and, immediately after the turbine trip, the steam dump, if available, would relieve approximately 40% of the full power steam flow. Feedwater system is assumed to maintain the steam generator water level.

The power level reached during the transient depends on the available reactivity associated to the RCCA bank, moderator temperature coefficient and Doppler power coefficient. Once the rods are fully withdrawn, power will reach an equilibrium level dictated by the balance between reactivity inserted by the rods and reactivity feedback due to the coolant temperature increase and power level increase (Doppler defect).

Several signals may be generated to trip the reactor or actuate CMT both from PMS and DAS. For example, a High HL temperature DAS setpoint may be reached that actuates the PRHR and a PMS high HL temperature with concomitant low SG water level WR may actuate the CMTs to borate the RCS and shut down the reactor even if a mechanical CMF disables the reactor trip function.

In any case the power level will stabilize at the power extracted via steam dump and SG safety valves. This level is expected to be well below the 3200 psia limit since the transient is similar to a reactor trip case with sustained feedwater (see Case 2 section, for which a pressure peak of 2605 psia is evaluated) but at a slightly higher reactor coolant system temperature/power level.

Should the failure to scram be caused by a PMS failure or by a RTB CMF, the DAS trip function would be available. Once the event is diagnosed by the Operator, the Operator can then use DAS to trip the reactor.

The event from zero power would be similar, but turbine would be tripped since the beginning of the accident.

4.4.2 RCCA Misalignment Due to a Dropped RCCA

For the mechanical CMF, there are three different possibilities deriving from this initiating event.

1. If the reactivity associated to the dropped rod/rods is low enough no PMS setpoint is reached and the plant will tend to reach a new equilibrium condition.

2. If the reactivity associated to the dropped rod/rods is large enough that a PMS low reactor coolant pressure is reached, assuming a mechanical CMF, the PMS will actuate the turbine trip. From here on the transient will proceed as a turbine trip but it will be less limiting due to the initial negative reactivity addition.

3. If the reactivity is large enough that the Low T_{cold} Temperature is reached, PRHR and CMT are actuated. Also in this case the behavior is much better than the turbine trip since the core reactivity is much lower than in the turbine trip case and also because the boration by CMT actuation will rapidly shut down the core.

For all the other cases in which no mechanical common mode failure is assumed, the DAS can trip the reactor if a DAS ATWS signal is reached. In all the other cases, in which the plant reaches a new equilibrium condition without reaching a DAS setpoint, the transient will be terminated by the operator that manually shut down the plant via DAS. In this case, reactor coolant system pressure is bounded by the ATWS turbine trip event.

4.4.3 CVS Malfunction that Results in a Decrease in Boron Concentration

At power condition, the event is similar to the bank withdrawal at power. In fact, boron dilution results in a slow addition of positive reactivity.

Considering that boron dilution is manually initiated under strict administrative controls requiring close operator surveillance, and that the dilution event proceeds quite slowly, it can be concluded that there is enough time for the operator to isolate the sources of dilution. It should also be noted that, if the failure to trip is due to mechanical or RTB CMF, the PMS is still able to automatically isolate the possible sources of unborated water.

In lower plant operating mode, reactor trip is not required and boron dilution events are mitigated by terminating the dilution. The Chapter 15 analyses show that even, considering the failure of the PMS, there is enough time for the operator to manually terminate the dilution.

4.5 Increase in RCS Inventory

4.5.1 Inadvertent Operation of Passive Core Cooling System During Power Operation

The inadvertent actuation of one or both the CMTs results in a significant addition of boron in the RCS. At equilibrium condition, the boron concentration increase due to one CMT is about 600 ppm. This increase in boron concentration, even considering a very low boron worth, is sufficient to assure the core shutdown.

Should a PMS signal result in a turbine trip, the combined reactivity effects of temperature increase and boron concentration increase, the overall transient would be much less severe than a turbine trip ATWS event.

4.5.2 CVS Malfunction

CVS malfunctions resulting in addition of unborated water have been assessed in subsection 4.4.3.

Would the CVS malfunction result in addition of water at the same boron concentration as the RCS, the event would be mitigated by the PMS by isolating the CVS itself on High 2 PRZ water level, without any need for a reactor trip. No reactor trip is needed for this scenario. If the chemical and volume control system injects highly borated water, the reactor experiences a negative reactivity excursion that also results in a decrease in core power. The transient can eventually lead to a PMS actuation on Low Steam Pressure or Low T_{cold} Setpoint. These signals actuate the reactor trip, turbine trip, reactor coolant pump trip and CMT actuation, PRHR actuation and steam and feedwater lines isolation. Should the PMS provide the protective functions but a mechanical CMF prevents reactor trip, the resulting transient would be less limiting than the turbine trip event, since it would start from a lower power level and temperature. Moreover, CMT's injection would also result in an additional boration that would shutdown the reactor.

4.6 Decrease in RCS Inventory

4.6.1 Inadvertent Opening of PRZ Safety Valves

The inadvertent opening of a PRZ safety valve causes a depressurization of the RCS. RCS pressure does not increase during this event and RCS pressure boundary is never challenged. Following the RCS depressurization, several PMS signals are actuated and protection systems may be actuated depending on the assumed failure mode. Should a mechanical CMF cause the failure of the reactor trip, PMS would actuate, on a Low PRZ Pressure "S" signal, the turbine trip, PRHR and CMTs. The combined effect of the void formation in the core and the CMT's boration would quickly shutdown the core. Should the assumed failure be the PMS CMF, the void generation in the core would soon insert a large negative reactivity in the core. Eventually, a lower PRZ water level would be reached and DAS would actuate CMT, PRHR and trip the turbine. Of course, operator has the possibility to manually actuate all the required mitigating actions via DAS once the event has been identified.

4.6.2 Failure of Small Line Carring Primary Coolant Outside Containment

The loss of coolant through one of these lines reduces pressurizer water level and actuated the makeup. The size of these lines is such that makeup system can maintain pressurizer level. No reactor trip is anticipated for this event. Subsection 15.6.2 of the AP1000 DCD presents the results of an analysis of this event without reactor trip for 30 minutes until the operator isolates the break.

5 Conclusions

Based on the above discussion, on the basis of the cases analyzed, it can be concluded that, as for the AP600, also for the AP1000 plant the loss of normal feedwater event represents the most limiting ATWS transient.

In particular, it should be noted that three different accident initiators, that cover more than 95% of the ATWS contributors frequency, have been analyzed in detail.

Table 12 summarizes the results of the analyses related to the most important ATWS initiating events.

TABLE 12
ATWS SUMMARY TABLE

Event	Peak Pressure (psia)
Turbine trip without feedwater system operable with turbine bypass system operable	2943.
Turbine trip with feedwater system operable with turbine bypass system operable	2605
Turbine trip without feedwater system operable without turbine bypass system operable (This analysis would also cover the loss of condenser vacuum transients.)	2832
Loss of normal feedwater event with turbine bypass system operable	3135
Loss of normal feedwater event without turbine bypass system operable	2949
Loss of normal feedwater event with turbine bypass system operable, more realistic SG UA	2748
Complete loss of forced coolant flow with MFW system operable with turbine bypass system operable	2530
Complete loss of forced coolant flow with MFW system operable without turbine bypass system operable	2543
Complete loss of forced coolant flow induced by the loss of ac power (mechanical CMF), MFW system not operable, no steam dump system operable, turbine trip at the initiation of the transient	2682

The above results are consistent with the results previously obtained for the AP600 plant reported in the AP600 ATWS Analysis Report.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.037 (Response Revision 1)

Question:

In Tier 2 Information, Section 5.3.2.6.1.2 discusses the least squares adjustment procedure proposed to be applied in the dosimetry evaluation. The estimate of the uncertainties associated with the dosimeter activation measurement involves use of variances and covariances. If a code which has not been approved by the staff is used, then the values of the variances and covariances should be listed and their applicability to AP1000 justified.

Westinghouse Revision 0 Response:

Uncertainty information, including the associated variance and covariance matrix data, are used as inputs to the least squares adjustment code, however, they are not part of the code itself.

To elaborate on this point, the application of the least squares methodology requires the following input:

1) The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2) The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.

3) The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For a given application, the calculated neutron spectrum is obtained from the results of plant specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set. The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross-sections and uncertainties are obtained from the Sandia National Laboratory - Radiation Metrology Laboratory (SNLRML) dosimetry cross-section library. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)". There are no additional data or data libraries built into the least squares adjustment code system. All of the required input is supplied externally at the time of the analysis.

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum are input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties also follows the guidance provided in ASTM Standard E 944.

The following provides a summary of the uncertainties associated with a typical least squares evaluation in a present day LWR dosimetry set:

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards. In all cases, the latest available versions of the applicable standard are used in the dosimetry evaluations.

From these standards, it is noted that the achievable uncertainties in the measured specific activities of each of the sensors comprising typical multiple foil sensor sets are as follows:

Reaction	Precision	Bias
$^{63}Cu(n,\alpha)^{60}Co$	1%	3%
$^{46}Ti(n,p)^{46}Sc$	1%	3%
$^{54}Fe(n,p)^{54}Mn$	1%	3%
$^{58}Ni(n,p)^{58}Co$	1%	3%
$^{238}U(n,f)FP$	1%	5%
$^{237}Np(n,f)FP$	1%	5%
$^{59}Co(n,\gamma)^{60}Co$	1%	5%

These uncertainties include the impacts of counting statistics, sample weighing, detector calibration, source/detector geometry corrections, and product nuclide branching ratios.

In determining reaction rates from the measured specific activities, the following additional uncertainties are incurred:

Reaction	Fission Yield	Product Half-Life	Competing Reactions
$^{63}Cu(n,\alpha)^{60}Co$		0.02%	
$^{46}Ti(n,p)^{46}Sc$		0.2%	
$^{54}Fe(n,p)^{54}Mn$		0.2%	
$^{58}Ni(n,p)^{58}Co$		0.2%	
$^{238}U(n,f)FP$	1%	0.1%	4%
$^{237}Np(n,f)FP$	2%	0.1%	1%
$^{59}Co(n,\gamma)^{60}Co$		0.02%	

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures typically result in the following net uncertainties associated with the sensor reaction rates that are input to the least squares evaluation:

Reaction	Reaction Rate Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	5%
$^{46}Ti(n,p)^{46}Sc$	5%
$^{54}Fe(n,p)^{54}Mn$	5%
$^{58}Ni(n,p)^{58}Co$	5%
$^{238}U(n,f)FP$	10%
$^{237}Np(n,f)FP$	10%
$^{59}Co(n,\gamma)^{60}Co$	5%

The listed uncertainty values are at the 1σ level.

In addition to the use of ASTM National Consensus Standards in the evaluation of sensor reaction rates, these procedures have been periodically tested via round robin counting exercises included as a part of the NRC Sponsored Light Water Reactor Surveillance Dosimetry Improvement Program (LWR-SDIP) as well as by evaluation of fluence counting standards provided by the National Institute of Science and Technology (NIST). A summary of some of these counting validations is as follows:

1980 Round robin counting of foil sets irradiated at the Thermal Shield Back (TSB) and Pressure Vessel Face (PVF) positions of the PCA simulator.

1981 Round robin counting of additional foil sets included in the first metallurgical simulated surveillance capsule, also irradiated in the PCA benchmark mockup.

These two counting exercises involved direct comparisons with measurements obtained by The Hanford Engineering Development Laboratory (HEDL). At the time of these irradiations, HEDL was a prime contractor providing measurement services for the PCA benchmark and was cross-calibrated with NIST and the MOL Laboratory in Belgium.

1985 Counting and evaluation of $^{46}Ti(n,p)^{46}Sc$, $^{54}Fe(n,p)^{54}Mn$, and $^{58}Ni(n,p)^{58}Co$ certified fluence standards supplied by NIST.

Comparisons with fluence standards involve the determination not only of the reaction rate of each foil, but also of the spectrum averaged cross-section in the NIST ^{235}U irradiation facility. Thus, the comparisons with the certified fluence standards test both the measurement process and the energy dependent reaction cross-sections used in the evaluation.

1992 Counting of NIST foils irradiated in an ex-vessel dosimetry experiment at the Trojan power reactor.

This exercise involved duplicate counting of a subset of irradiated foils by both Westinghouse and NIST to assure adequate cross-calibration of the laboratories so that data could be confidently mixed in the overall fluence evaluations performed by NIST and ORNL.

Results of these counting intercomparisons are summarized as follows:

	[Westinghouse]/[HEDL]		[Westinghouse]/[HEDL]		
Reaction	1980	1981	1985	1992	Average
$^{63}Cu(n,\alpha)^{60}Co$	1.041	1.018		0.969	1.009
$^{46}Ti(n,p)^{46}Sc$	1.036		1.012	1.030	1.026
$^{54}Fe(n,p)^{54}Mn$	1.006	1.008	1.011	1.056	1.020
$^{58}Ni(n,p)^{58}Co$	1.006	0.990	1.028	1.029	1.013
$^{238}U(n,f)FP$	1.014	1.014			1.014
$^{237}Np(n,f)FP$	1.006	1.017			1.012
$^{59}Co(n,\gamma)^{60}Co$	1.017	1.017			1.017

These comparisons demonstrate that the procedures used by Westinghouse in the determination of reaction rates have produced accurate and stable results over an extended period. The cross-comparisons with HEDL and NIST support the reaction rate uncertainties used by Westinghouse in performing LWR fluence evaluations.

Dosimetry Cross-Section Uncertainties

The reaction rate cross-sections used in the neutron fluence evaluations are taken from the SNLRML library. This data library provides reaction cross-sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross-sections and uncertainties are provided in a fine multigroup structure for use in least squares adjustment applications. These cross-sections were compiled from the most recent cross-section evaluations and they have been tested with respect to their accuracy and consistency for least squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources. Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors may be obtained in the RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," dated July 1994.

For sensors of interest to LWR dosimetry applications, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
$^{63}Cu(n,\alpha)^{60}Co$	4.08-4.16%
$^{46}Ti(n,p)^{46}Sc$	4.51-4.87%
$^{54}Fe(n,p)^{54}Mn$	3.05-3.11%
$^{58}Ni(n,p)^{58}Co$	4.49-4.56%
$^{238}U(n,f)FP$	0.54-0.64%
$^{237}Np(n,f)FP$	10.32-10.97%
$^{59}Co(n,\gamma)^{60}Co$	0.79-3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with typical sensor sets used in LWR irradiations.

Calculated Neutron Spectrum

The neutron spectrum input to the least squares adjustment procedure is obtained directly from the results of plant specific transport calculations for each sensor location. The spectrum at each location is input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data are treated equally with the measurements.

While the uncertainties associated with the reaction rates are obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties are supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum is constructed from the following relationship:

$$M_{g'g} = R_n^2 + R_g * R_{g'} * P_{g'g}$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties $R_{g'}$ and R_g specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{g'g} = [1-\theta]*\delta_{g'g} + \theta * e^{-H}$$

where

$$H = \frac{(g-g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g' and is 0.0 otherwise.

A typical set of parameters defining the input uncertainties for the calculated spectrum is as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	29%
(E < 0.68 eV)	52%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

For this typical spectrum correlation matrix, the combination of the normalization uncertainty with the group dependent random uncertainties results in the following overall uncertainty in the calculated spectrum for the three energy ranges defined above.

(E > 0.0055 MeV)	21%
(0.68 eV < E < 0.0055 MeV)	56%
(E < 0.68 eV)	74%

These uncertainty assignments are consistent with an industry consensus uncertainty of 15-20% (1σ) for the integral flux above 1.0 MeV and provide for a reasonable increase in the uncertainty for neutrons in the intermediate and thermal energy ranges.

As a final remark, it should be noted that the counting of the dosimetry sensor sets, dosimetry cross-section libraries, and transport calculations will be based on the latest guidelines and standards available when the AP1000 plant has been built and the dosimetry is evaluated.

NRC Additional Comment:

The W response did not address the values of the covariances in the adjustment process. However, in March 2001 the staff issued Regulatory Guide (RG) 1.190 which addresses fluence methodology.

Please verify that future vessel fluence calculations will adhere to the guidance of RG 1.190.

Westinghouse Additional Response:

AP1000 DCD subsection 5.3.2.6.2.2 references the AP1000 compliance to RG 1.190 and is shown below:

5.3.2.6.2.2 Plant-Specific Calculations

The location, selection, and evaluation of neutron dosimetry and the associated radiometric monitors, as well as fast (E > 1.0 MeV) neutron fluence assessments of the AP1000 reactor pressure vessel, are conducted in accordance with the guidelines that are specified in Regulatory Guide 1.190.

Table 1.9-1 Regulatory Guide / DCD Section Cross-References will be updates as shown on the next page.

Design Control Document (DCD) Revision:

Table 1.9-1 (Sheet 15 of 15)

REGULATORY GUIDE/DCD SECTION CROSS-REFERENCES

Division 1 Regulatory Guide	DCD Chapter, Section or Subsection
1.183 Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors (Rev. 0, July 2000)	2.3 4.2 6.5.1 15.4 15.6.3 15.7
1.184 Decommissioning of Nuclear Power Reactors (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.185 Standard Format and Content for Post-shutdown Decommissioning Activities Report (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.186 Guidance and Examples of Identifying 10 CFR 50.2 Design Bases (Rev. 0, December 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.187 Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments (Rev. 0, November 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.189 Fire Protection for Operating Nuclear Power Plants (Rev. 0, April 2001)	This regulatory guide is not applicable to AP1000 design certification.
1.190 Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Rev. 0, March 2001)	~~This regulatory guide is not applicable to AP1000 design certification.~~5.3.2.6.2.2

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.077 (Response Revision 1)

Question:

The guidance for a LOOP assumed for the FLB analysis is provided in SRP 15.2.8. Specifically, item b of the acceptance criteria states that "Assumptions as to whether offsite power is lost and the time of loss should be make conservatively. Offsite power may be lost simultaneously with the concurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost."

Discuss the determination of the time of an LOOP assumed for the limiting FLB analysis and address the compliance with the SRP guidance related to the time of an LOOP.

NRC Additional Comment:

The last sentence of the fourth paragraph on page 440.077-1 states that "[t]his means that rod motion follows the time at which the double ended break open by two seconds,..."

Identify the contributors to the trip delay time of two seconds following the low SG level NR trip signal.

Westinghouse Additional Response:

The delay time of two seconds referred to in the original response refers to the standard low steam generator water level trip delay used in Westinghouse protection system designs. The contributors to this delay time includes the sensor response time, signal processing time, trip-breaker opening time, and the RCCA gripper opening time.

Westinghouse Original Response:

For the AP1000 plant, the main effects due to the loss of offsite power are related to the loss of forced reactor coolant flow and coincident opening of the trip breakers resulting in the plant shutdown. The passive safety systems do not rely on safety-related ac power, and therefore their operation is not affected by the loss of offsite power. The initiation of the passive systems requires the alignment of safety-related valves energized by safety Class 1E dc power system. Actuation times are essentially the same both with and without ac power available, except for the PRHR actuation that, if ac power is available, is delayed by about 45 seconds with respect to the time at which the low SG water level (narrow range) setpoint is reached. However, for this analysis the PRHR is started on a low SG water level (wide range) signal or following the low steamline pressure "S" signal that follows immediately after.

For the AP1000 feedline break analysis, the loss of ac power is conservatively assumed to occur at the time of reactor trip (rod motion). According to the accident methodology followed to simulate the accident, the full double ended break opens in the feedline (control interaction assumption, refer to answer to RAI 440.076) at the time at which the low SG level NR reactor trip setpoint is reached. This means that rod motion follows the time at which the double ended break open by two seconds, and, hence two full power seconds are input to the coolant, during the blowdown phase, before the rods start to fall in the core.

Should the loss of ac power occur at the time of the break, control rods will be inserted sooner (on the loss of ac power) and hence a lower RCS heat up and pressurization would be experienced by the plant.

The assumption of the loss of ac power at the time of reactor trip (rod motion) is conservative since it results in the maximum possible power input to the RCS and also in an immediate drop of the RCS forced flow and hence of the heat transfer between primary and secondary systems.

Should a loss of ac power occur any later in the transient, the energy stored in the RCS will be lower than for the case analyzed due to the better heat transfer with the steam generator secondary side.

Finally, should the offsite power be available, an automatic reactor coolant pump trip signal will occur on a low steam line pressure "S" signal. This signal occurs quite soon for a double ended feedline break event (about 15 seconds from the reactor trip) but could occur with a significant delay for small feedline breaks. In the later case, PRHR performance will be significantly higher due to the forced flow through the loop and overall RCS conditions will be less limiting. In this case the assumption of loss of ac power at the time of the trip is conservative.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.025 (Response Revision 1)

Question:

(Section 16.1, Bases for TS 3.4.17 Required Action C.2) The Completion Time of Required Action C.2 seems to allow for cooling down to the temperature (assumed to be 275°F) at which the residual heat removal system (RNS) may be placed in operation after entering Mode 4. The Bases should explain this more explicitly because the usual time to reach Mode 4 (420°F) on a shutdown action is 12 hours.

Westinghouse Response:

Action time for MODE 4 entry will be changed, as indicated below, to 12 hours.

In addition, the wording in the Bases for TS 3.4.17 Action B.1 was in error and has been corrected, as indicated below.

DCD Revision 3 incorporates the changes identified in the original response to this RAI. DCD Revision 3 will be updated to address this change.

Design Control Document (DCD) Revision:

From DCD Chapter 16, TS 3.4.17, pg. 3.4-32:

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, and 3,
MODE 4 with the RCS not being cooled by the RNS.

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 cooling provided by the RNS.	6 hours ~~24~~**12** hours

From DCD Chapter 16, TS B 3.4.17, pg. B 3.4-72:

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 **both**~~two~~ flow paths are determined to be inoperable, the RVHV is degraded such that the ~~remaining~~ system ~~capacity may not be~~**is not**~~adequate~~ **available** for some DBA non-LOCA analysis **for which may be required**. A Completion Time of 6 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.

From DCD Chapter 16, TS B 3.4.17, pg. B 3.4-73:

ACTIONS (continued) 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 6 hours and to MODE 4 within ~~24~~**12** 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.

PRA Revision:

None

NRC Additional Comments:

Please clarify the editorial error in the revised first sentence in Action B.1 Bases discussion.

Westinghouse Additional Response:

The word "it" was added to the sentence near the end and the word analysis was pluralized.

Design Control Document (DCD) Revision:

Bases 3.4.16 (renumbered as 3.4.16 in Revision 3, previously 3.4.17):

If both flow paths are determined to be inoperable, the RVHV is degraded such that the system is not available for some DBA non-LOCA **analyses** ~~analysis~~ for which **it** may be required.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.027 (Response Revision 1)

Question:

(Section 16.1, Bases for TS 3.4.12)

A. The Bases Background discussion does not state the size of the Stage 4 automatic depressurization system (ADS) flow path squib and motor operated valves. The staff suggests including this information for completeness.

B. The last sentence of the Bases Background discussion states "The [probabilistic risk assessment] PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of two (or more) flow paths." Is this referring only to the ADS Stage 4 flow paths? What does the "(or more)" mean - that all ADS flow paths can be inoperable without compromising core cooling? Please clarify.

Westinghouse Response:

A. The Bases Background discussions do not include numerical sizes of valves or other components. For example, no sizes are included in the LTOP System (3.4.15), RV Head Vent (3.4.17, Accumulators (3.5.1), CMTs (3.5.2), PRHR HX (3.5.4), and IRWST (3.5.6). No change is proposed.

B. The PRA success criteria for the ADS following a LOCA or a non-LOCA with failure of all other core cooling features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1,2,3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation

Design Control Document (DCD) Revision:

The following revision to DCD Technical Specifications B 3.4.12 was included in DCD, Rev.3.

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of two (or more) flow paths. **The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1,2,3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.**

PRA Revision:

None

NRC Additional Comments:

Please clarify the last paragraph of TS 3.4.11, Rev 3 Bases Background discussion regarding multiple ADS flow path failures. (This was previously TS 3.4.12 before Rev. 3.)

Westinghouse Additional Response:

The last paragraph will be revised as shown below to clarify the allowable flow path failures for the ADS PRA success criteria.

DCD Revision 3 incorporates the changes identified in the original response to this RAI. DCD Revision 3 will be updated to address this change.

Design Control Document (DCD) Revision:

Last paragraph of TS 3.4.11, Bases Background

The number and capacity of the ADS flow paths are selected so that adequate safety injection is provided from the accumulators, IRWST and containment recirculation for the limiting DBA loss of coolant accident (LOCA). For small break LOCAs the limiting single failure is the loss of one fourth stage flow path (Ref. 2). The PRA (Ref. 3) shows that adequate core cooling can be provided with the failure of **up to [seven] (all ADS stage 1 to 3 and [one] ADS stage 4)** ~~two (or more)~~ flow paths. The ADS PRA success criteria following a LOCA or non-LOCA with failure of other decay heat removal features is for 3 of 4 ADS stage 4 valves to open. All of the ADS stage 1, 2, 3 valves can fail to open. This ADS capacity is sufficient to support PXS gravity injection and containment recirculation operation.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.028 (Response Revision 1)

Question:

(Section 16.1, TS 3.4.12 Action A and TS 3.4.13 Action A) Condition A lists three possible conditions. Does each condition get a separate Completion Time? How does this action requirement work? None of the examples in Section 1.3 address this format. Please describe a specific example for each Specification. There may be a clearer way to render these action statement conditions.

Westinghouse Response:

TS 3.4.12 Action A lists three separate Conditions that each have their own 72 hour Completion Time to return the affected flowpath(s) to operable status. TS 3.4.13 Action A lists two separate Conditions that each have their own 72 hour Completion Time to return the affected flowpath(s) to operable status.

The application of the Completion Times as described above for the multiple conditions within a Condition Statement in TS 3.4-12 and TS 3.4-13 (as well as the identified STS TSs above) is consistent with the NRC discussion for the application of Completion Times for this type of Condition Statement as described by the NRC reviewer in Item A of RAI 630.039.

The format of these Actions are consistent with the format of the following STS, Rev. 2 Action statements: TS 3.3.6 Action B; TS 3.4.12 Action G; TS 3.4.16 Action C; TS 3.5.4 Action A.

Since the format is consistent with formats used in the STS, Rev. 2, no change to the AP1000 TS or addition to the examples in Section 1.3 is deemed necessary, in order to prevent deviation from the STS Rev. 2 where there is no technical difference due to the AP1000 plant design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None