ENCLOSURE

ATTACHMENT 9a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3933NP REPORT, REVISION 0

MONTICELLO ATWS-I EVALUATION FOR ATRIUM 11 FUEL

JUNE 2021

(38 pages follow)

Monticello ATWS-I Evaluation for ATRIUM 11 Fuel

ANP-3933NP
Revision 0

June 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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Nomenclature

Acronym	Definition
2RPT	Two Recirculation Pump Trip
ATWS	Anticipated Transient Without Scram
ATWS-I	Anticipated Transient Without Scram With Instability
BOC	Beginning of Cycle
CPR	Critical Power Ratio
CPROM	Critical Power Reduced Order Model
ECPR	Experimental Critical Power Ratio
EFW	Extended Flow Window
EOC	End of Cycle
EOI	Emergency Operating Instruction
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
L&C	Limitation and Condition
LHGR	Linear Heat Generation Rate
PCT	Peak Clad Temperature
PHE	Peak Hot Excess
TR	Topical Report
TTWB	Turbine Trip With Bypass

1.0 INTRODUCTION

This document presents the results of calculated BWR instability transients that are not terminated by scram and thus power and flow oscillations are allowed to grow to large amplitudes. This class of transients is referred to as Anticipated Transients Without Scram with Instability (ATWS-I). The code used for simulating this event is RAMONA5-FA, which was developed and approved in Reference 1.

The scope of this analysis covers the Monticello EPU/EFW operating domain using an equilibrium ATRIUM 11 core.

2.0 RAMONA5-FA ATWS-I METHODOLOGY

The Framatome generic methodology for the evaluation of Anticipated Transients Without Scram with Instability (ATWS-I) is presented in Reference 1. This report documents a RAMONA5-FA based method for evaluating the fuel specific portion of the event. This method is intended to cover the initial ATWS-I event through the time that operator actions suppress core oscillations. This method is not intended for use in evaluating containment effects or over pressurization during the event.

A key improvement of the RAMONA5-FA code is the addition of **[**

] This model is described in Section 5.5.5 of Reference 1, and the benchmarking of the ATRIUM 10XM fuel type with this new correlation is given in Appendix A of Reference 1. This correlation was extended to ATRIUM 11 using the procedure provided in Section A.4 of Reference 1, with the results of the benchmarking given in Section 4.0 of this report.

Analyses performed here are in compliance with the calculation procedure defined in Section 8.0 of Reference 1. Acceptance requires the limiting peak clad temperature results remain below 2200 °F (1204 °C). Section 6.0 of this report also provides additional descriptions of how each Limitation and Condition for the topical report was implemented.

3.0 SCENARIO IDENTIFICATION

3.1 *Turbine Trip With Bypass*

The first event, the TTWB, is initiated by a turbine trip. The recirculation pumps are tripped by an ATWS high pressure pump trip. Core flow drops rapidly to natural circulation which drives core power lower. As the core settles at natural circulation, the feedwater temperature begins to decrease due to the loss of extraction steam. The feedwater temperature decrease causes core power to rise. If no action is taken, then core power will rise to the point that it crosses the instability boundary. Without the ability to scram, and if mitigating operator action is sufficiently delayed, oscillations will begin to grow and will reach large magnitudes, accompanied by reverse flow at the inlet of the hot channels. Once the oscillations reach sufficient magnitude, cyclical dryout and rewetting of the cladding surface will begin. If the oscillations continue to grow, they can reach sufficient magnitude that rewetting is no longer possible and large excursions of clad temperature that could potentially challenge acceptance criteria will occur.

In order to prevent large amplitude oscillations plant Emergency Operating Instructions (EOI) or Emergency Operating Procedures (EOP) instruct the operators to reduce water level upon recognition of an ATWS scenario. This reduction in water level has two primary effects. The first is to reduce core flow by reducing the density head on the downcomer side of the loop. The resulting reduction in core flow also serves to reduce core power. At this point in the transient scenario, the cold feedwater is spraying through a steam environment and falling a significant distance until it reaches the water level. Passing through the steam environment heats the feedwater to the point that it approaches saturation. This significantly reduces the core inlet subcooling, which not only directly adds a stabilizing force to the system, but also reduces core power further stabilizing the system. The suppression of oscillations and reduction in power terminate any temperature excursion in the fuel and ends the fuel-specific portion of the ATWS-I event.

For this event, the rate of feedwater temperature decrease and the time of operator action are of primary importance. As such, the values used for these two input assumptions are the same as the current Monticello ATWS-I analysis of record. The feedwater temperature decrease is assumed to start 10 seconds after the valve closure. Temperature is then assumed to decrease with a 30 second time constant until the final temperature is reached. The time critical operator action for water level reduction is assumed to begin at 90 seconds, consistent with the current analysis of record for Monticello.

3.2 *Two Recirculation Pump Trip*

The two recirculation pump trip event (2RPT) evolves similarly to the TTWB event with two exceptions, the event initiation and the feedwater temperature excursion. Since the 2RPT event does not involve a turbine isolation, the turbine remains online and extraction steam to the feedwater heaters is maintained so the feedwater temperature remains significantly higher than the TTWB event. Because of this, the power excursion in the 2RPT event is mild which results in a less severe event for the same operator intervention times. One difference between the two transient scenarios is that the TTWB event should generate an automatic scram very early in the event leading to earlier identification of the ATWS scenario by the operators. There may not be a scram signal at the initiation of the 2RPT which means the event may be allowed to progress further due to a delayed identification of ATWS by the operator. As such, the 2RPT may become limiting if operator action in the TTWB event occurs fast enough to suppress oscillations prior to dryout. For the evaluations presented in this report, a stable limit cycle without sustained dryout is demonstrated prior to the time of operator action assumed in the current analysis of record for Monticello.

4.0 ATRIUM 11 CPROM CORRELATION

A new dryout correlation was presented in Appendix A of Reference 1. This correlation, named Critical Power Reduced Order Model (CPROM), has been developed based on Framatome correlation development guidelines similar to dryout licensing correlations such as ACE. The CPROM correlation range of applicability is wide **[ ]** making it well-suited to fitting into transient models of post-dryout that include cyclical dryout and rewetting with possible failure to rewet. CPROM is an integral part of the RAMONA5-FA transient model described in Section 5.5.5 of Reference 1.

4.1 *CPROM Correlation for ATRIUM 11*

[ ]

Table 4-1 [ ]

[

]

[

]

Figure 4-1 Calculated versus measured critical power, [ ]

Figure 4-2 [ ]

Figure 4-3 [ ]

Figure 4-4 [ ]

Figure 4-5 [ ]

Figure 4-6 [ ]

Table 4-2 Statistics [ ]

Table 4-3 Statistics [ ]

Table 4-4 Statistics [ ]

Table 4-5 Statistics [ ]

[ ]

Figure 4-7 Calculated versus measured critical power, [ ]

Figure 4-8 [ ]

Figure 4-9 [ ]

Figure 4-10 [ ]

Figure 4-11 [ ]

Figure 4-12 [ ]

Table 4-6 Statistics [ ]

Table 4-7 Statistics [ ]

Table 4-8 Statistics [ ]

Table 4-9 Statistics [ ]

The figure below shows comparison of the calculated and measured critical power for the combined data set. The mean critical power ratio is **[ ]** and the standard deviation of the calculated versus measured critical power for the entire database is **[ ]** and the number of data points is **[ ]**.

Figure 4-13 Calculated versus measured critical power

4.2 *ATRIUM 11 CPROM Bounds of Applicability*

The bounds of applicability are determined by the data available to benchmark the correlation. For ATRIUM 11, the **[**

]

5.0 DISCUSSION OF RESULTS

5.1 *Turbine Trip With Bypass*

For this evaluation, **[ ]** are analyzed consistent with the procedure identified in Section 8.0 of Reference 1. An equilibrium ATRIUM 11 core was utilized. The events were initiated from rated power at the EFW boundary. As discussed in Section 6.0, in order to ensure that these results will be bounding of future cycles, **[**

]

Table 5-1 gives the TTWB results from the Monticello EFW ATRIUM 11 analysis including the results of the sensitivity analyses. For each case (assuming operator intervention to lower the water level after **[ ]**), the peak clad temperature (PCT) is provided. Relevant system and limiting channel plots for the limiting case can be found in Figure 5-1 through Figure 5.6.

5.2 *Two Recirculation Pump Trip*

Table 5-2 gives the 2RPT results from the Monticello EFW ATRIUM 11 analyses. The base statepoints analyzed were chosen consistent with those analyzed for the TTWB event. For each case it was assumed that no operator intervention to lower the water level occurred. The results show **[**

]

Table 5-1 Monticello EFW ATRIUM 11 TTWB ATWS-I Calculations

Table 5-2 Monticello EFW ATRIUM 11 2RPT ATWS-I Calculations

Figure 5-1 ATRIUM 11 [ ] TTWB Core Power Response

Figure 5-2 ATRIUM 11 [ ] TTWB Core Inlet Subcooling Response

Figure 5-3 ATRIUM 11 [ ] TTWB Core Inlet Flow Response

Figure 5-4 ATRIUM 11 [ ] TTWB Water Level Response

Figure 5-5 ATRIUM 11 [ ] TTWB Limiting Channel Inlet Flow Results

Figure 5-6 ATRIUM 11 [ ] TTWB Limiting Channel PCT Response

6.0 COMPLIANCE WITH LIMITATIONS AND CONDITIONS

The SE for the Reference 1 topical report lists seven limitations and conditions that applications of the methodology must satisfy. The following section discusses how the calculations in this package comply with those limitations and conditions.

Limitation and Condition 1

> *The gap conductance sensitivity shall be repeated or otherwise justified for transitions to new fuel designs.*

A gap conductance sensitivity study was performed for the limiting statepoint, **[** **]**. The gap conductance was varied by **[** **]**. This sensitivity showed that the PCT for the limiting channel can vary between **[** **]**. This shows the overall impact is **[** **]** within the available margin for ATRIUM 11 fuel at Monticello.

Limitation and Condition 2

> *If the acceptance criteria for the first paragraph in Step 3 of Section 8.0 of the TR are met, additional justification must still be provided to demonstrate adequate margin in operator action timing for variations in neutron kinetics response from specific core designs. This justification may be provided by following Steps 3.a through 3.c, as amended by the response to RAI 15, or providing an alternative justification on a plant-specific basis.*

In order to ensure that this calculation will bound future core designs, **[**

]

[

]

ensure that it will remain bounding of future cycles.

Limitation and Condition 3

> *Plant-specific evaluations that are intended to be bounding of all core designs must be confirmed to provide reasonable assurance that neutron kinetics characteristics such as possible differences in dominant oscillation modes or the potential for multiple oscillation modes to be active simultaneously are bounded by the analysis of record.*

As discussed in the compliance to L&C 2, **[**

] providing for a bounding calculation regardless of oscillation mode.

Limitation and Condition 4

> *Due to the unique neutron kinetics characteristics associated with transition cycles, all transition cycles must be dispositioned in a manner consistent with Limitations and Conditions #2 and #3.*

As discussed in the compliance to L&C 2, **[**

]. These effects introduce significant conservatism and will bound the effects associated with transition cycles.

Limitation and Condition 5

> *The ATWS-I analysis must be performed for both the TTWB and 2RPT events during the initial implementation of this methodology, to confirm which event is limiting. Subsequent evaluations may only consider the event determined to be limiting, except which changes are made to the plant design or operation that may affect stability behavior during ATWS, such as: turbine bypass capability, fraction of steam-driven feedwater pumps, and changes expected to significantly increase core inlet subcooling during ATWS events.*

Analyses were performed for both TTWB and 2RPT events and the results are reported in Section 2. These results demonstrate that **[ ]**.

Limitation and Condition 6

> *The steam line and valve modeling options shall be confirmed to accurately capture the expected plant-specific system performance during ATWS-I events.*

The steam line and valve models were built based on plant geometry and setpoints as supplied by Xcel Energy. This allows for an accurate representation of the evolution of the event. A review of the system transient plots show reasonable behavior during the event.

Limitation and Condition 7

> *Plant-specific applications must justify that the selected settings and modeling options are appropriate, including core and vessel nodalization, time step control parameters, and noise parameters. In particular, the modeling should be reasonably consistent with both the characteristics of the plant in question and the validation basis for the RAMONA5-FA ATWS-I methodology as discussed in this SE.*

All nodalization is consistent with the nodalization of the benchmarks and sample problem from the original Reference 1 topical report. In addition, the time step control and noise parameters were identical to those used in the topical report. A nodalization study was performed to demonstrate that the selected nodalization is reasonably representative of the plant.

7.0 CONCLUSIONS

The results of the Monticello EPU/EFW ATRIUM 11 ATWS-I analyses have been presented. Both the TTWB and the 2RPT event have been simulated. The results of these analyses demonstrate that the limiting peak clad temperatures remain below the 2200 °F (1204 °C) temperature criteria.

8.0 REFERENCES

1. ANP-10346P-A Revision 0, *ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA*, Framatome Inc., October 2019.

ENCLOSURE

ATTACHMENT 9b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3933P REPORT, REVISION 0

MONTICELLO ATWS-I EVALUATION FOR ATRIUM 11 FUEL

JUNE 2021

(3 pages follow)

AFFIDAVIT

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3933P Revision 0, "Monticello ATWS-I Evaluation for ATRIUM 11 Fuel," dated June 2021 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on: June 7, 2021

Alan B. Meginnis

ENCLOSURE

ATTACHMENT 10a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3929NP REPORT, REVISION 0

MONTICELLO ATRIUM 11 CONTROL ROD DROP ACCIDENT ANALYSES WITH THE AURORA-B CRDA METHODOLOGY

JUNE 2021

(41 pages follow)

Monticello ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology

ANP-3929NP
Revision 0

June 2021

TRADEMARKS

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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Nomenclature

Acronym	Definition
ASME	American Society of Mechanical Engineers
AST	alternate source term
BOC	beginning of cycle
BPWS	banked position withdrawal sequence
BWR	boiling water reactor
CFR	code of federal regulations
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
CWSR	cold work stress relief aka SRA
CZP	cold zero power
EFP	end of full power
EOC	end of cycle
GDC	general design criteria
GMUL	gas multiplier
LBRF	licensing basis release fraction
LTR	licensing topical report
LWR	light water reactor
MOC	middle of cycle corresponding to peak reactivity
NRC	Nuclear Regulatory Commission, U.S.
PCMI	pellet clad mechanical interaction
RIA	reactivity insertion accident
RPS	reactor protection system
SE	safety evaluation
SER	safety evaluation report
SRA	stress relief annealed
SRP	standard review plan
SSRF	steady state release fraction
TFGR	transient fission gas release
USAR	updated safety analysis report
ΔH	transient change in enthalpy
ΔH_p	prompt enthalpy rise
ΔH_tot	total enthalpy rise

1.0 INTRODUCTION

The Framatome AURORA-B CRDA methodology has been used to evaluate the Monticello ATRIUM 11 equilibrium fuel cycle (Reference 1). The methodology includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic (T-H) and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods to show compliance with regulatory guidance criteria as presented in Regulatory Guide 1.236 (Reference 3). The report summarizes the application of the AURORA-B CRDA methodology (Reference 4) on the Monticello ATRIUM 11 equilibrium cycle.

The control rod drop calculations were performed with the AURORA-B CRDA methodology. All startup sequences were evaluated and no fuel rod failures were identified through middle of cycle. Evaluations of the drops at the end of full power and end of cycle identified potential fuel rod failures in one startup sequence.

2.0 REGULATORY BASIS

The current regulatory basis for the acceptance criteria for the Monticello licensing is fuel failure at 170 cal/g and violent expulsion of fuel at 280 cal/g consistent with Reference 5 (SRP 15.4.9, Revision 2). This demonstration evaluation using the methodology of Reference 4 is applied using the criteria of RG 1.236 (Reference 3). It is anticipated that the acceptance and failure criteria of RG 1.236 will be applied in future applications of this methodology at Monticello.

3.0 INITIAL METHODOLOGY DEMONSTRATION

The initial application of the AURORA-B CRDA methodology involves sensitivity studies and determination of an evaluation boundary. The determination of the evaluation boundary provided in Appendix A is a demonstration of the process discussed in Reference 4 for Monticello with ATRIUM 11 fuel. This same process will be followed for transition cores.

3.1 *Initial Conditions*

Sensitivity studies are performed **[**

]

3.2 *Group Pull Sequence*

All allowed pull orders are evaluated such that each control rod group, with the exception of groups 5 and 6, is pulled as the second group as indicated in Table 3.1. The third and fourth groups are assumed to be banked. It is assumed that the first and second groups selected for withdrawal are completely withdrawn prior to pulling control rods in the third group. For clarification since both the first and second groups must be out before the third group, both pull sequences A1234 and A2134 have the same starting control rod pattern for the third group. Therefore the sequences A1234 and A2143 also cover sequences A2134 and A1243.

Table 3.1 Group Pull Sequences

Group Numbers	Analyzed Groups for both A and B Sequences
1st and 2nd Groups	(1,2), (2,1), (3,4), (4,3)
3rd and 4th Groups	(3,4), (4,3), (1,2), (2,1)

3.3 *Inoperable Control Rod Positions*

A maximum of 8 inoperable control rods are allowed for this plant with up to three inoperable per group. The size of the Monticello core does not accommodate more than two inoperable control rods per group per sequence without having the inoperable control rods infringe upon the quarter of the core where the rod drop analysis is conducted. Having inoperable control rods in the quarter of the core where the rod drop analysis is conducted prevents the power in the core from peaking properly in the quarter of interest and has the potential to dampen the severity of rod drop analysis results. Thus, two inoperable control rods were selected for each of the first four groups withdrawn in each sequence. The assignment of inoperable control rods adhered to the separation criteria on group bases.

Given the uniform core configuration for the equilibrium cycle, the prior analyses in Reference 4 being limited by drops based on inoperable rod configurations, and that investigations showed drops without inoperable rods are overall non-limiting in this analysis, only drops with inoperable control rod configurations were fully evaluated. The selected inoperable control rods for drops in the second, third, and fourth groups are identified in Figure 3.1. For each set of inoperable rods, all control rods in the next group are dropped to evaluate the impact of the inoperable rods. Therefore the position of the inoperable control rods was evaluated based on dropping all rods.

Figure 3.1 Inoperable Rods for 1st Through 4th Group Pulls

3.4 *Time in Cycle*

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3.5 *Group Critical Position*

The first step is to evaluate the end of group or bank position k-effective values to determine where criticality is anticipated to occur for the given control rod withdrawal sequence. The near critical range, determined per Section 7.4.1 of Reference 4, is given in Table 3.2. The calculated k-effective values at the end of groups 1 through 4 for the A and B sequence withdrawals are given in Table 3.3. [

]

Table 3.2 Near Critical Range

[ ]

Table 3.3 Group Out Eigenvalues

[ ]

3.6 *Determination of Static Control Rod Worth*

Based on the results of the group worth, static control rod worths were then determined. From the static rod drops, [

] The selected second group rods for transient evaluation are given in Table 3.4 and the third group rods are given in Table 3.5.

Table 3.4 Selected Rods with Inoperable Control Rods 2nd Group

[ ]

Table 3.5 Selected Rods with Inoperable Control Rods 3rd and 4th Group

[ ]

3.7 *Transient Evaluation*

The evaluation of each rod drop is performed with the AURORA-B system. The initial pre-rod drop state point is established with the MICROBURN-B2 core simulator. The initial conditions used for the transient calculation are identified in Table 3.6.

Table 3.6 Initial Conditions

[ ]

The channel grouping with a [ ] is used for this analysis. (Figure 3.2 illustrates the assemblies evaluated for the drop of control rod [ ] .) Once the channel grouping is defined, the power history information is processed to obtain the fuel rod characteristics for use in the RODEX-4 fuel rod mechanical models.

The maximum prompt enthalpy increase for the peak fuel rod and the maximum total enthalpy reported include the application of the uncertainty multiplier of [ ] on the enthalpy increase. The prompt enthalpy increase along with total enthalpy for the second group drops is given in Table 3.7. Likewise Table 3.8 contains the prompt enthalpy increase and total enthalpy for third group control rods (the third group bounded the fourth group.) Although there are high worth banked drops, the actual nodal enthalpy increase is small for the BOC banked drops compared to drops later in cycle with a top peaked power shape.

Figure 3.2 Map of Assemblies Evaluated for Drop of Control Rod [ ]

Table 3.7 Maximum Prompt Enthalpy Rise and Total Enthalpy 2nd Group

Table 3.8 Maximum Prompt Enthalpy Rise and Total Enthalpy 3rd Group

4.0 EVALUATION AGAINST CLADDING FAILURE CRITERIA

4.1 *High Temperature Cladding Failure*

[

]

Table 4.1 Assemblies with Fuel Rod High Temperature Failures

[ ]

Figure 4.1 Total Enthalpy Versus High Temperature Cladding Failure Threshold All Drops

Figure 4.2 High Temperature Nominal and High Burnup for Drops [ ] and [ ]

4.2 *PCMI Cladding Failure*

The ATRIUM 11 fuel is clad with stress relief annealed (SRA) Zircaly-2 cladding. (Framatome uses the term Cold Work Stress Relieved CWSR to refer to SRA material.) Therefore, the SRA low temperature failure threshold is applied.

To establish the minimum failure threshold, the maximum fuel rod nodal hydrogen at end of cycle was tabulated for each assembly using the hydrogen model of Reference 8 [

]

[ ]

Figure 4.3 Minimum Failure Threshold Based on EOC Hydrogen Content

Since 150 Δcal/g is the maximum of the failure threshold curve, [

] The rod drops are evaluated with assumed inoperable control rods. [

]

[

]

Figure 4.4 PCMI Cladding Failure Results

4.3 *Molten Fuel Cladding Failure Threshold*

[ ]

5.0 RADIOLOGICAL CONSEQUENCES

The dose consequences for the CRDA determined for Monticello are summarized in the USAR. The licensing basis does evaluation based on ATRIUM 10XM fuel determined that 1,206 (850 8x8 equivalent from the USAR) fuel rods could fail for Monticello. Since the actual number of ATRIUM 11 allowed fuel rod failures has not been determined at this time, it is assumed that the allowed number of failures will be similar to that of ATRIUM 10XM. Therefore, demonstrating that there is significantly less than 1,206 fuel rod failures will confirm that the radiological consequences are bounded by those given in the USAR.

Evaluation of dose consequences for fuel rod failures

Since fuel rod failures had been identified, revised release fractions or total release fraction (TOTR) are determined using the Licensing Basis Release Fractions (LBRF) from RG 1.183 conservatively applied as the steady state release fractions (SSRF) with the transient fission gas release (TFGR) as described in RG 1.236. A ratio of the new TOTR to the LBRF used in the original licensing basis is then generated following the method provided in ANP-10333PA.

The transient release terms, from Reference RG 1.236, expressed as a fraction are:

Peak Pellet BU < 50 GWd/MTU:

$$TFGR = \frac{[(0.26 * \Delta H) - 13]}{100} \geq 0$$

Peak Pellet BU ≥ 50 GWd/MTU:

$$TFGR = \frac{[(0.26 * \Delta H) - 5]}{100} \geq 0$$

The total fuel rod release fraction TOTR is dependent on the nuclide group and the enthalpy dependent TFGR average over the 24 nodes of fuel for a full length fuel rod. (If the failure were in a different length fuel rod, the number of axial nodes would be adjusted accordingly.)

Burnup < 50:

$$TOTR = SSRF + \frac{\sum_k \frac{[(0.26 * \Delta H) - 13]}{24}}{100} * GMUL$$

Burnup ≥ 50:

$$TOTR = SSRF + \frac{\sum_k \frac{[(0.26 * \Delta H) - 5]}{24}}{100} * GMUL$$

Where,

- ΔH fuel enthalpy increase (cal/g)
- SSRF is the steady state release fraction
- Three multipliers (GMUL) are established in DG 1327 (Reference 10) to be applied to the above TFGR term:

Group	GMUL	Applied to
Stable long lived isotopes (e.g., Kr-85)	1.0	Kr-85
Cs-134 and Cs-137	1.414	Alkali Metals
Short-lived radioactive isotopes (i.e., I, Xe and Kr noble gases except Kr-85)	0.333	Iodines, nobles, halogens

As noted above, the LBRF are utilized for Monticello as the SSRF for the respective groups.

For this analysis of the ATRIUM 11 fuel, a maximum of **[** **]** fuel rods for any control rod drop case exceeded one or more failure criteria. Based on the enthalpy increase, the enthalpy dependent release terms were determined for the fuel rods. The transient fission gas release fractions are provided in Table 5.1 based on nodal values of the peak fuel rod enthalpy increase. **[**

] The total release fraction and the ratio to the licensing bases release fraction are provided in Table 5.2. The actual number of fuel rod failures is provided in Table 5.3. **[**

]

[ ] ; therefore this event remains within the current evaluated dose consequences for Monticello.

Table 5.1 Transient Fission Gas Release Fractions

[ ]

Table 5.2 Total Fission Gas Release Fractions

[ ]

Table 5.3 Fuel Rod Failures

[ ]

[ ]

[ ]

6.0 SYSTEM PRESSURE AND CPR

The impact of the CRDA on system pressure was addressed in Reference 4 and does not cause stresses to exceed Emergency Condition (Service Level C), as defined in Section III of the ASME Boiler and Pressure Vessel code. This generic evaluation on the impact of CRDA on system pressure remains applicable for Monticello.

The CPR response was evaluated in Section 7.7 of Reference 4 and results in a conclusion that the CRDA in the power range is **[**

]

7.0 CORE COOLABILITY

Two criteria are identified in Reference 3 for allowable limits with respect to core coolability.

- Peak radial average enthalpy < 230 cal/g
- The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions

[

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Table 7.1 Peak Radial Average Fuel Enthalpy

[ ]

8.0 LIMITATIONS AND CONDITIONS

The SER for the Reference methodology included a number of limitations and conditions. Some of the conditions are from the base AURORA-B AOO methodology (Reference 6) and additions specific to the CRDA are included. The number of the limitations and conditions below is consistent with that found in the AURORA-B CRDA SER.

1. AURORA-B may not be used to perform analyses that result in one or more of its CCDs (S-RELAP5, MB2-K, MICROBURN-B2, RODEX4) operating outside the limits of approval specified in their respective TRs, SEs, and plant-specific license amendment requests (LARs). In the case of MB2-K, MB2-K is subject to the same limitations and conditions as MICROBURN-B2. *(This is Conditions 1 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

This condition is met for application of the AURORA-B CRDA methodology to Monticello.

14. The scope of the NRC staff's approval of AURORA-B does not include the ABWR design. *(This is condition 14 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

This condition is met for Monticello since it is a BWR/3.

20. The implementation of any new methodology within the AURORA-B EM (i.e., replacement of an existing CCD) is not acceptable unless the AURORA-B EM with the new methodology incorporated into it has received NRC review and approval. An existing NRC-approved methodology cannot be implemented within the AURORA-B EM without NRC review of the updated EM. *(This is a revised version of Condition 20 of the SE for the base AURORA-B TR, rewritten to be specific to the CRDA application. It remains applicable to CRDA analyses for BWRs/2-6.)*

The evaluation model will be implemented for Monticello as described in the base AURORA-B and AURORA-B CRDA Topical Reports. No CCD as described in the TR are replaced and therefore the intent of this condition is met.

21. NRC-approved changes that revise or extend the capabilities of the individual CCDs comprising the AURORA-B EM may not be incorporated into the EM without prior NRC approval. *(This is Condition 21 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6)*

[

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22. As discussed in Section 3.3.1.5 and Section 4.0 of Reference 6 (the SE for the base AURORA-B TR), the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EM. Other CPR correlations (existing and new) that would be used with the AURORA-B EM must be reviewed and approved by the NRC or must be developed with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel". Furthermore, if transient thermal-hydraulic simulations are performed in the process of applying AREVA CPR correlations to co-resident fuel, these calculations should use the AURORA-B methodology. *(This is Condition 22 of the SE for the base AURORA-B TR. It remains applicable to at-power CRDA analyses for BWRs/2-6.)*

This condition is met within ANP-10333PA for at power evaluations. The ACE ATRIUM 11 Correlation has been reviewed and approved by the NRC (Reference 7).

23. Except when prohibited elsewhere, the AURORA-B EM may be used with new or revised fuel designs without prior NRC approval provided that the new or revised fuel designs are substantially similar to those fuel designs already approved for use in the AURORA-B EM (i.e., thermal energy is conducted through a cylindrical fuel pellet surrounded by metal cladding, flow in the fuel channels develops into a predominantly vertical annular flow regime, etc.). New fuel designs exhibiting a large deviation from these behaviors will require NRC review and approval prior to their implementation in AURORA-B. *(This is Condition 23 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

This condition is met as ATRIUM 11 does exhibit the structural similarities described in the restriction.

24. Changes may be made to the AURORA-B EM in the [ ] areas discussed in Section 4.0 of Reference 6 (the SE for the base AURORA-B TR) without prior NRC approval. (This is Condition 24 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)

This condition is met through the use of the Framatome software development procedures.

25. The parallelization of individual CCDs may be performed without prior NRC approval as discussed in Section 4.0 of Reference 6 (the SE for the base AURORA-B TR). (This is Condition 25 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)

No confirmation is required for this condition.

26. AREVA must continue to use existing regulatory processes for any code modifications made in the [ ] areas discussed in Section 4.0 of

Reference 6 (the SE for the base AURORA-B TR). *(This is Condition 26 of the SE for the base AURORA-B TF. It remains applicable to CRDA analyses for BWRs/2-6.)*

This condition is met through the use of the Framatome software development procedures which include 10CFR50.59 licensing considerations.

27. The control rod model at each location in the core used for CRDA analyses with the AURORA-B EM shall use a control rod geometry and composition that is verified to bound the control rod worth for the physical control rod used in the location, for all axial elevations.

[

] Therefore, this condition is met.

28. Licensees utilizing AURORA-B to perform CRDA analyses using the methodology described in this TR shall confirm that the recommended maximum rod velocity of 3.11 ft/s is conservative for their control rods.

The licensee has confirmed that this condition is met for the control rods at Monticello.

29. If the check to verify that the total enthalpy is limiting at 10 percent core flow CZP conditions by [

] fails, AREVA shall perform a more comprehensive evaluation to verify that they have identified the limiting initial conditions for that plant. This evaluation should consider a range of flow values and corresponding plant-specific minimum temperatures that is sufficiently broad to clearly identify the combination of initial conditions which maximizes the total enthalpy for the limiting rod.

[

]

Monticello with ATRIUM 11 fuel for determining the total enthalpy.

30. When individual control rods are evaluated using the CRDA analysis methodology, if necessary, alternate distributions of inoperable rods should be utilized to ensure inclusion of at least one evaluation within each group of 4 quadrant symmetric control rods that maximizes the change in face- and/or diagonally-adjacent uncontrolled cells as a results of the candidate control rod withdrawal.

[

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31. The evaluation boundary curve used to determine candidate control rods for further evaluation based on their static rod worths must be verified to bound the following local characteristics of the fuel being evaluated: design pin peaking factors, fuel assembly design, location in or adjacent to the outermost ring of control rods, and average burnup for the 16 fuel assemblies surrounding the rod of interest.

This condition is addressed in Appendix A for this demonstration analysis.

32. If the highest worth rod at a given core statepoint results in a total enthalpy that is higher than the minimum high temperature failure threshold (i.e., lowest threshold for all rod internal pressures), additional rods must be considered for evaluation. This may be done by evaluating the next highest worth rods at the core statepoint of interest until the minimum high temperature failure threshold is met, or by using an approach analogous to the evaluation boundary curve for the PCMI failure threshold (as subject to condition 29).

The highest control rod worth did result in a total enthalpy which exceeded the minimum high temperature failure threshold. Therefore, additional control rods were evaluated to address this condition (see Section 4.1).

33. If the methodology described in ANP-10333 is used to analyze the CRDA event with a fuel assembly design that has a different fuel rod geometry and/or manufacturing tolerances than the one used as a basis for the sensitivity study on gap width, the sensitivity study shall be repeated for the new fuel assembly design, using bounding values consistent with the uncertainty range for [ ] limiting increase in the peak total enthalpy, the total uncertainty shall be increased accordingly for total enthalpies calculated based on the new fuel assembly design.

The ATRIUM 11 product line requires an evaluation of the gap sensitivity study. The sensitivity studies were performed with a bounding value for the uncertainty range of [ ]. The resulting increase in peak total enthalpy [ ]

34. The uncertainty designate in the CRDA TR of [ ] for the enthalpy rises calculated using the CRDA analysis methodology may not be reduced without prior NRC approval.

The uncertainty of [ ] percent is used in this evaluation.

Other conditions:

In Section 3.0 of the SER, the NRC discussed the evolving CRDA criteria at the time of approval of ANP-10333. The final criteria (RG 1.236) must be reviewed to confirm that any changes relative to DG-1327 other than clarifications, editorial changes, or numeric thresholds must be evaluated.

[

]

[

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9.0 REFERENCES

1. ANP-3881P, Revision 0, Monticello ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report, Framatome, November 2020.

2. NUREG-0800, Section 15.4.9, Revision 3, SPECTRUM OF ROD DROP ACCIDENTS (BWR). *Standard Review Plan: LWR Edition,* US NRC: Washington, DC. March 2007.

3. RG 1.236, Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents, June 2020. (NRC ADAMS ML20055F490)

4. ANP-10333P-A, Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Framatome, March 2018.

5. NUREG-0800, Section 15.2.9, Revision 2, SPECTRUM OF ROD DROP ADDICENTS (BWR). *Standard Review Plan: LWR Edition,* US NRC: Washington, DC. July 1981.

6. ANP-10300P-A, Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome, January 2018.

7. ANP-10335P-A, Revision 0, ACE/ATRIUM 11 Critical Power Correlation Topical Report, Framatome, May 2018.

8. BAW-10247PA, Revision 0, Supplement 1P-A Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding, AREVA, April 2017.

9. ANP-3924P, Revision 0, Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel, Framatome Inc., June 2021.

10. DG-1327, Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents, July 2019 (NRC ADAMS ML18302A106).

Appendix A Evaluation Threshold Determination

Limitation and Condition 31 states:

31. The evaluation boundary curve used to determine candidate control rods for further evaluation based on their static rod worths must be verified to bound the following local characteristics of the fuel being evaluated: design pin peaking factors, fuel assembly design, location in or adjacent to the outermost ring of control rods, and average burnup for the 16 fuel assemblies surrounding the rod of interest.

The process to generate an evaluation threshold is demonstrated based upon the process described in the response to RAI-5 in ANP-10333Q1P (included in Reference 4). The peak fuel rod enthalpy rise was elevated using a multiplication factor of [ ] to double the uncertainty. The elevated enthalpy rise values were then tabulated against the static control rod worth.

[ ]

Figure A.1 Data for Evaluation Boundary

Figure A.2 Example Evaluation Boundary for the Monticello ATRIUM 11 Core

Based on the example evaluation boundary in Figure A.2 created using the data in Figure A.1, any case below and to the right of the boundary should be investigated. When using an evaluation boundary line, consideration of high burnup assemblies with lower PCMI threshold limits is required and they should be conservatively assessed.

The local characteristics of fuel used to establish the evaluation boundary with respect to design fuel rod peaking factors, fuel assembly design, core location, and the average burnup of the 16 assemblies around the dropped rod have been are provided in Table A.1. This table is for use with future core licensing to confirm the applicability of the evaluation boundary curve.

Table A.1 Example Evaluation Boundary Characteristics

[ ]

[ ]

ENCLOSURE

ATTACHMENT 10b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3929P REPORT, REVISION 0

MONTICELLO ATRIUM 11 CONTROL ROD DROP ACCIDENT ANALYSES
WITH THE AURORA-B CRDA METHODOLOGY

JUNE 2021

(3 pages follow)

AFFIDAVIT

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3929P Revision 0, "Monticello ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology," dated June 2021 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: June 22, 2021

Alan B. Meginnis

ENCLOSURE

ATTACHMENT 11a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3925NP REPORT, REVISION 0

MONTICELLO ATRIUM 11 TRANSIENT DEMONSTRATION

JULY 2021

(88 pages follow)

Monticello ATRIUM 11 Transient Demonstration

ANP-3925NP
Revision 0

July 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

Tables

Figures

Nomenclature

AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
AST	alternate source term
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram recirculation pump trip
BOC	beginning of cycle
BWR	boiling water reactors
COLR	core operating limits report
CRDA	control rod drop accident
EFW	extended flow window
EM	evaluation model
EOFP	end of full power
EOFPLB	end of full power licensing basis
EOOS	equipment out of service
FoM	figure of merit
FWCF	feedwater controller failure
HPCI	high pressure coolant injection; also used to designate the inadvertent HPCI start-up event
LAR	license amendment request
LFWH	loss of feedwater heating
LHGR	linear heat generation rate
LHGRFACp	power dependent linear heat generation rate multipliers
LOCA	loss of coolant accident
LRNB	generator load rejection no bypass
LTR	licensing topical report
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MCPRp	power dependent minimum critical power ratio
MSIV	main steam isolation valve
NEOC	near end of cycle
NRC	Nuclear Regulatory Commission, U.S.
NSS	nominal scram speed

Nomenclature (continued)

Pbypass	power below which direct scram on TSV/TCV closure is bypassed
PRFO	pressure regulator failure open
RHR	residual heat removal
SRV	safety/relief valve
SRVOOS	safety/relief valve out-of-service
TCV	turbine control valve
TSV	turbine stop valve
TTNB	turbine trip no bypass
USAR	updated safety analysis report
ΔMCPR	change in minimum critical power ratio

1.0 INTRODUCTION

This report summarizes the results of a subset of transient analyses performed to support the license amendment request (LAR) to include the Reference 1 and Reference 2 Licensing Topical Reports (LTR) into the Monticello Nuclear Generation Plant (MNGP) Technical Specifications.

For a typical reload, a full assessment of the power/flow map, cycle exposure, and scram speed are done on a cycle specific basis for the actual core configuration to develop thermal limits. The intention of this report is to demonstrate the applicability of the AURORA-B AOO methodology (Reference 1) to Monticello for the transient analyses that are typically limiting on a cycle-specific basis. Therefore, this document is a subset of transient analysis typically performed for each cycle.

The analyses presented in Section 4 of this document are based upon a representative equilibrium cycle of ATRIUM 11 fuel, Reference 3. A variety of power/flow state points are performed at a cycle exposure and scram speed discussed in each subsection of Section 4.0.

The AURORA-B AOO analysis is used to calculate the change in the minimum critical power ratio (ΔMCPR) during the anticipated operational occurrence (AOO). The ΔMCPR is combined with the safety limit MCPR to establish or confirm the plant operating limits for MCPR.

The AURORA-B AOO analysis is also used to calculate the maximum reactor vessel pressure and the maximum dome pressure during the ASME and ATWS events. The calculated maximum reactor vessel pressure is compared to the ASME acceptance criterion (110% of vessel design pressure) and the calculated maximum steam dome pressure is compared to the pressure safety limit in the plant Technical Specifications. For the ATWS event, the calculated maximum reactor vessel pressure is compared to ASME Service Level C (120% of design pressure) to demonstrate that the event

acceptance criterion is met. Meeting the acceptance criteria confirms that the plant safety valve performance (number of valves available, capacity per valve, and setpoints) is acceptable.

The ACE/ATRIUM 11 critical power correlation (Reference 2) is used to evaluate the thermal margin of the ATRIUM 11 fuel.

2.0 ANTICIPATED OPERATIONAL OCCURRENCES

2.1 *AURORA-B AOO Evaluation Model*

AURORA-B is a comprehensive evaluation model developed for predicting the dynamic response of boiling water reactors (BWRs) during transient, postulated accident, and beyond design-basis accident scenarios. The evaluation model (EM) contains a multi-physics code system with flexibility to incorporate all the necessary elements for analysis of the full spectrum of BWR events that are postulated to affect the nuclear steam supply system of the BWR plant. Deterministic analysis principles are applied to satisfy plant operational and Technical Specification requirements through the use of conservative initial conditions and boundary conditions.

The foundation of AURORA-B AOO is built upon three computer codes, S-RELAP5, MB2-K, and RODEX4. Working together as a system, they make up the multi-physics evaluation model that provides the necessary systems, components, geometries, processes, etc. to assure adequate predictions of the relevant BWR event characteristics for its intended applications. The three codes making up the foundation of the code system are:

- S-RELAP5 – This code provides the transient thermal-hydraulic, thermal conduction, control systems, and special process capabilities (i.e. valves, jet-pumps, steam separator, critical power correlations, etc.) necessary to simulate a BWR plant.

- MB2-K – This code uses advanced nodal expansion methods to solve the three-dimensional, two-group, neutron kinetics equations. The MB2-K code is consistent with the MICROBURN-B2 steady state core simulator. MB2-K receives a significant portion of its input from the steady state core simulator.

- RODEX4 – A subset of routines from this code are used to evaluate the transient thermal-mechanical fuel rod (including fuel/clad gap) properties as a function of temperature, rod internal pressure, etc. The fuel rod properties are used by S-RELAP5 when solving the transient thermal conduction equations in lieu of standard S-RELAP5 material property tables.

2.2 *Description of [ ] Analysis Process*

The AURORA-B AOO methodology (Reference 1) includes an evaluation of the impact of code uncertainties on Figures of Merit (FoM) (e.g. ΔMCPR, peak pressure) **[ ]** that has wide acceptance in the nuclear industry.

[ ]

Table 2.1 [

]

[ ]

2.2.1 **Sampled Parameters [ ]**

The set of code and modeling uncertainty parameters to be sampled for AOO calculations is shown in Table 2.2.

[ ]

[ ]

2.2.2 **Sampling Ranges**

The sampling ranges shown in Table 2.2 are applicable to Monticello.

Per the approved methodology (Reference 1), the sampling ranges address uncertainties inherent in the S-RELAP5 models **[**

]

[ ]

* Framatome Inc. formerly known as AREVA Inc.

A description of the basis for the sampling ranges used for each of the above sampled variables is found in Reference 1 Safety Evaluation, Sections 3.6.4.1 – 3.6.4.17.

Table 2.2 Sampling Ranges for Uncertainty Parameters

2.3 *Application [ ] for Demonstration Cases*

The statistical analysis process presented in the previous sections will be used to determine the **[ ]** values for FoMs associated with the nominal transient simulations performed to demonstrate the methodology application to the equilibrium ATRIUM 11 core. Section 3.6.5 of the Safety Evaluation (Reference 1) allows for subsequent analyses to utilize the **[ ]** to determine base conservative measures to be applied for calculation of the key FoM in future reload licensing. **[**

]

3.0 AURORA-B AOO METHODOLOGY IMPLEMENTATION TO MONTICELLO USAR CHAPTER 14 EVENTS

3.1 *Disposition of Events*

The objective of the disposition of events is to identify the limiting events which must be analyzed to support operation at the Monticello Nuclear Generation Plant with the introduction of ATRIUM 11 fuel. Events and analyses identified as potentially limiting are either evaluated generically or on a cycle-specific basis.

The first step is to identify the licensing basis of the plant. Included in the licensing basis are descriptions of the postulated events/analyses and the associated criteria. Fuel-related system design criteria must be met, ensuring regulatory compliance and safe operation. The licensing basis, related to fuel and applicable for reload analysis, is contained in the Updated Safety Analysis Report (USAR), the Technical Specifications, Core Operating Limits Reports (COLR), and other reload analysis reports.

For the introduction of ATRIUM 11 fuel, Framatome reviewed all fuel-related design criteria, events, and analyses identified in the licensing basis. In many cases, when operating limits are established to ensure acceptable consequences of an abnormal operational occurrence (AOO) or accident, the fuel-related aspects of the system design criteria are met. All fuel-related events were reviewed and dispositioned into one of the following categories:

1. No further analysis required. This classification may result from one of the following:
   a. The consequences of the event are bound by consequences of a different event.
   b. The consequences of the event are benign, i.e., the event causes no significant change in margins to the operating limits.
   c. The event is not affected by the introduction of ATRIUM 11 fuel and/or the current analysis of record remains applicable.
2. Address event each reload. The consequences of the event are potentially limiting and need to be addressed each reload.

3. Address for initial reload. This classification may result from one of the following:
   a. The analysis is performed using conservative bounding assumptions and inputs such that the initial reload results will remain applicable for future reloads of the same fuel design.
   b. Results from the first reload will be used to quantitatively demonstrate that the results remain applicable for future reloads of the same fuel design because the consequences are benign or bound by those of another event.

A disposition of events summary is presented in Table 3.1. The disposition summary presents a list of the events and analyses, the corresponding USAR section, the disposition status, and any applicable comments.

Table 3.1 Disposition of Events Summary for Introduction of ATRIUM 11 Fuel at Monticello

USAR Section	Event /Analysis	Disposition Status	Comments
14.4.1	Generator Load Rejection Without Bypass	Address for initial reload.	Results for this event are expected to be similar to, but less limiting than the turbine trip without bypass event (USAR Section 14.4.5) due primarily to the difference in valve closure time. The generator load rejection with bypass is bound by the generator load rejection without bypass.
14.4.2	Loss of Feedwater Heating	Address each reload.	Because the change in feedwater temperature for MNGP occurs in less than 80 seconds, the generic LFWH analysis is not applicable. This event is a potentially limiting AOO and will be addressed each reload using the AURORA-B AOO methodology.
14.4.3	Rod Withdrawal Error – Low Power	No further analysis required.	Consequences of a RWE below the low power setpoint are bound by RWE at power due to required strict compliance with BPWS.
14.4.3	Rod Withdrawal Error – At Power	Address each reload.	This event is a potentially limiting AOO.
14.4.4	Feedwater Controller Failure – Maximum Demand	Address each reload.	This event is a potentially limiting AOO and will be addressed each reload using the AURORA-B AOO methodology.
14.4.5	Turbine Trip Without Bypass	Address each reload.	This event is a potentially limiting AOO and will be addressed each reload using the AURORA-B AOO methodology. Turbine trip with bypass is bound by turbine trip without bypass.
14.5.1	Vessel Pressure ASME Code Compliance Model	Address each reload.	The ASME overpressurization event will be addressed each reload using the AURORA-B AOO methodology.
14.5.2	Standby Liquid Control System Shutdown Margin	Address each reload.	Standby liquid control system shutdown capability will be addressed each reload.

USAR Section	Event /Analysis	Disposition Status	Comments
14.5.3	Stuck Rod Cold Shutdown Margin	Address each reload.	This event is potentially limiting and will be addressed each reload.
14.6	Plant Stability Analysis	Address each reload.	Analyses will be performed with the Best-estimate Enhanced Option III (BEO-III) stability solution each cycle beginning with the ATRIUM 11 fuel transition. Backup stability protection (BSP) calculations will continue to be performed each cycle.
14.7.1	Control Rod Drop Accident Evaluation	Address each reload.	This event is potentially limiting and will be addressed each reload with the AURORA-B CRDA methodology. Evaluation of ATRIUM 11 with AST is required.
14.7.2	Loss-of-Coolant Accident	Address for initial reload.	LOCA break spectrum and exposure-dependent LOCA calculations will be performed for ATRIUM 11 fuel using the AURORA-B LOCA methodology. Consequences of the LOCA are evaluated to determine appropriate fuel-specific MAPLHGR limits which are independent of cycle specific assembly designs. Limiting power history, gad LHGR confirmation, and MAPLHGR checks are performed for follow-on reloads.
14.7.3	Main Steam Line Break Accident Analysis	No further analysis required.	Consequences of a main steam line break outside containment are independent of fuel design since the radioactive release is dependent on primary coolant activity and not on fuel design parameters.
14.7.4	Fuel Loading Error Accident	Address each reload.	While this event is classified as an infrequent event, it will be conservatively analyzed using the AOO acceptance criteria. The fuel loading error is addressed for each reload and addresses a mislocated or misoriented assembly.

USAR Section	Event /Analysis	Disposition Status	Comments
14.7.5	One Recirculation Pump Seizure Accident Analysis	Address each reload.	While this event is classified as an accident, it will be conservatively analyzed using the AOO acceptance criteria with the AURORA-B AOO methodology. Pump seizure in two-loop operation is expected to be non-limiting and will be verified as non-limiting for the initial reload. Pump seizure in single-loop operation is a potentially limiting AOO and will be addressed each reload.
14.7.6	Refueling Accident Analysis	Address for initial reload.	The number of fuel rods assumed to fail during a fuel handling accident for an ATRIUM 11 assembly dropping over the core will be analyzed in support of the fuel transition. This is independent of operation of ATRIUM 11.
14.7.7	Accident Atmospheric Dispersion Coefficients	No further analysis required.	The atmospheric dispersion coefficient values in the analysis of record are not fuel type dependent and remain valid.
14.7.8	Core Source Term Inventory	Address for initial reload.	Evaluation of ATRIUM 11 with AST is required.
14.8	Anticipated Transients Without Scram (ATWS)	Address each reload.	The ATWS overpressurization event will be addressed each reload using the AURORA-B AOO methodology. [ ] (Reference 4). Peak cladding temperature and oxidation are bound by LOCA. ATWS with Instability (ATWS-I) is addressed in Reference 5.
14.10.1	Adequate Core Cooling for Transients With a Single Failure	No further analysis required.	Previous analyses demonstrated that the reactor core remained covered for the worst combination of conditions, so the results are not fuel type dependent.

USAR Section	Event /Analysis	Disposition Status	Comments
14A Section 5.0	GE SIL 502 (Revision 1) Single Turbine Control Valve Slow Closure Event	No further analysis required.	Application of the generic analysis to Monticello demonstrates this analysis is far from limiting. Previous analyses by Framatome also demonstrated this analysis was non-limiting. This event is bound by other AOO and does not need to be addressed each cycle.
14A Section 5.0	Pneumatic System Degradation (Turbine Trip with Bypass and degraded scram speed)	Address for initial reload.	Results for this event are expected to be bound by other AOO (i.e., FWCF) and the event will be analyzed with AURORA-B AOO for the initial reload of ATRIUM 11 fuel to confirm the results remain bounded.
14A Section 5.0	Loss of Stator Cooling	Address for initial reload.	Results for this event are expected to be non-limiting and the event will be analyzed with AURORA-B AOO for the initial reload of ATRIUM 11 fuel to confirm the results remain non-limiting.
14A Table 5.1	Main Steam Isolation Valve Closure (One / All Valves)	No further analysis required.	Consequences of this event are bound by the turbine trip without bypass (USAR Section 14.4.5). Closure of all MSIVs with failure of the valve position scram function is addressed each reload to show compliance with the ASME vessel overpressure protection (USAR Section 14.5.1). The MSIV closure event is addressed each reload as a potentially limiting ATWS overpressure event (USAR Section 14.8).
14A Table 5.1	Loss of Condenser Vacuum	No further analysis required.	Consequences of this event are bound by the turbine trip without bypass (USAR Section 14.4.5).
14A Table 5.1	Pressure Regulator Failure – Full Close (Downscale)	Address each reload.	This event, with one pressure regulator out-of-service, is a potentially limiting AOO and will be addressed each reload using the AURORA-B AOO methodology.
14A Table 5.1	Loss of Auxiliary Power – All Grids	No further analysis required.	Consequences of this event are bound by the turbine generator load rejection without bypass (USAR Section 14.4.1).

USAR Section	Event /Analysis	Disposition Status	Comments
14A Table 5.1	Inadvertent High Pressure Coolant Injection	Address each reload.	This event is a potentially limiting AOO and will be addressed each reload using the AURORA-B AOO methodology.
14A Table 5.1	Pressure Regulator Failure – Full Open	No further analysis required.	Consequences of this event relative to thermal operating limits are non-limiting. The TS 2.1.1.1 low dome pressure safety limit protects the lower boundary of the ACE/ATRIUM 11 critical power correlation. Previous analyses conservatively showed that this safety limit is protected with the current MSIV steam line low pressure trip setpoint. This event is also analyzed as an initiator for ATWS overpressure (USAR 14.8).
14A Table 5.1	Inadvertent Opening of Safety/Relief Valve	No further analysis required.	The depressurization for this event is less severe than the pressure regulator failure – full open event. Since the power level settles out at nearly the initial power, this event is considered benign.
14A Table 5.1	Loss of Feedwater Flow	No further analysis required.	This event does not pose any direct threat to the fuel in terms of a thermal power increase from the initial conditions. The fuel will be protected provided the water level inside the shroud does not drop below the top of active fuel. Previous evaluations for a different fuel design showed that the lowest level following a loss of feedwater event remained well above the top of active fuel. The long term water level transient is dependent upon the decay heat which is **[** **]**. This is a benign event.
14A Table 5.1	Loss of Auxiliary Power Transformers	No further analysis required.	Consequences of this event are bound by the turbine generator load rejection without bypass (USAR Section 14.4.1).
14A Table 5.1	Recirculation Flow Control Failure – Decrease	No further analysis required.	Consequences of this event are bound by a recirculation pump trip.

USAR Section	Event /Analysis	Disposition Status	Comments
14A Table 5.1	Trip of Two Recirculation Pumps	No further analysis required.	Consequences of this event are benign and bound by the turbine generator load rejection without bypass (USAR Section 14.4.1).
14A Table 5.1	Slow Recirculation Control Failure – Increase (MCPRf)	Address each reload.	This event is used to determine the flow-dependent MCPR operating limit and will be addressed each reload.
14A Table 5.1	Slow Recirculation Control Failure – Increase (LHGRFACf)	Address each reload.	This event is used to determine the flow-dependent LHGR setdown factors and will be addressed each reload.
14A Table 5.1	Fast Recirculation Control Failure – Increase	Address each reload.	This event is a potentially limiting AOO and will be addressed each reload using the AURORA-B AOO methodology.
14A Table 5.1	Startup of an Idle Recirculation Loop	No further analysis required.	The consequences of this event are bound by inadvertent HPCI startup.

3.2 *Cycle-Specific Calculation Plan Development and Analysis Considerations for Limit Generation and Plant Operation*

The disposition of events evaluation for a plant defines the transient events to be analyzed on a cycle specific basis. Prior to a specific cycle's licensing campaign, a calculation plan is generated, which defines the minimum analysis set required to license a given cycle. This plan is reviewed and approved by Monticello. Additional analyses may be added during the evaluation process if unexpected trends arise. These are added on an as-needed basis to ensure that the thermal limits used to monitor actual plant operation are appropriately developed.

The calculation plan will also define all operational flexibility options that are to be supported. System pressurization transient results are sensitive to scram speed assumptions. The calculation plan will provide details regarding what scram insertion

speeds will be supported for cycle specific analyses. Additional items discussed are equipment out-of-service options (EOOS) and exposure windows.

3.2.1 **Exposure Analysis**

Cycle specific thermal limits are developed based on exposure ranges discussed in the calculation plan. The limiting exposure for rated power pressurization transients is typically at end of full power (EOFP) when the control rods are fully withdrawn. To provide additional margin to the operating limits, thermal limits are typically developed for multiple exposures ranges for the cycle being evaluated. The typical exposure ranges cover beginning-of-cycle (BOC) to a near end-of-cycle (NEOC) core average exposure, BOC to the end of full power licensing basis (EOFPLB) analysis, and BOC to power coastdown.

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3.2.2 **State Point Determination**

The statepoints to be analyzed are also discussed in the calculation plan. The initial transition to AURORA-B methods will include **[**

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4.0 ANALYSIS OF PLANT TRANSIENTS

Framatome's licensing methodology is based upon core conditions established by a detailed step-through calculation. In support of demonstrating the AURORA-B AOO method to Monticello, plant transients are analyzed for a small subset of power and flow conditions at a cycle exposure and scram speed discussed in each subsection. The transient analyses, presented in this section, are performed using plant parameters provided by the utility for a full core of ATRIUM 11 fuel.

The transient events chosen to demonstrate the application of the AURORA-B AOO method are typical limiting events for Monticello as determined from previous cycle analyses and a review of Chapter 14 of the updated safety analysis report (USAR), as shown in Section 3.1 of this report.

4.1 *Transient Events*

4.1.1 Load Rejection No Bypass (LRNB)

Load rejection causes a fast closure of the turbine control valves. The resulting compression wave travels through the steam lines into the vessel and creates a rapid pressurization. The increase in pressure causes a decrease in core voids, which in turn causes a rapid increase in power. Fast closure of the turbine control valves also causes a reactor scram. Turbine bypass system operation, which also mitigates the consequences of the event, is not credited. The excursion of the core power due to the void collapse is terminated primarily by the reactor scram and revoiding of the core.

To demonstrate the AURORA-B AOO transient methodology models the LRNB event appropriately, LRNB analyses were performed for the following range of conditions within the approved extended flow window (EFW) power/flow map:

- 100% core power, with 105% and 80% core flow
- 85% core power, with 105% core flow
- 60% core power, with 108.3% core flow

- 40% core power above Pbypass, with 111.1% core flow
- 40% core power below Pbypass, with 111.1% core flow

Table 4.1 presents the change in MCPR for the LRNB event. The transient analyses are performed at the end of full power (EOFP) cycle exposure, utilizing NSS insertion times. Table 4.2 presents the sequence of event timing for the LRNB event at 100% power with 105% core flow. Figure 4.1 – Figure 4.3 show the responses of various reactor and plant parameters during the LRNB event initiated at 100% of rated power and 105% of rated core flow with NSS insertion times.

4.1.2 **Turbine Trip No Bypass (TTNB)**

A turbine trip event can be initiated as a result of several different signals. The initiating signal causes the TSV to close in order to prevent damage to the turbine. The TSV closure creates a compression wave traveling through the steam lines into the vessel causing a rapid pressurization. The increase in pressure results in a decrease in core voids, which in turn causes a rapid increase in power. Closure of the TSV also causes a reactor scram which helps mitigate the pressurization effects. Turbine bypass system operation, which also mitigates the consequences of the event, is not credited. The excursion of the core power due to the void collapse is terminated primarily by the reactor scram and revoiding of the core.

To demonstrate the AURORA-B AOO transient methodology models the TTNB event appropriately, TTNB analyses were performed for the following range of conditions within the approved EFW power/flow map:

- 100% core power, with 105% and 80% core flow
- 85% core power, with 105% core flow
- 60% core power, with 108.3% core flow
- 40% core power above Pbypass, with 111.1% core flow
- 40% core power below Pbypass, with 111.1% core flow

Table 4.1 presents the change in MCPR for the TTNB event. The transient analyses are performed at the EOFP cycle exposure, utilizing NSS insertion times. Table 4.3 presents the sequence of event timing for the TTNB event at 100% power with 105% core flow. Figure 4.4 – Figure 4.6 show the responses of various reactor and plant parameters during the TTNB event initiated at 100% of rated power and 105% of rated core flow with NSS insertion times.

4.1.3 **Feedwater Controller Failure (FWCF)**

The increase in feedwater flow due to a failure of the feedwater control system to maximum demand results in an increase in the water level and a decrease in the coolant temperature at the core inlet. The increase in core inlet subcooling causes an increase in core power. As the feedwater flow continues at maximum demand, the water level continues to rise and eventually reaches the high water level trip setpoint. The initial water level is conservatively assumed to be at the low level normal operating range to delay the high-level trip and maximize the core inlet subcooling resulting from the FWCF. The high water level trip causes the TSVs to close in order to prevent damage to the turbine from excessive liquid inventory in the steam line. Valve closure creates a compression wave traveling back to the core, causing void collapse and subsequent rapid power excursion. The closure of the TSVs also initiates a reactor scram. The turbine bypass valves are assumed operable and provide some pressure relief. The core power excursion is mitigated in part by pressure relief, but the primary mechanisms for termination of the event are reactor scram and revoiding of the core.

To demonstrate the AURORA-B AOO transient methodology models the FWCF event appropriately, FWCF analyses were performed for the following range of conditions within the approved EFW power/flow map:

- 100% core power, with 105% and 80% core flow
- 85% core power, with 105% core flow
- 60% core power, with 108.3% core flow
- 40% core power above Pbypass, with 111.1% core flow

- 40% core power below Pbypass, with 111.1% core flow

Table 4.1 presents the change in MCPR for the FWCF event. The transient analyses are performed at the EOFP cycle exposure, utilizing NSS insertion times. Table 4.4 presents the sequence of event timing for the FWCF event at 100% power with 105% core flow. Figure 4.7 – Figure 4.9 show the responses of various reactor and plant parameters during the FWCF event initiated at 100% of rated power and 105% of rated core flow with NSS insertion times.

4.1.4 **Inadvertent HPCI Start-Up (HPCI)**

The inadvertent HPCI start-up results in the injection of cold water to the downcomer from the HPCI pump through the feedwater sparger. Injection of this subcooled water increases the subcooling at the inlet to the core and results in an increase in core power. The feedwater control system will attempt to control the water level in the reactor by reducing the feedwater flow. As long as the mass of steam leaving the reactor through the steam lines is more than the mass of HPCI water being injected, the water level will be controlled and a new steady-state condition will be established. In this situation, the event is similar to a loss of feedwater heating event. If the steam flow is less than the HPCI flow, the water level will increase until the high water level setpoint is reached. In this situation, the event is similar to a feedwater controller failure event.

The HPCI flow in Monticello is only injected into one of the two feedwater lines and thus through the feedwater spargers on only one side of the reactor vessel, resulting in an asymmetric flow distribution of the injected HPCI flow. The asymmetric injection of the HPCI flow may cause an asymmetric core inlet enthalpy distribution and a larger enthalpy decrease for part of the core. **[**

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To demonstrate the AURORA-B AOO transient methodology models the HPCI event appropriately, HPCI analyses were performed for the following range of conditions within the approved EFW power/flow map:

- 100% core power, with 105% and 80% core flow
- 85% core power, with 105% core flow
- 60% core power, with 108.3% core flow
- 40% core power, with 111.1% core flow

Table 4.1 presents the change in MCPR for the HPCI event. The rated power HPCI cases reached the high water level trip setpoint resulting in larger ΔMCPR values than the off-rated cases which did not reach the setpoint. Framatome will investigate the impact of plant parameters on this trend as part of the initial transition (Reference 4). The transient analyses are performed at the EOFP cycle exposure, utilizing NSS insertion times. Table 4.5 presents the sequence of event timing for the HPCI event at 100% power with 105% core flow. Figure 4.10 – Figure 4.12 show the responses of various reactor and plant parameters during the HPCI event initiated at 100% of rated power and 105% of rated core flow with NSS insertion times.

4.1.5 **Loss of Feedwater Heating (LFWH)**

The loss of feedwater heating event analysis supports an assumed 95.3°F decrease in the feedwater temperature. The temperature is assumed to decrease linearly over 39 seconds. The result is an increase in core inlet subcooling, which reduces voids, thereby increasing core power and shifting axial power distribution toward the bottom of the core. As a result of the axial power shift and increased core power, voids begin to build up in the bottom region of the core, acting as negative feedback to the increased subcooling effect. The negative feedback moderates the core power increase. Although there is a substantial increase in core thermal power during the event, the increase in steam flow is much less because a large part of the added power is used to overcome the increase in inlet subcooling. The increase in steam flow is accommodated by the pressure control system via the TCVs or the turbine bypass valves.

To demonstrate the AURORA-B AOO transient methodology models the LFWH event appropriately, LFWH analyses were performed for the following range of conditions within the approved EFW power/flow map:

- 100% core power, with 105% and 80% core flow
- 85% core power, with 105% core flow
- 60% core power, with 108.3% core flow
- 40% core power, with 111.1% core flow

Table 4.1 presents the change in MCPR for the LFWH event. The transient analyses are performed at the EOFP cycle exposure, utilizing NSS insertion times. Table 4.6 presents the sequence of event timing for the LFWH event at 100% power with 105% core flow. Figure 4.13 – Figure 4.15 show the responses of various reactor and plant parameters during the LFWH event initiated at 100% of rated power and 105% of rated core flow with NSS insertion times.

4.1.6 **Fast Flow Runup**

Several possibilities exist for causing an unplanned increase in core coolant flow resulting from a recirculation flow control system malfunction. Increasing recirculation flow results in an increase in core flow which causes an increase in power level and a shift in power towards the top of the core by reducing the void fraction in that region. If the flow increase is relatively rapid and of sufficient magnitude, the neutron flux could exceed the scram set point, and a scram would be initiated. In addition, in some instances, core power increase could result in steam flows higher than can be relieved by the pressure control system. In this case, the system will pressurize until either the high neutron flux or high pressure scram is reached. While Table 6-39 of the AURORA-B topical report (Reference 1) lists the fast flow runup as a non-pressurization event, this event will instead be run as both a pressurization and non-pressurization event for any case where pressurization may possibly occur, and the most limiting result will be chosen.

Various failures can occur which can result in a speed increase of both recirculation pumps or failure of one of the motor generator set speed controllers can result in a speed increase in one recirculation pump. The failure of recirculation flow control system, affecting both pumps, is provided with rate limits and therefore is considered a slow event. The failure of one of the motor generator speed controllers generally results in the most rapid rate of recirculation flow increase and this is referred to as fast flow runup.

To demonstrate the AURORA-B AOO transient methodology models the fast flow runup event appropriately, fast flow runup analyses were performed for the following range of conditions within the approved EFW power/flow map:

- 100% core power, with 80% core flow
- 85% core power, with 60.6% core flow
- 60% core power, with 60% core flow
- 40% core power, with 70% core flow

Table 4.1 presents the change in MCPR for the fast flow runup event. The transient analyses are performed at the EOFP cycle exposure, utilizing NSS insertion times. Table 4.7 presents the sequence of event timing for the fast flow runup event at 100% power with 80% core flow. Figure 4.16 – Figure 4.18 show the responses of various reactor and plant parameters during the fast flow runup event initiated at 100% of rated power and 80% of rated core flow with NSS insertion times.

4.1.7 **ASME Overpressurization Analysis**

This section describes the maximum overpressurization analyses performed to demonstrate compliance with the ASME Boiler and Pressure Vessel Code. The analysis shows that the safety/relief valves (SRV) at Monticello have sufficient capacity and performance to prevent the reactor vessel pressure from reaching the safety limit of 110% of the design pressure.

To demonstrate the applicability of the AURORA-B AOO (Reference 1) methodology for ASME overpressurization analyses, main steam isolation valve (MSIV), TSV, and TCV closure analysis was performed for 102% power and 105% flow and 102% power and 80% flow at the latest full power exposure in the cycle design. The valve closure results in a rapid pressurization of the core. The increase in pressure causes a decrease in void which in turn causes a rapid increase in power. The following assumptions were made in the analysis:

- The most critical active component (direct scram on valve closure) was assumed to fail. However, scram on high neutron flux and high dome pressure is available.
- Opening of the turbine bypass valves was not credited.
- 3 SRV out-of-service (SRVOOS).
- Opening the SRV at the relief setpoints was not credited (open at safety setpoint)
- SRV open at 1145 psig (approximately 3% drift over the Technical Specification opening setpoint).
- TSSS insertion times were used.
- The initial dome pressure was set at the maximum allowed 1040.0 psia.
- A fast MSIV closure time of 3.0 seconds was used.
- High-pressure recirculation pump trip (ATWS-RPT) was considered.

Results of the TSV closure overpressurization analysis (event with least margin to criteria) are presented in Table 4.8 and demonstrate that the maximum vessel pressure limit of 1375 psig and dome pressure limit of 1332 psig are not exceeded. Table 4.9 presents the sequence of event timing for the TSV closure event at 102% power with 105% core flow. Figure 4.19 – Figure 4.22 show the response of various reactor plant parameters during the TSV closure event.

4.1.8 **ATWS Overpressurization Analysis**

This section describes the analyses performed to demonstrate that the peak vessel pressure for the limiting ATWS event is less than the ASME Service Level C limit of 120% of the design pressure (1500 psig). To demonstrate the applicability of the

AURORA-B AOO (Reference 1) methodology for ATWS overpressurization analyses, the ATWS event analyses were performed at 100% power at 80% and 105% flow at the beginning of cycle (BOC) exposure based on historically limiting analyses. The MSIV closure and pressure regulator failure open (PRFO) events were evaluated. Failure of the pressure regulator in the open position causes the turbine control and turbine bypass valves to open such that steam flow increases until the maximum combined steam flow limit is attained. The system pressure decreases until the low steam line pressure setpoint is reached, resulting in the closure of the MSIVs. The resulting pressurization wave causes a decrease in core voids and an increase in core pressure thereby increasing the core power.

The following assumptions were made in the analyses:

- The analytical limit ATWS-RPT setpoint and function were assumed.
- 1 SRVOOS and the remaining 7 SRV open at safety setpoint.
- SRV open at 1145 psig (approximately 3% drift over the Technical Specification opening setpoint).
- All scram functions were disabled.
- Nominal values were used for initial dome pressure and feedwater temperature.
- The MSIV closure is based on a nominal closure time of 4.0 seconds for both events.

Results of the ATWS MSIV closure overpressurization analyses (event with least margin to criteria) are presented in Table 4.8 and demonstrate that the ATWS maximum vessel pressure limit of 1500 psig is not exceeded. Table 4.10 presents the sequence of event timing for the ATWS MSIV closure event at 100% power with 80% core flow. Figure 4.23 – Figure 4.26 show the response of various reactor plant parameters during the ATWS MSIV closure event, the event which results in the maximum vessel pressure.

Table 4.1 Base Case Transient Results

State Point Power / Flow (% of rated)	ATRIUM 11 ΔMCPR
Load Rejection No Bypass	
100 / 105	[ ]
100 / 80	[ ]
85 / 105	[ ]
60 / 108.3	[ ]
40 / 111.1	[ ]
40 / 111.1 below Pbypass	[ ]
Turbine Trip No Bypass	
100 / 105	[ ]
100 / 80	[ ]
85 / 105	[ ]
60 / 108.3	[ ]
40 / 111.1	[ ]
40 / 111.1 below Pbypass	[ ]
Feedwater Controller Failure	
100 / 105	[ ]
100 / 80	[ ]
85 / 105	[ ]
60 / 108.3	[ ]
40 / 111.1	[ ]
40 / 111.1 below Pbypass	[ ]

Table 4.1 Base Case Transient Results *(Continued)*

State Point Power / Flow (% of rated)	ATRIUM 11 ΔMCPR
Inadvertent HPCI Start-Up	
100 / 105	[ ]
100 / 80	[ ]
85 / 105	[ ]
60 / 108.3	[ ]
40 / 111.1	[ ]
Loss of Feedwater Heating	
100 / 105	[ ]
100 / 80	[ ]
85 / 105	[ ]
60 / 108.3	[ ]
40 / 111.1	[ ]
Fast Flow Runup	
100 / 80	[ ]
85 / 60.6	[ ]
60 / 60	[ ]
40 / 70	[ ]

Table 4.2 Sequence of Events Timing for the LRNB Event

Event	Time (sec)
TCV Closure Event	0.000
TCV Fast Closure Scram Signal	0.045
Reactor Scram	0.045
Time of TCV Full Closure	0.350
Peak Power	0.710
Peak Heat Flux	0.805
Time of critical heat flux	0.815
SRV Actuation	1.889
Peak Dome Pressure (1232.15 psia)	1.985
Peak Vessel Pressure (1269.74 psia)	2.010
Peak Steam Line Pressure (1239.57 psia)	2.373

Table 4.3 Sequence of Events Timing for the TTNB Event

Event	Time (sec)
TSV Closure Event	0.000
TSV Position Scram Signal	0.035
Reactor Scram	0.035
Time of TSV Full Closure	0.100
Peak Power	0.580
Time of critical heat flux	0.655
Peak Heat Flux	0.660
SRV Actuation	1.740
Peak Dome Pressure (1238.52 psia)	1.845
Peak Vessel Pressure (1276.63 psia)	1.870
Peak Steam Line Pressure (1248.07 psia)	2.220

Table 4.4 Sequence of Events Timing for the FWCF Event

Event	Time (sec)
FWCF Event Initiator	0.000
Level 8 – High Water Level – Trip	34.52
Level 8 – TSV Closure Signal	35.63
TSV Motion Scram Signal	35.66
Reactor Scram	35.66
Turbine Bypass Valves Open	35.76
Peak Power	36.19
Time of critical heat flux	36.27
Peak Heat Flux	36.28
SRV Actuation	37.48
Peak Dome Pressure (1216.30 psia)	37.56
Peak Steam Line Pressure (1221.54 psia)	37.58
Peak Vessel Pressure (1253.19 psia)	37.59

Table 4.5 Sequence of Events Timing for the HPCI Event

Event	Time (sec)
HPCI Event Initiator	0.000
Level 8 – High Water Level – Trip	50.34
Level 8 – TSV Closure Signal	51.44
TSV Motion Scram Signal	51.48
Reactor Scram	51.48
Turbine Bypass Valves Open	51.57
Peak Power	52.00
Peak Heat Flux	52.08
Time of critical heat flux	52.08
SRV Actuation	53.25
Peak Dome Pressure (1217.40 psia)	53.33
Peak Steam Line Pressure (1219.18 psia)	53.34
Peak Vessel Pressure (1255.62 psia)	53.35

Table 4.6 Sequence of Events Timing for the LFWH Event

Event	Time (sec)
LFWH Event Initiator	0.000
Level 8 – High Water Level – Trip	NA
Reactor Scram	NA
Turbine Bypass Valves Open	NA
Time of critical heat flux	143.225
Peak Power	148.280
Peak Heat Flux	160.575
Peak Dome Pressure (1028.93 psia)	192.465
Peak Steam Line Pressure (1020.00 psia)	192.470
Peak Vessel Pressure (1069.59 psia)	203.700

Table 4.7 Sequence of Events Timing for the Fast Flow Runup Event

Event	Time (sec)
Fast Flow Runup Event Initiator	0.00
Level 8 – High Water Level – Trip	NA
Reactor Scram	NA
Turbine Bypass Valves Open	7.35
Peak Power	15.26
Peak Heat Flux	15.56
Time of critical heat flux	15.72
Peak Steam Line Pressure (1033.63 psia)	17.18
Peak Vessel Pressure (1079.43 psia)	17.24
Peak Dome Pressure (1044.14 psia)	17.27

Table 4.8 ASME and ATWS Overpressurization Analysis Results

Event	Maximum Vessel Pressure Lower Plenum (psig)	Maximum Dome Pressure (psig)
ASME Overpressurization		
TSV closure (102P/105F)	1335	1298
TSV closure (102P/80F)	1320	1294
ATWS Overpressurization		
MSIV closure (100P/105F)	1329	1310
MSIV closure (100P/80F)	1352	1337

Table 4.9 Sequence of Events Timing for the ASME Overpressurization Event

Event	Time (sec)
TSV Closure Event Initiator	0.000
High Neutron Flux Setpoint	0.355
Reactor Scram	0.400
Peak Power	0.575
Peak Heat Flux	1.405
SRV Actuation	1.605
Recirculation Pump Trip	NA
Peak Steam Line Pressure (1318.09 psia)	2.725
Peak Dome Pressure (1312.44 psia)	3.039
Peak Vessel Pressure (1347.88 psia)	3.044

Table 4.10 Sequence of Events Timing for the ATWS Overpressurization Event

Event	Time (sec)
MSIV Closure Event Initiator (4.0 sec full closure time)	0.000
Recirculation Pump Trip Setpoint – High Pressure	4.230
SRV Actuation	4.650
Recirculation Pump Trip – High Pressure	4.735
Peak Power	4.755
Peak Heat Flux	4.875
Peak Vessel Pressure (1366.33 psia)	9.590
Peak Steam Line Pressure (1346.91 psia)	9.725
Peak Dome Pressure (1351.05 psia)	9.775

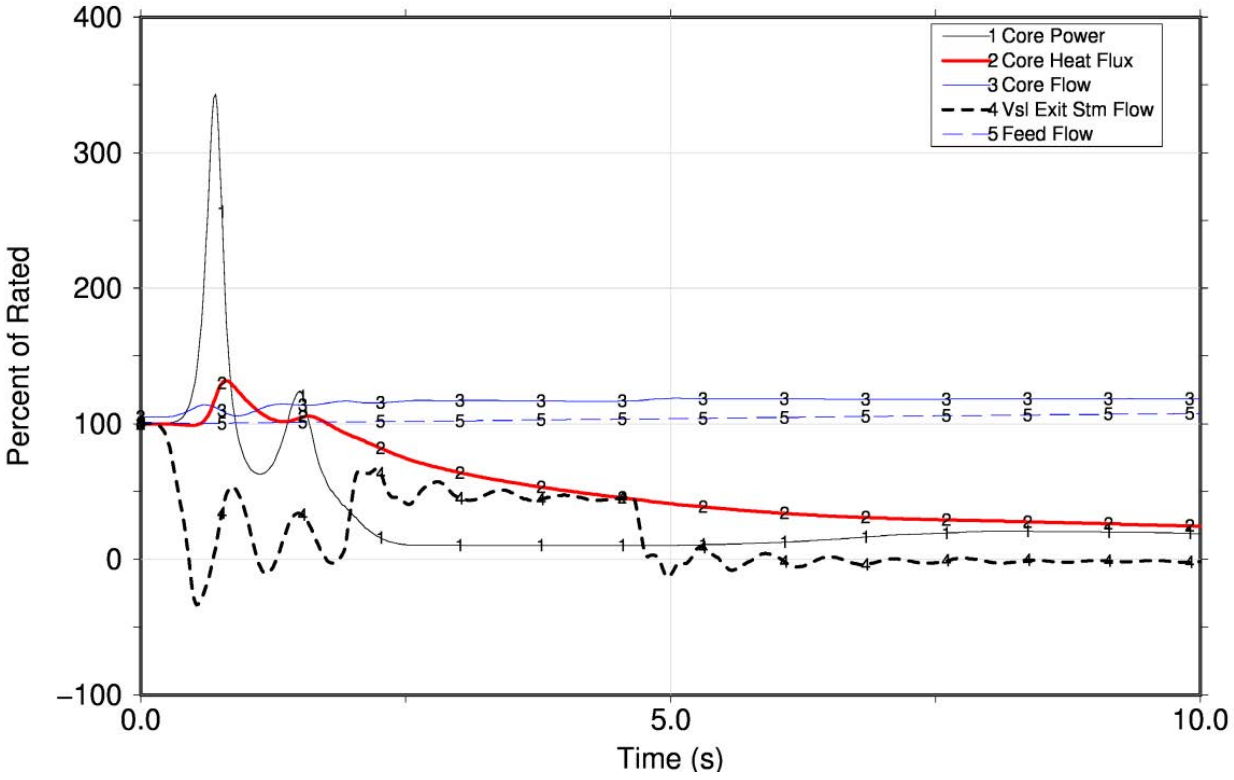


Figure 4.1 EOFP LRNB at 100P/105F – NSS Key Parameters

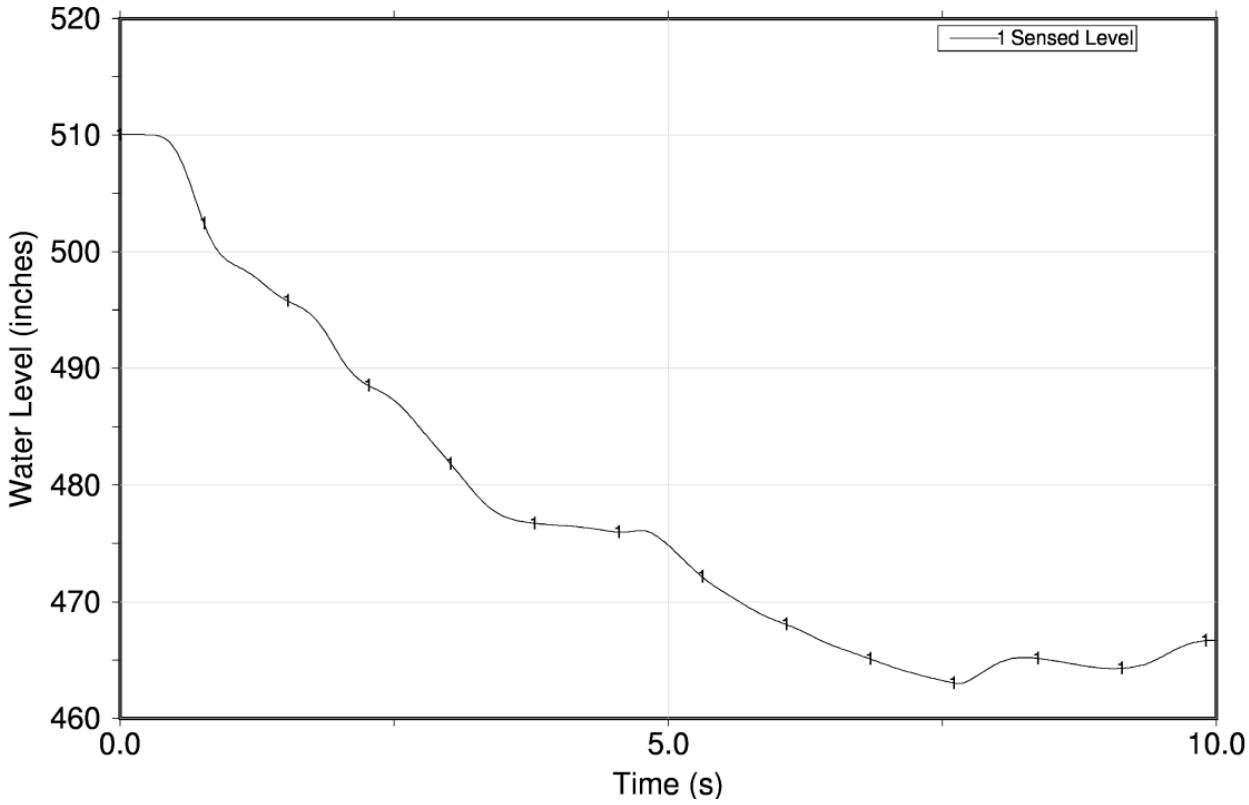


Figure 4.2 EOFP LRNB at 100P/105F – NSS Sensed Water Level

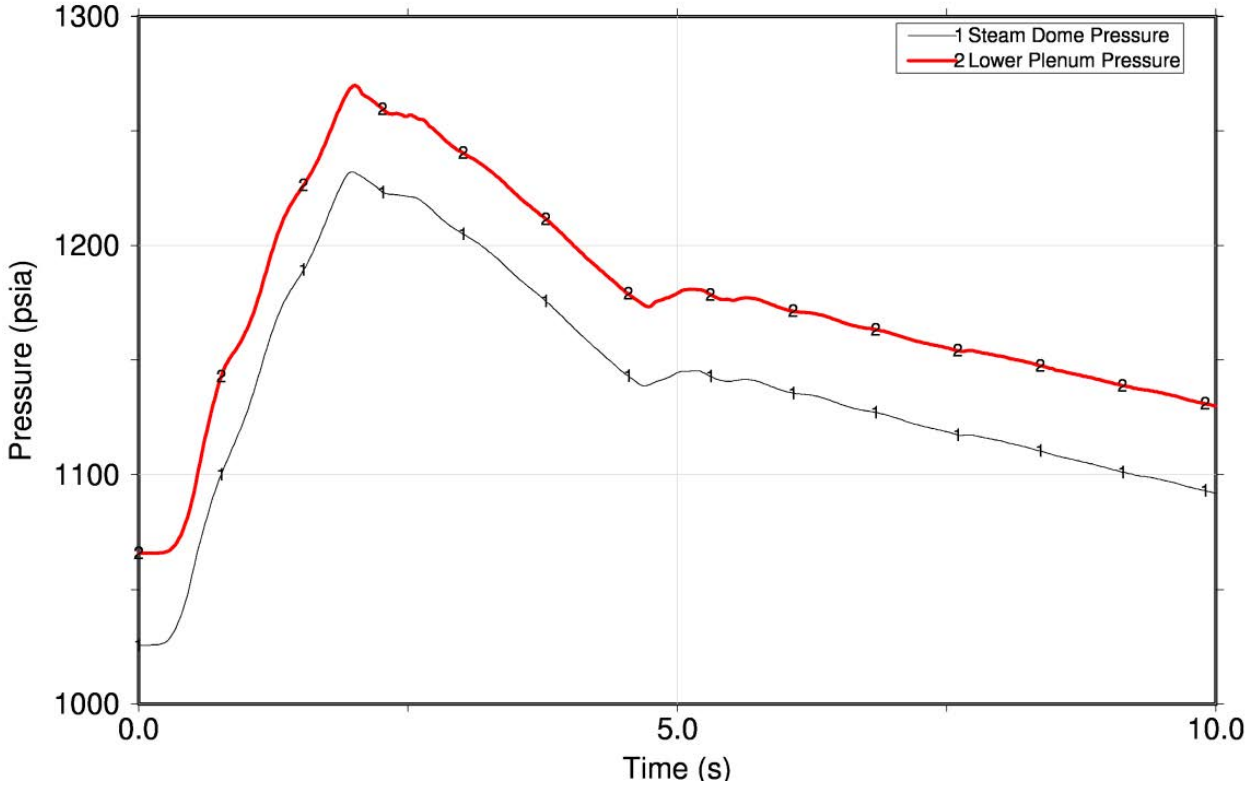


Figure 4.3 EOFP LRNB at 100P/105F – NSS Vessel Pressures

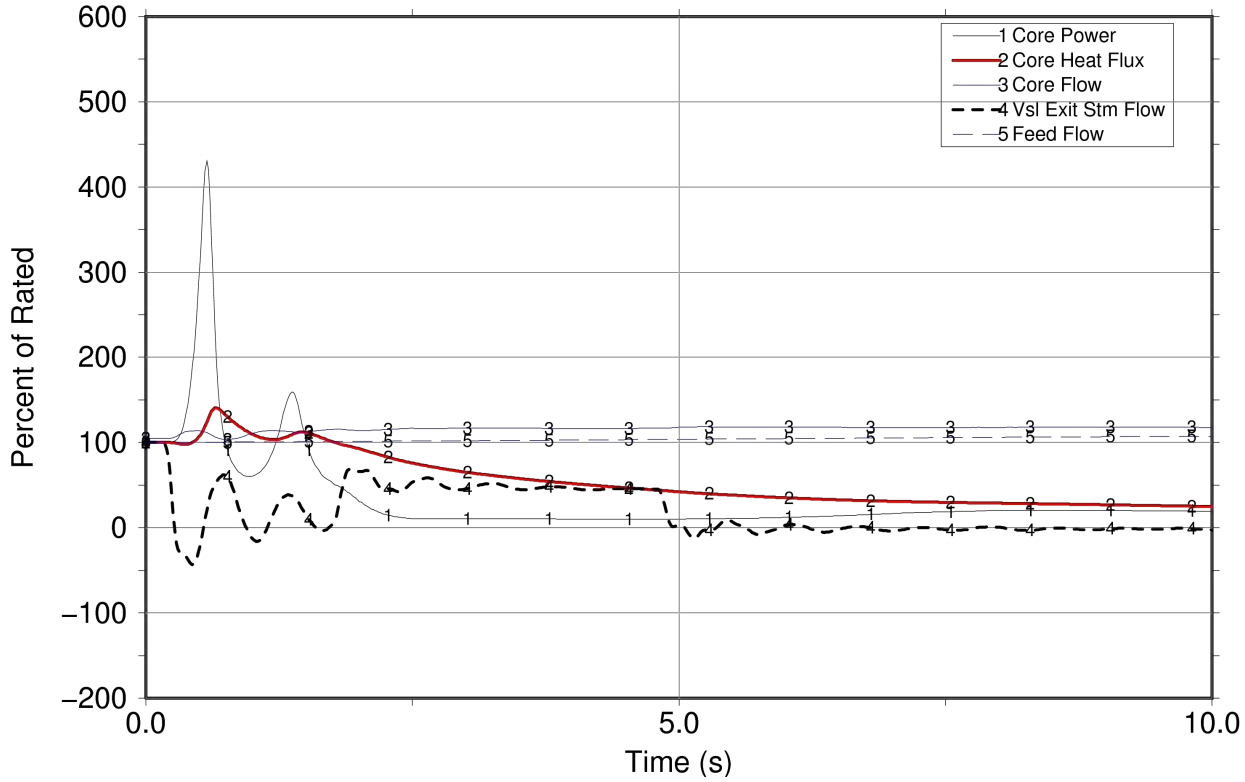


Figure 4.4 EOFP TTNB at 100P/105F – NSS Key Parameters

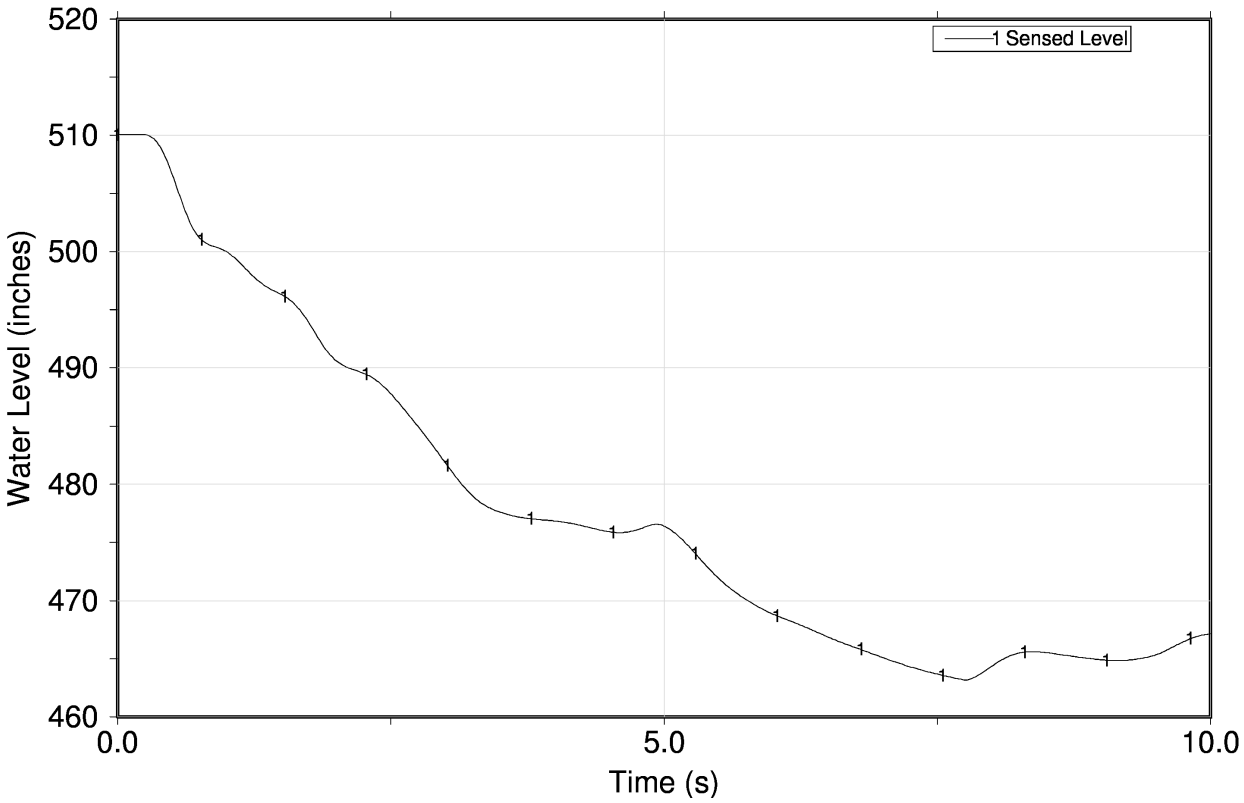


Figure 4.5 EOFP TTNB at 100P/105F – NSS Sensed Water Level

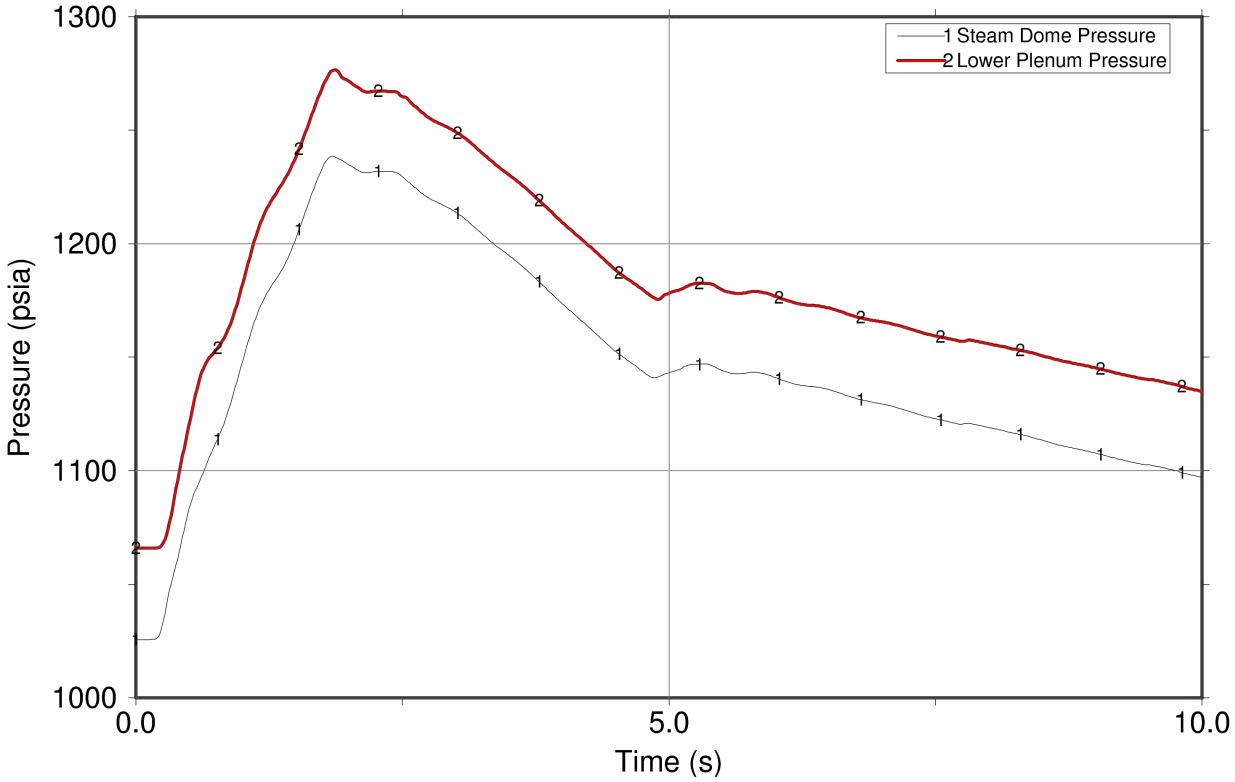


Figure 4.6 EOFP TTNB at 100P/105F – NSS Vessel Pressures

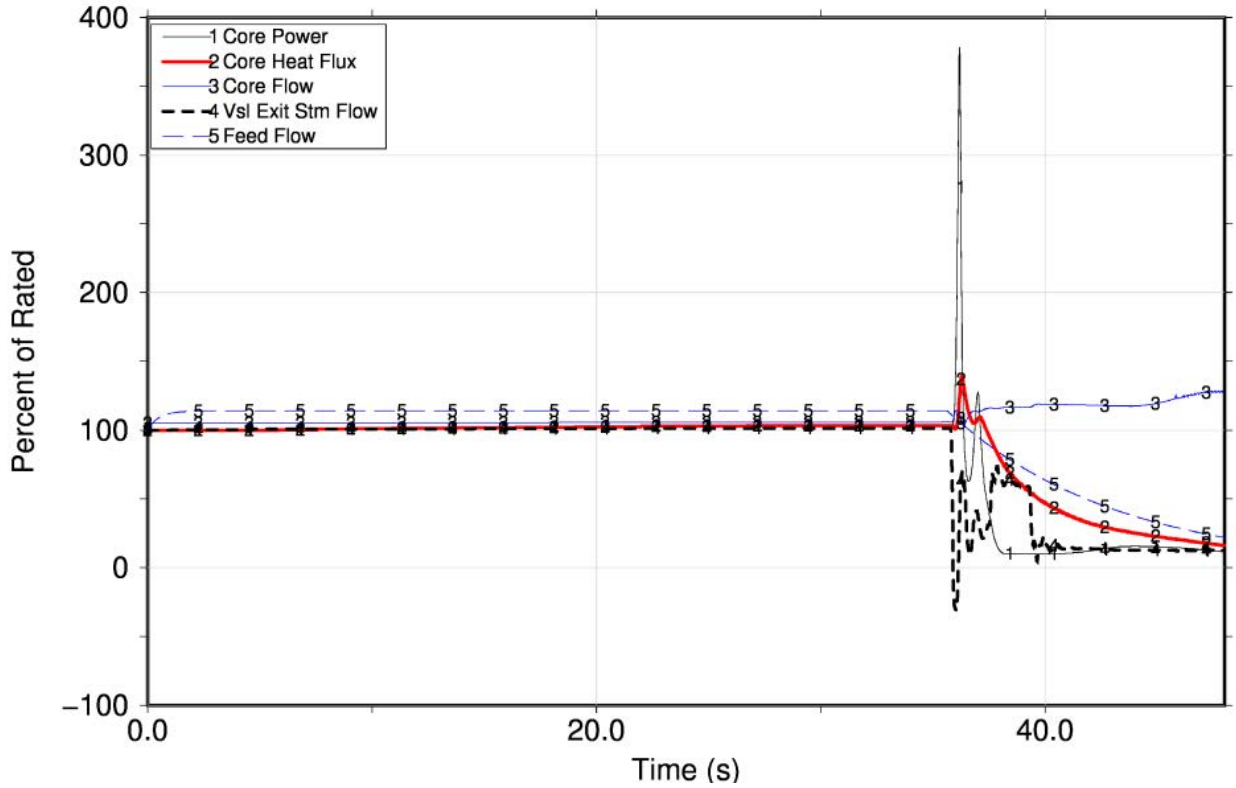


Figure 4.7 EOFP FWCF at 100P/105F – NSS Key Parameters

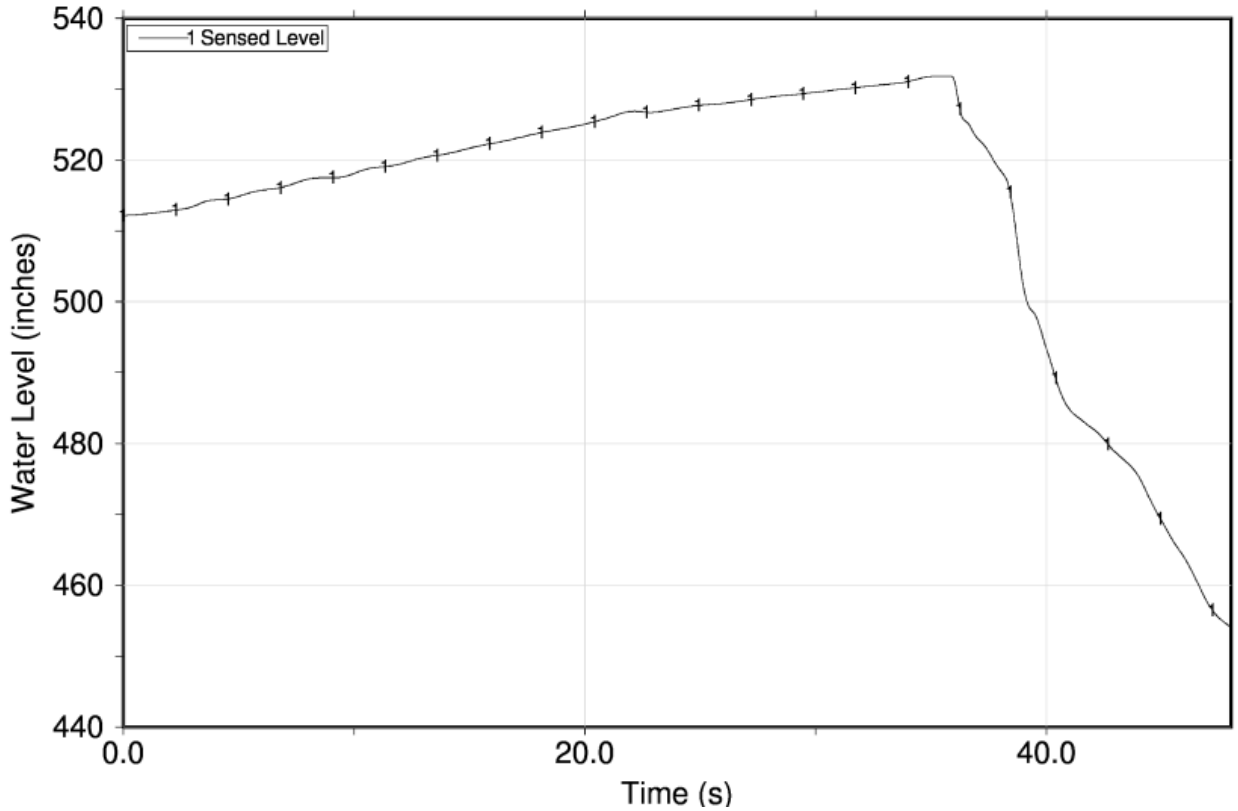


Figure 4.8 EOFP FWCF at 100P/105F – NSS Sensed Water Level

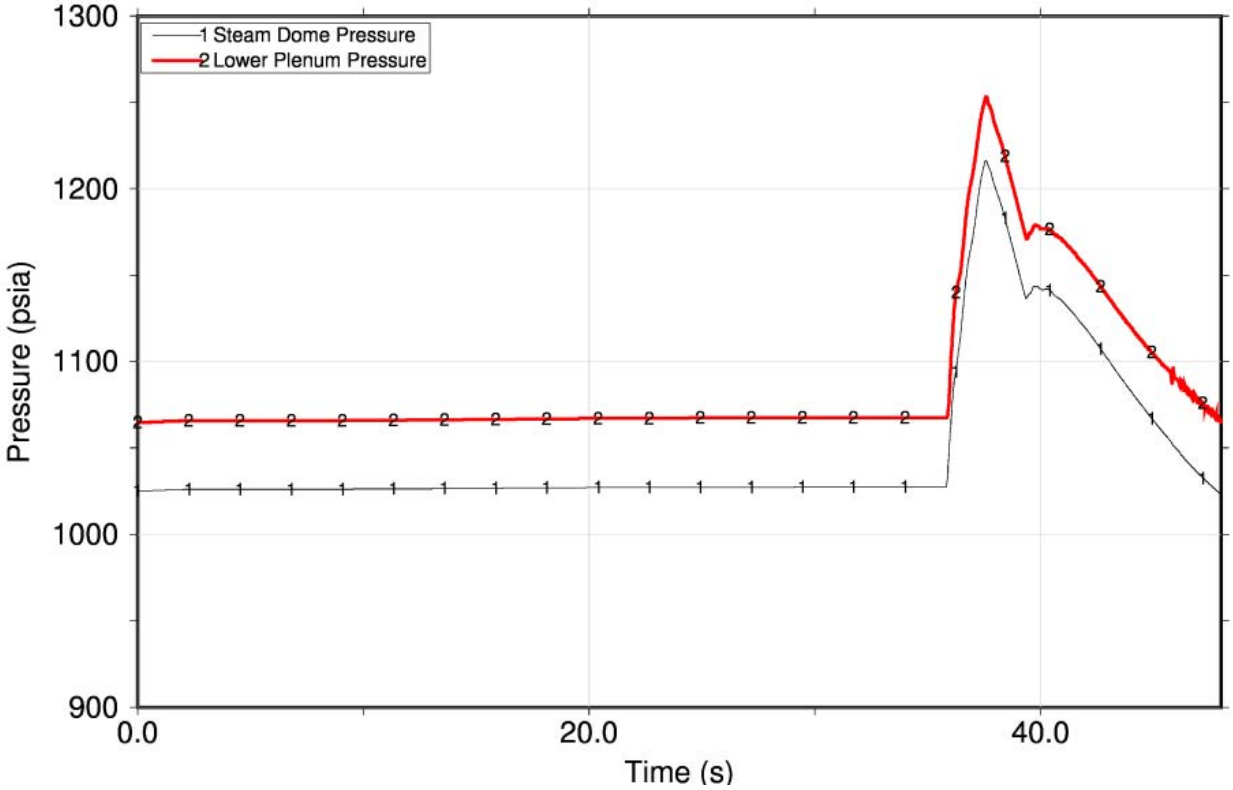


Figure 4.9 EOFP FWCF at 100P/105F – NSS Vessel Pressures

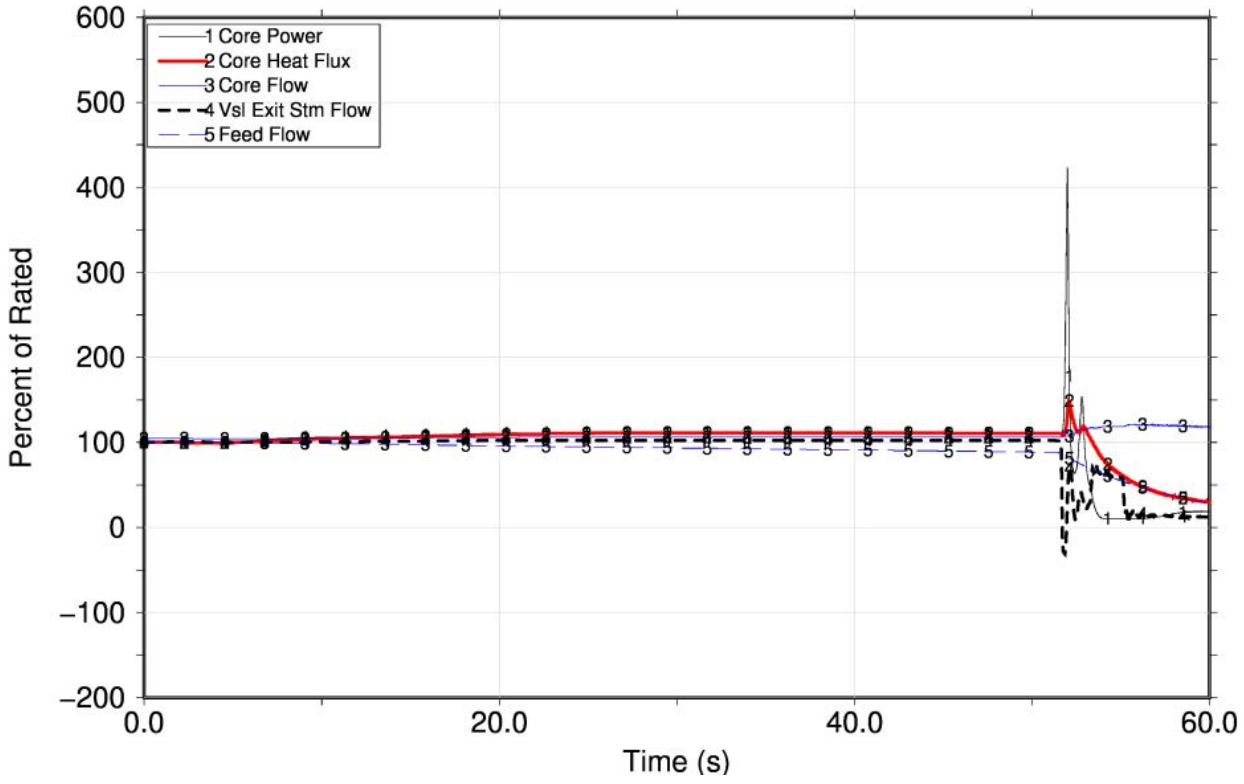


Figure 4.10 EOFP HPCI at 100P/105F – NSS Key Parameters

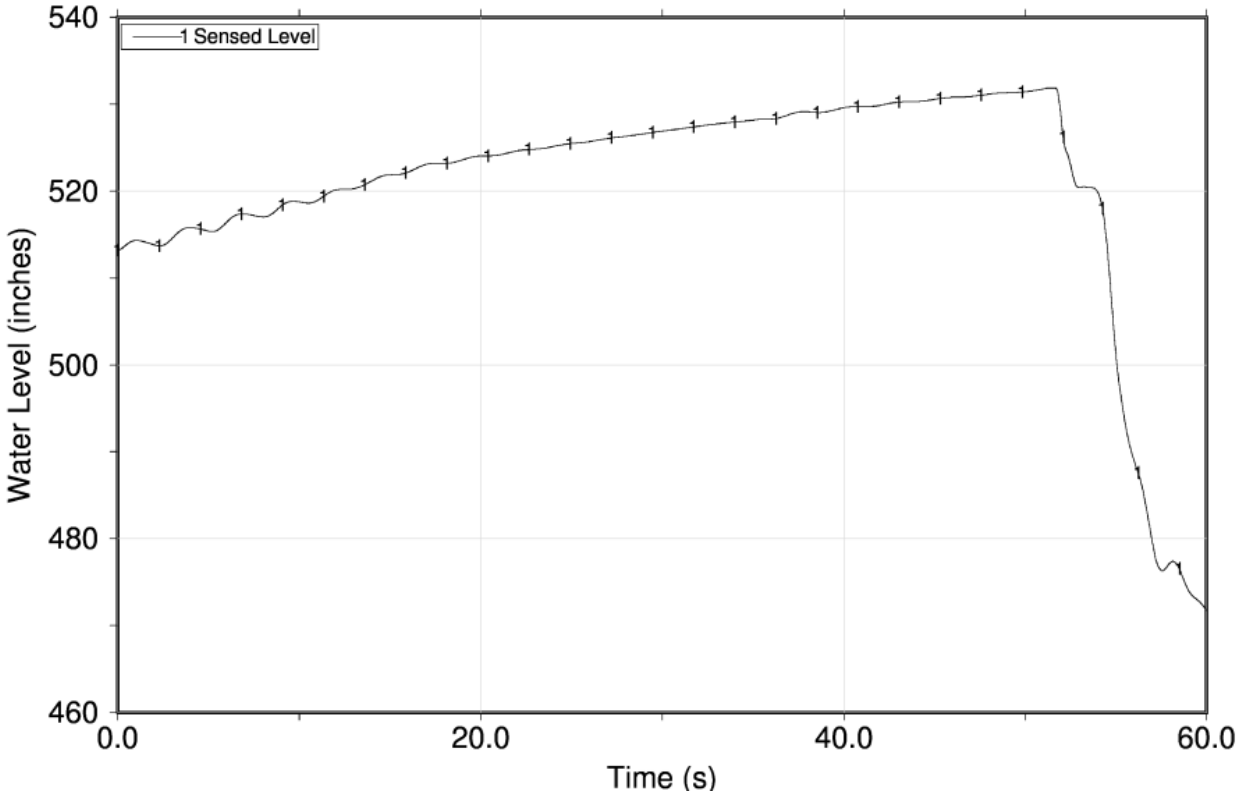


Figure 4.11 EOFP HPCI at 100P/105F – NSS Sensed Water Level

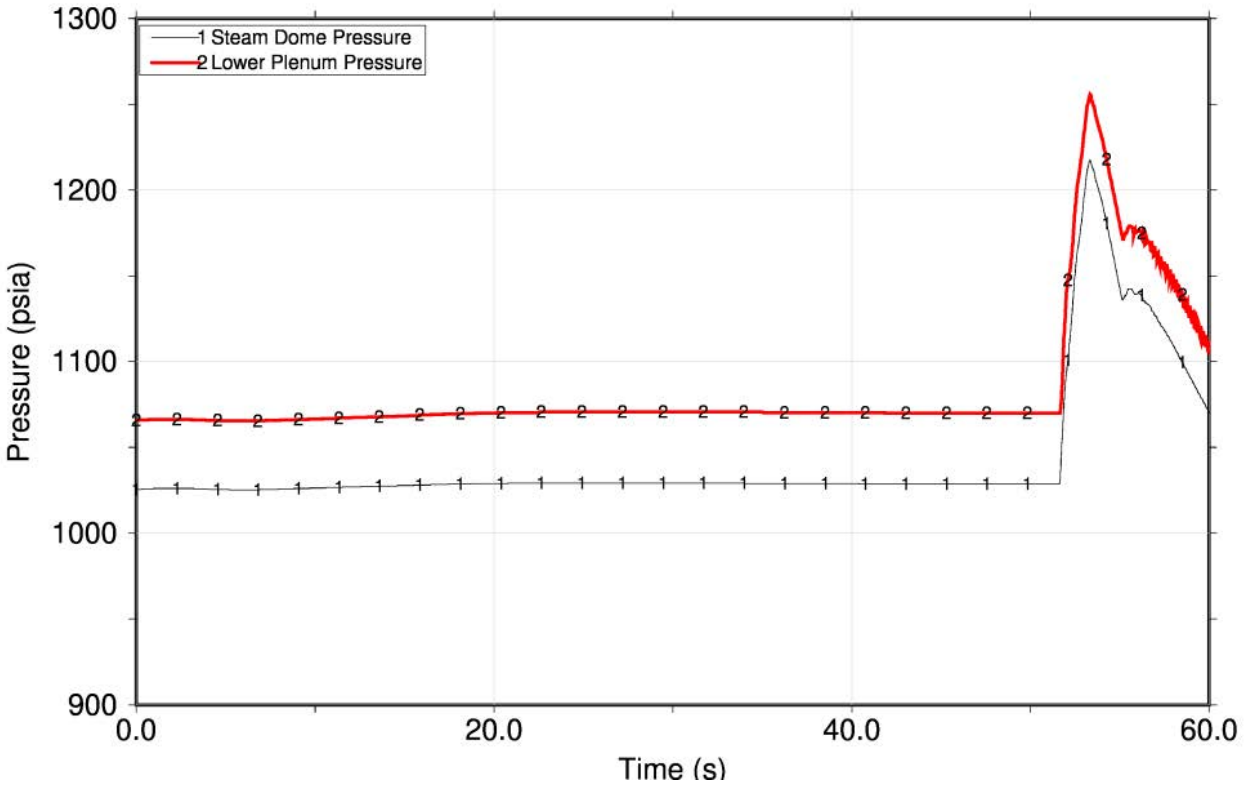


Figure 4.12 EOFP HPCI at 100P/105F – NSS Vessel Pressures

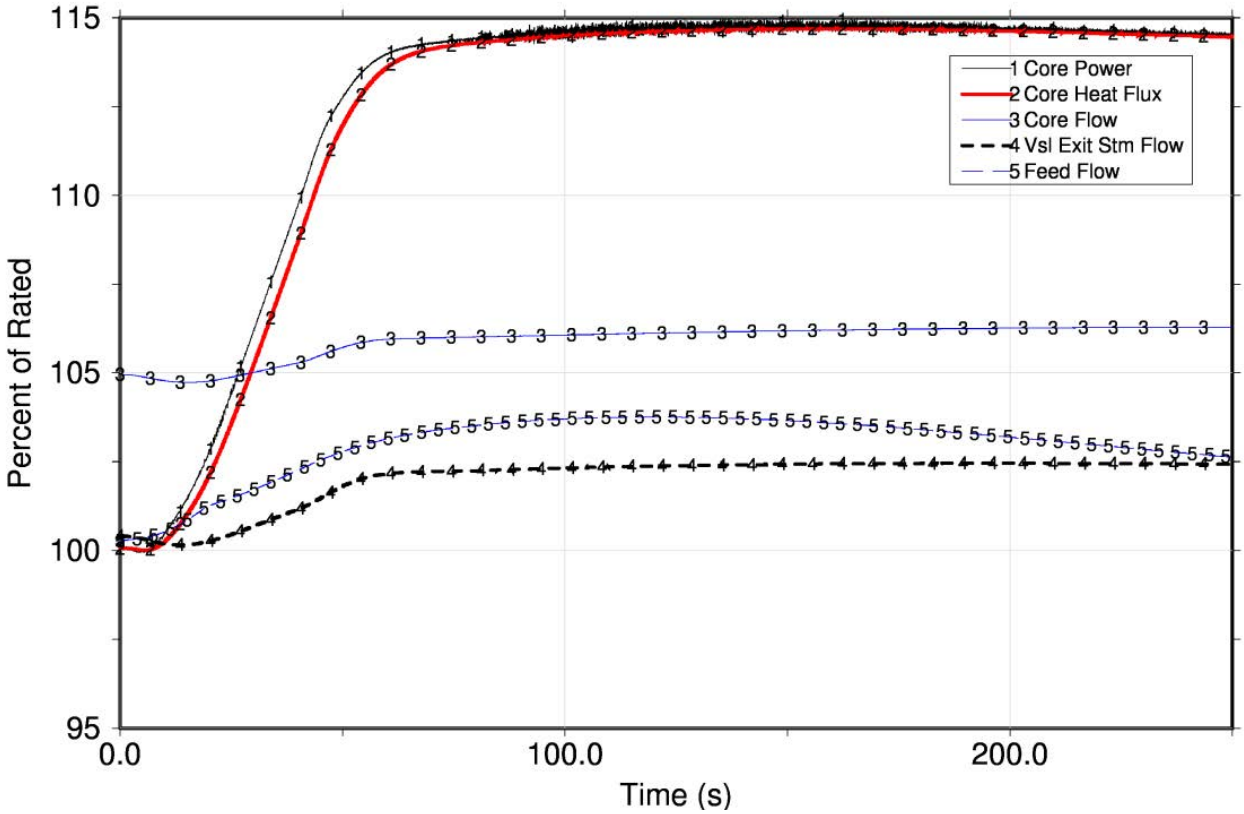


Figure 4.13 EOFP LFWH at 100P/105F – NSS Key Parameters

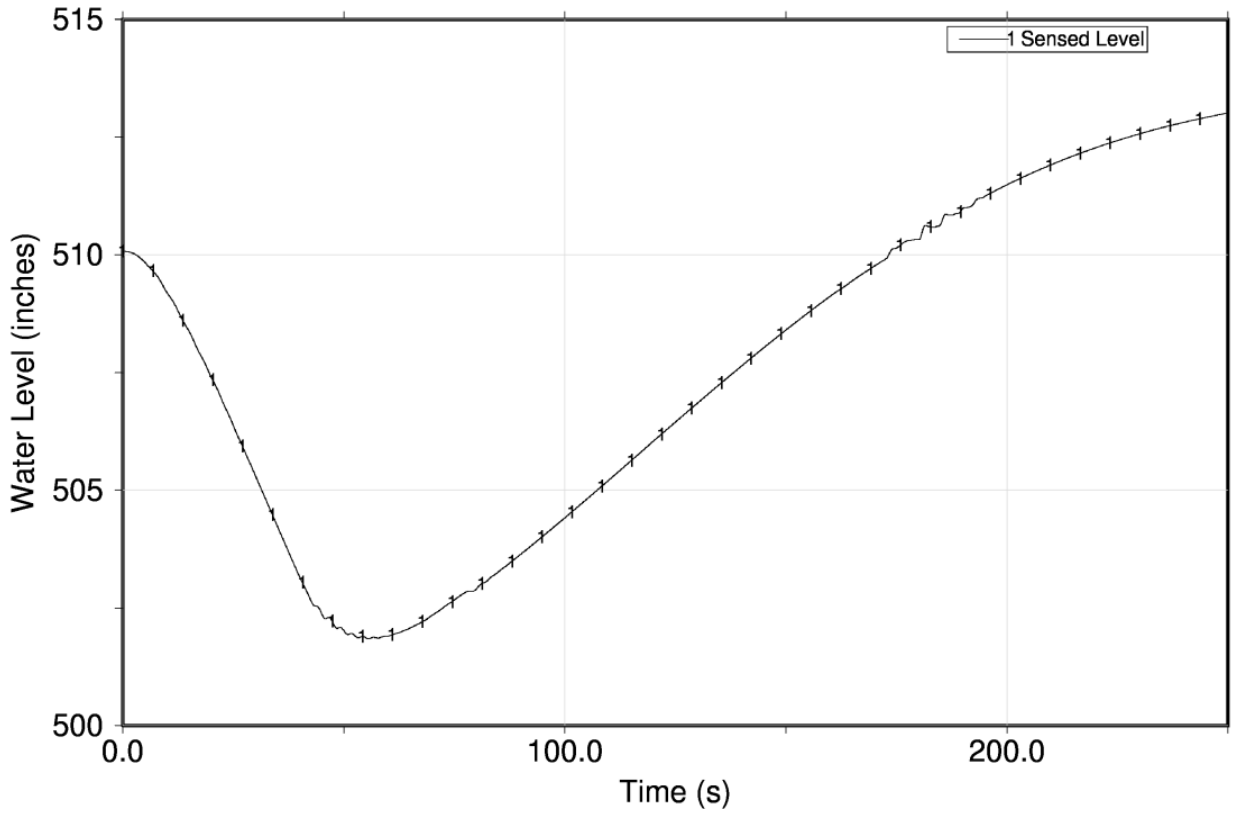


Figure 4.14 EOFP LFWH at 100P/105F – NSS Sensed Water Level

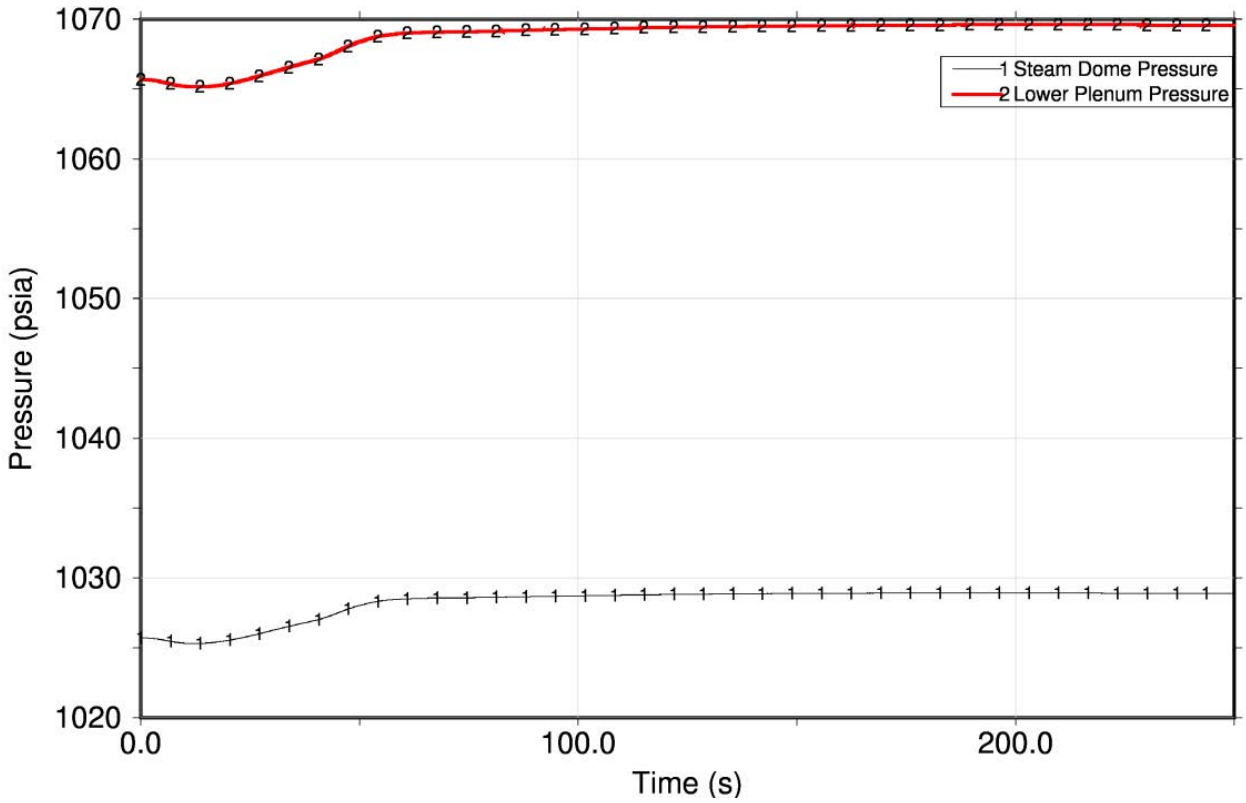


Figure 4.15 EOFP LFWH at 100P/105F – NSS Vessel Pressures

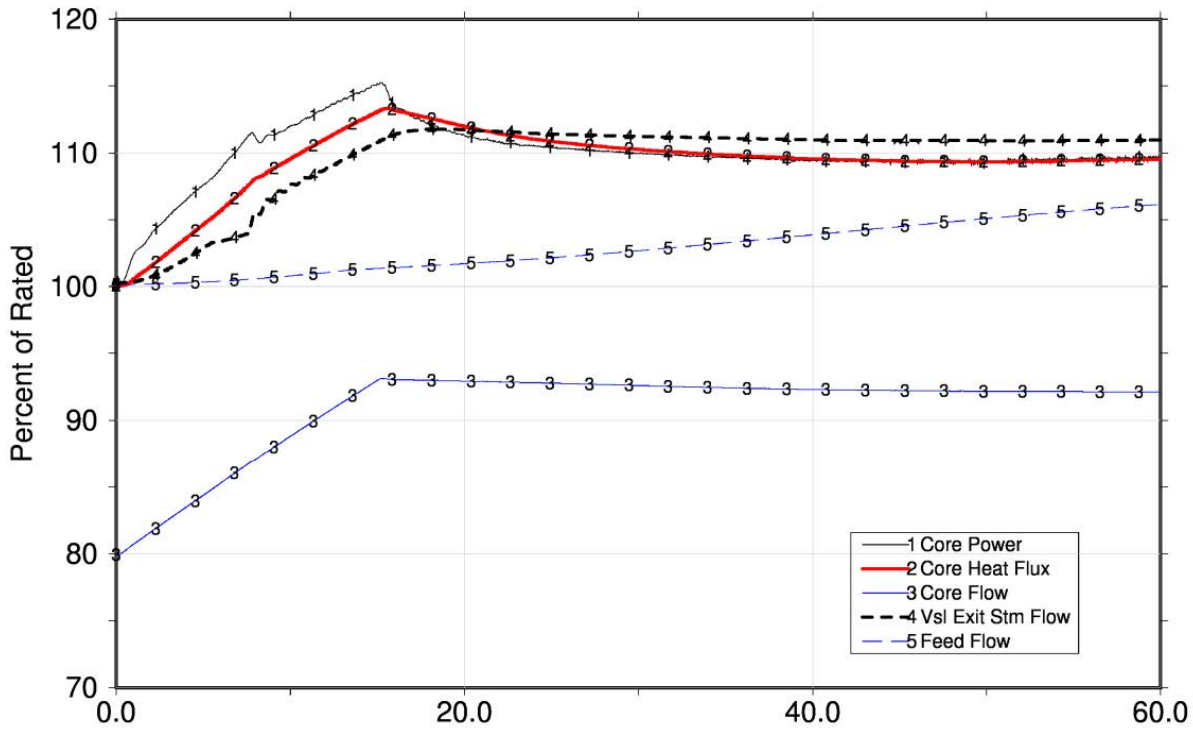


Figure 4.16 EOFP Fast Flow Runup at 100P/80F – NSS Key Parameters

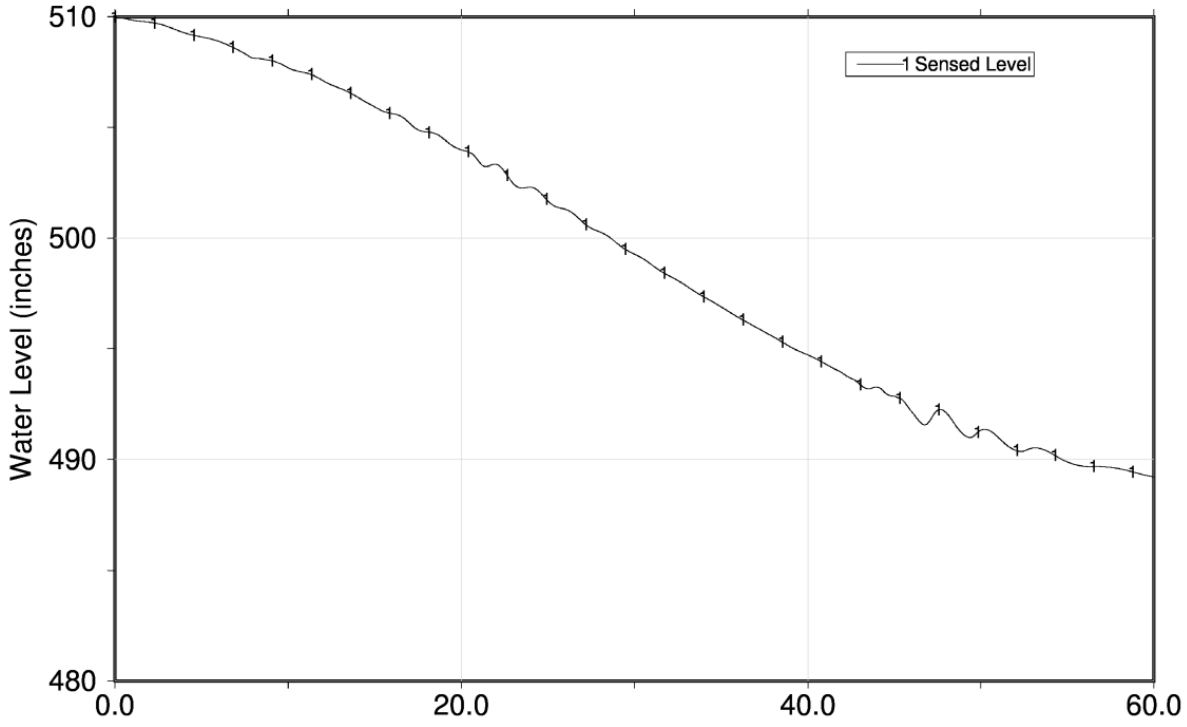


Figure 4.17 EOFP Fast Flow Runup at 100P/80F – NSS Sensed Water Level

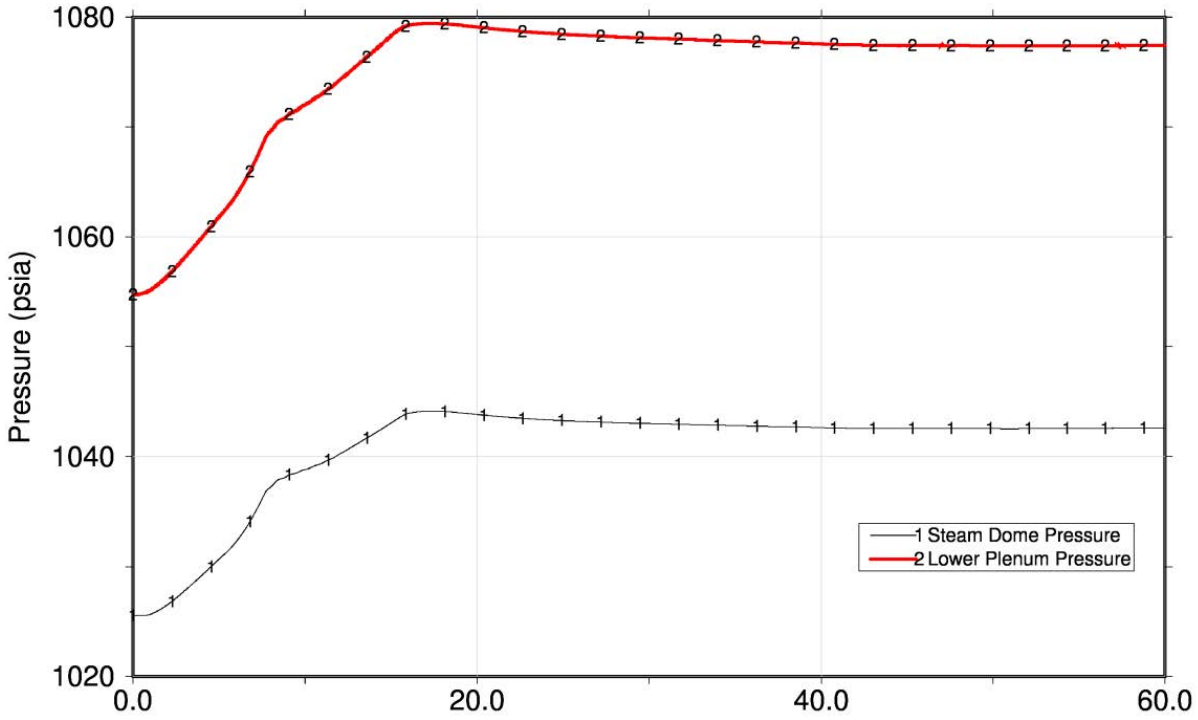


Figure 4.18 EOFP Fast Flow Runup at 100P/80F – NSS Vessel Pressures

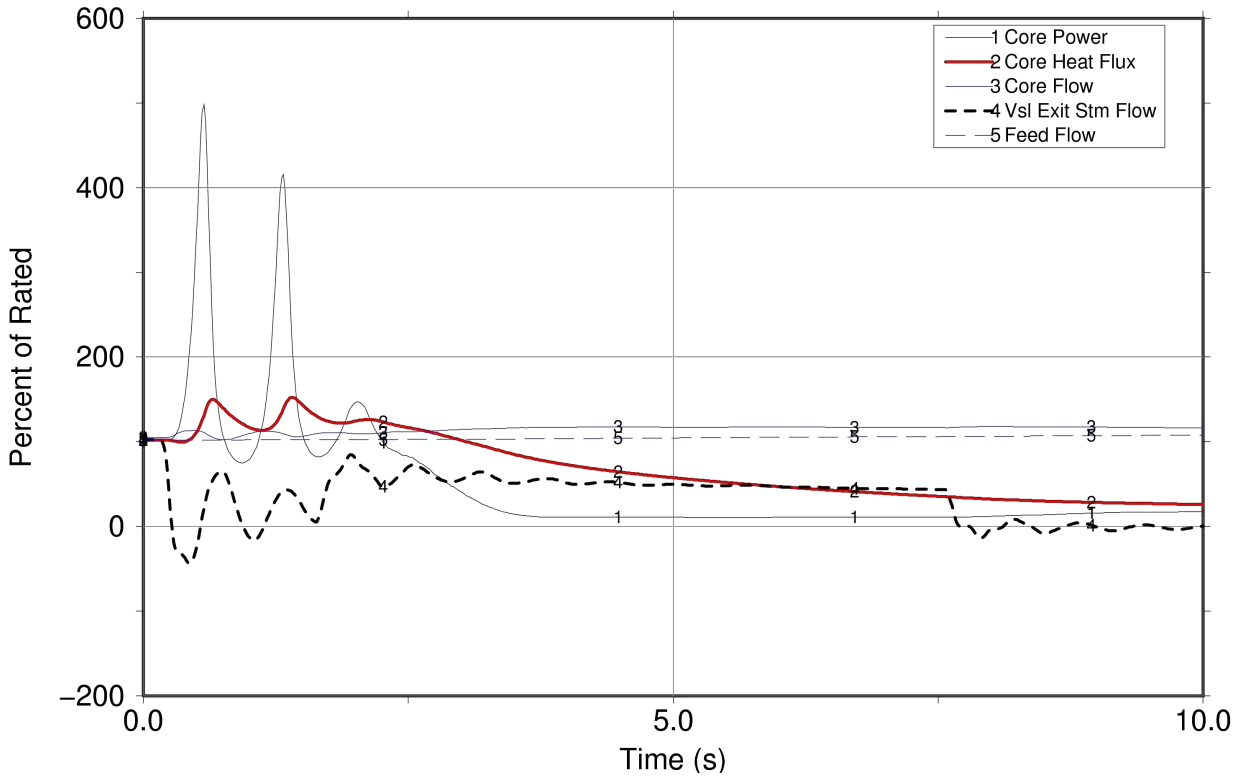


Figure 4.19 TSV Overpressurization Event at 102P/105F – Key Parameters

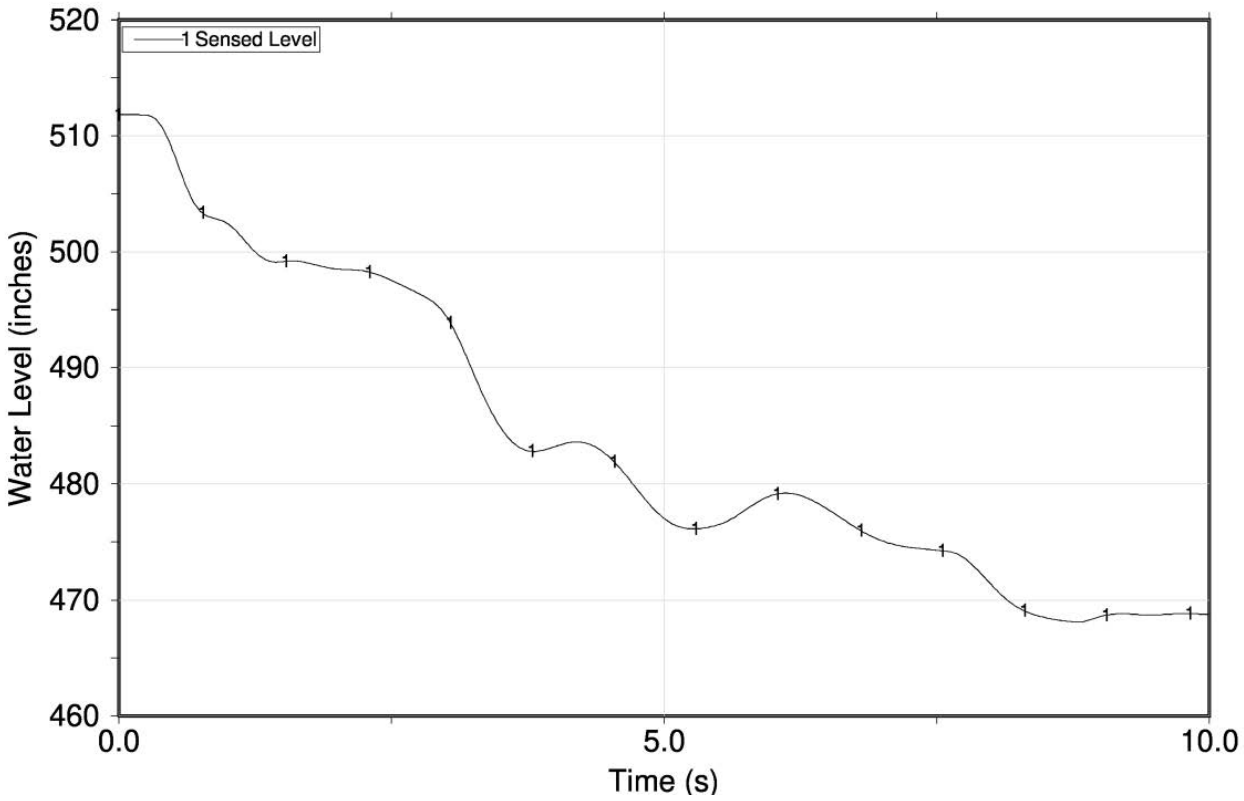


Figure 4.20 TSV Overpressurization Event at 102P/105F – Sensed Water Level

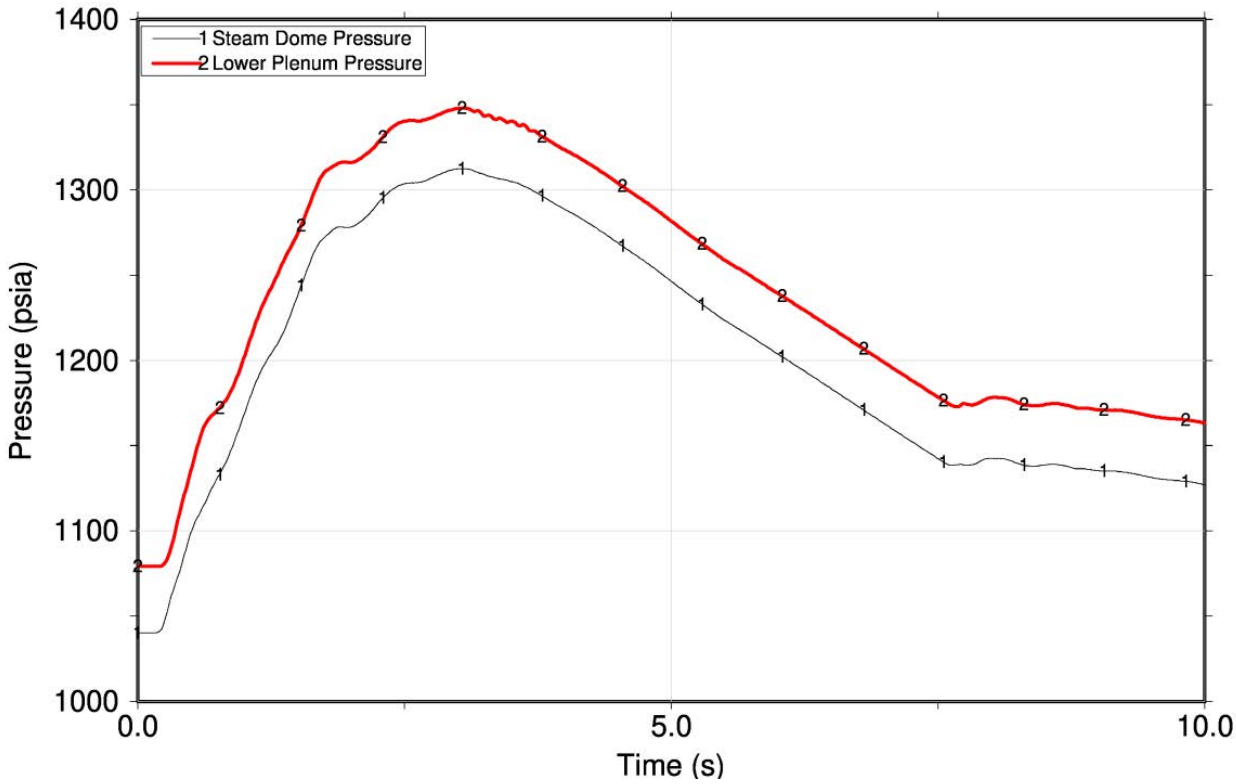


Figure 4.21 TSV Overpressurization Event at 102P/105F – Vessel Pressures

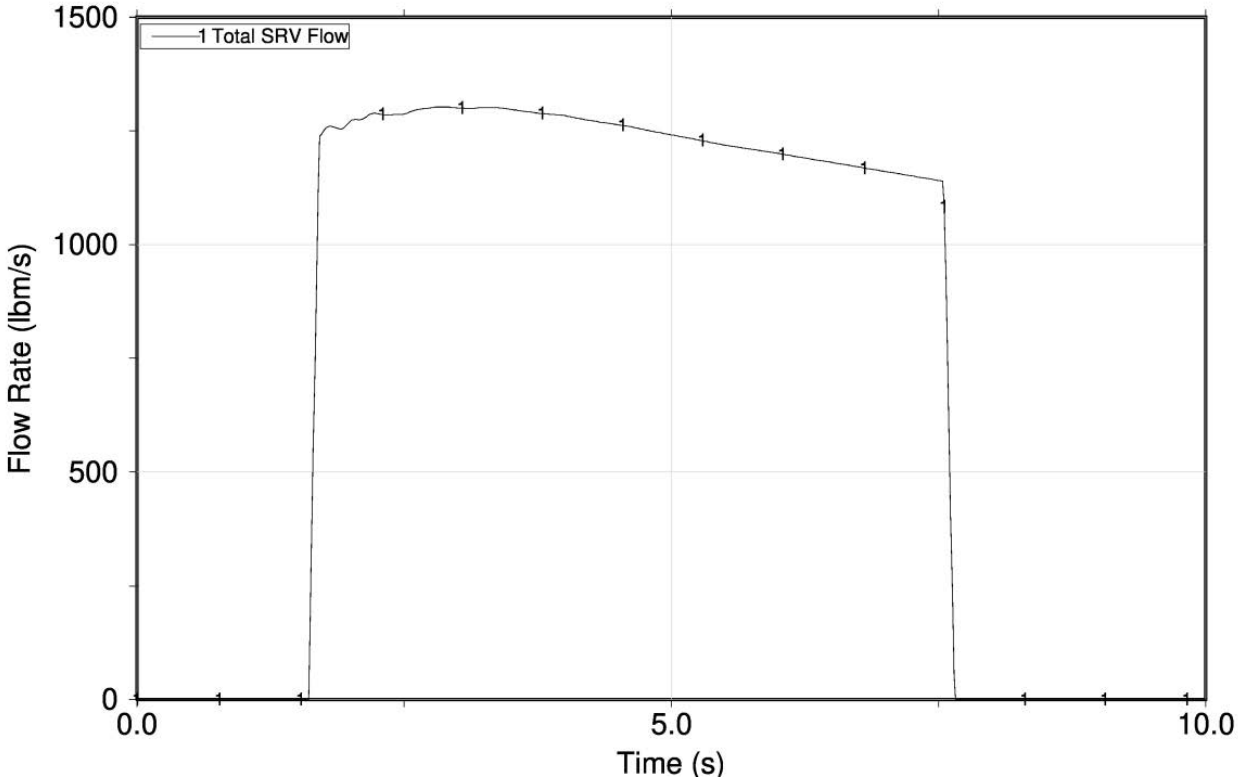


Figure 4.22 TSV Overpressurization Event at 102P/105F – Safety/Relief Valve Flow Rates

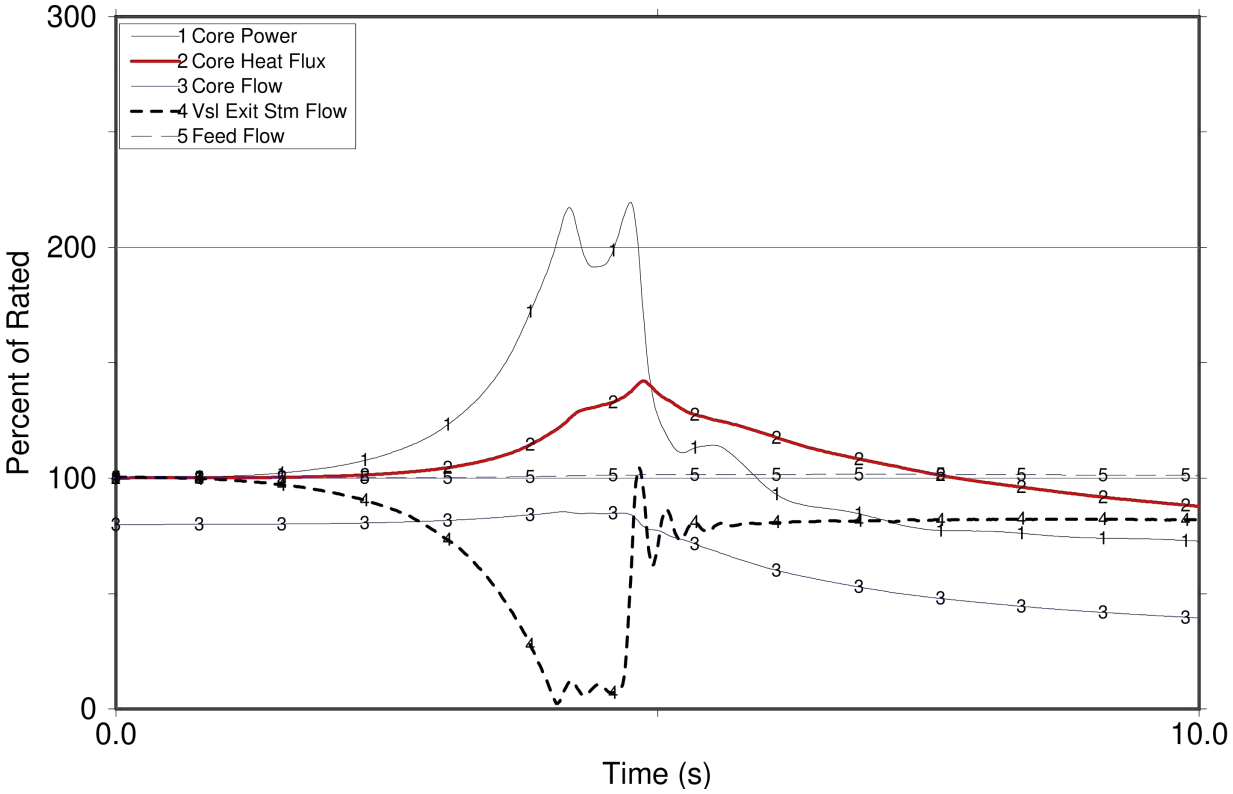


Figure 4.23 MSIV ATWS Overpressurization Event at 100P/80F – Key Parameters

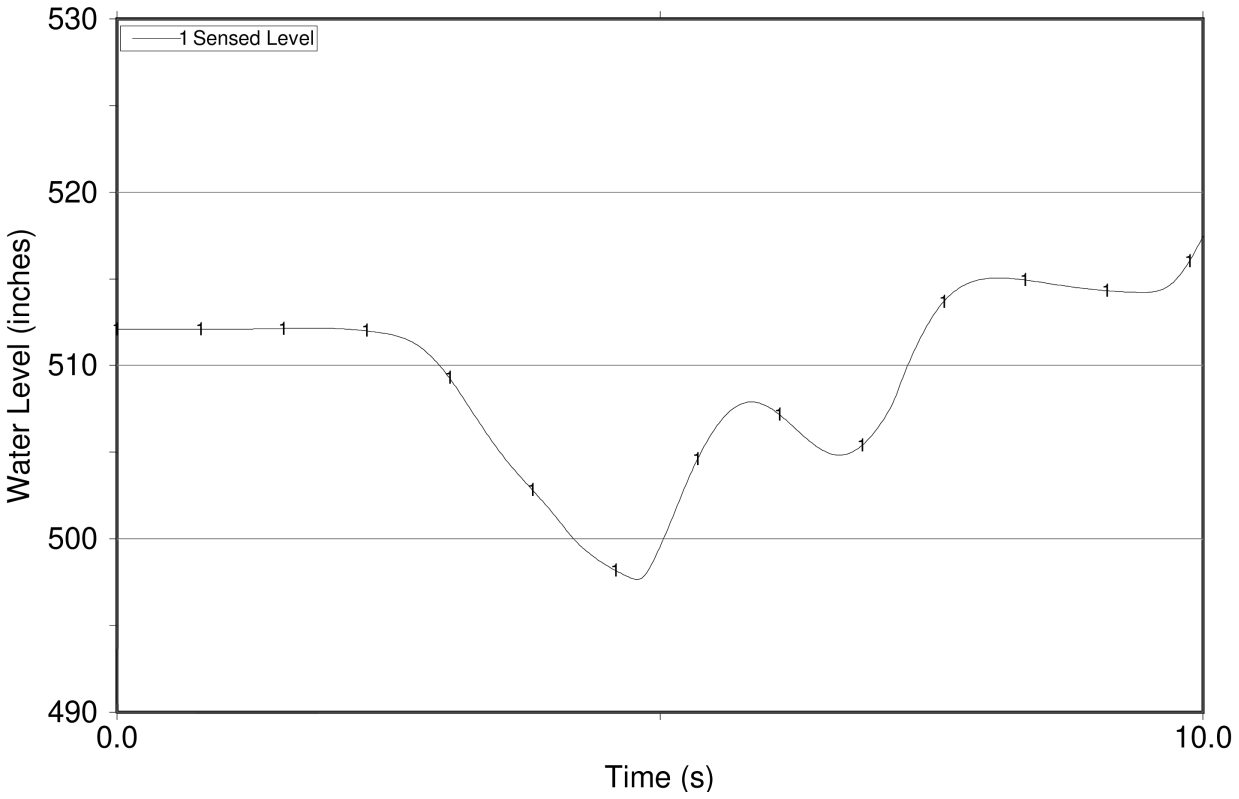


Figure 4.24 MSIV ATWS Overpressurization Event at 100P/80F – Sensed Water Level

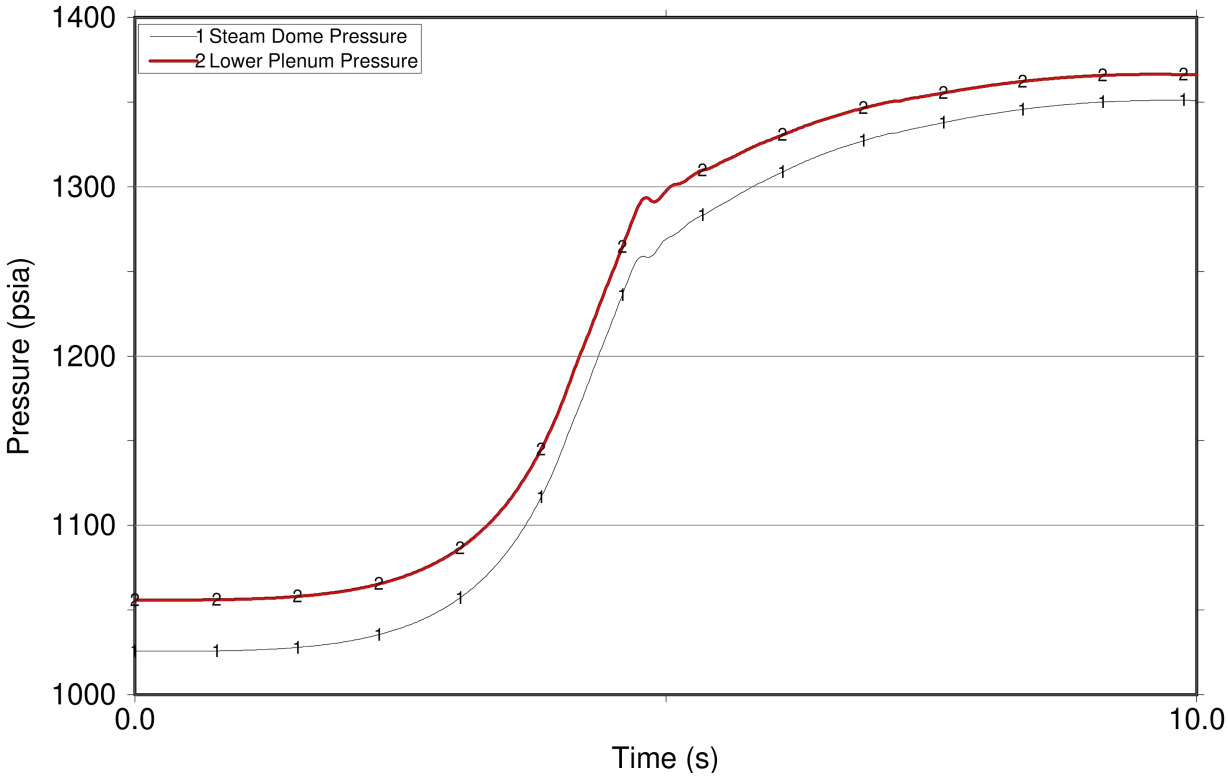


Figure 4.25 MSIV ATWS Overpressurization Event at 100P/80F – Vessel Pressures

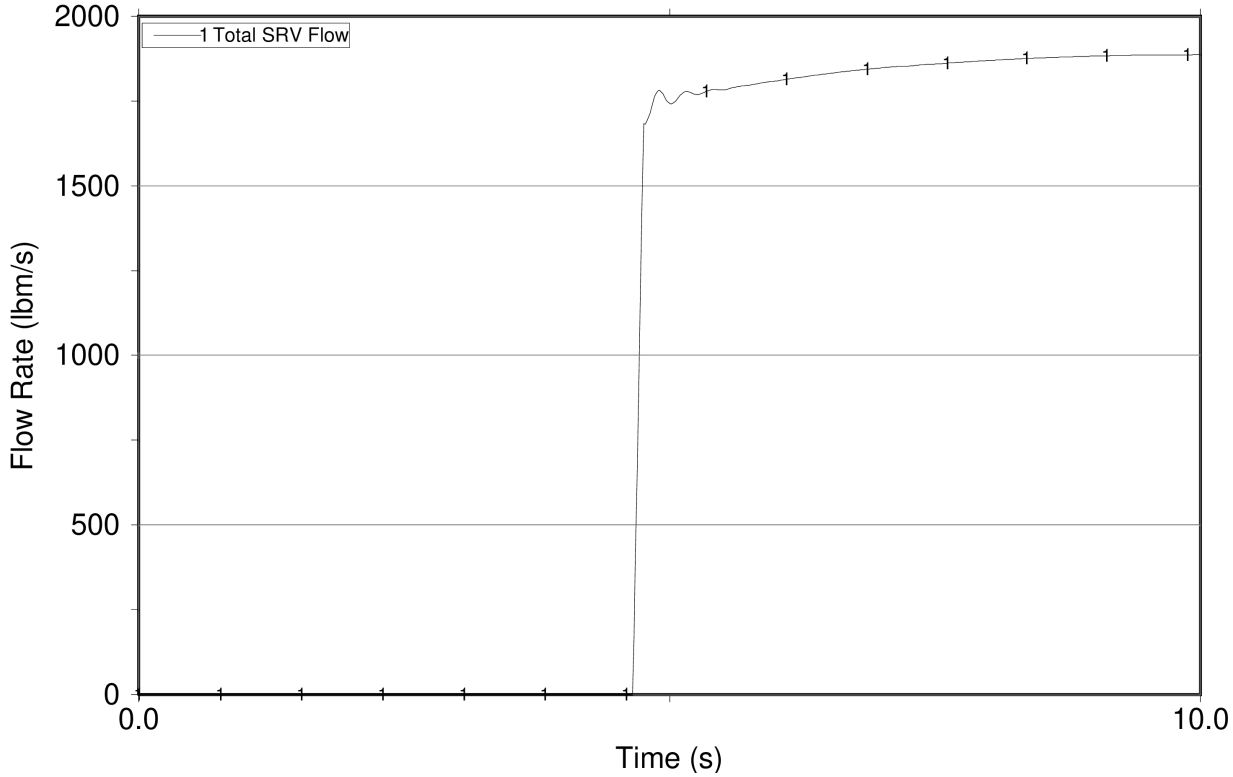


Figure 4.26 MSIV ATWS Overpressurization Event at 100P/80F – Safety/Relief Valve Flow Rates

5.0 PREVIOUSLY REQUESTED ADDITIONAL INFORMATION FROM THE NRC

The following subsections provide information responsive to applicable requests for additional information by the NRC which supports the introduction of ATRIUM 11 fuel to Monticello.

5.1 *Decay Heat and Station Blackout*

5.1.1 Decay Heat

In general, the decay heat results are **[**

]

Fuel Design	^{235}U Fission Fraction	^{238}U Fission Fraction	^{239}Pu Fission Fraction
GE14	0.4821 - 0.5363	0.0699 - 0.0997	0.3938 – 0.4182
ATRIUM 10XM	0.4635 - 0.5547	0.0703 - 0.1020	0.3750 - 0.4346
ATRIUM 11	0.4439 – 0.5362	0.0703 – 0.1058	0.3936 – 0.4502

[

]

5.1.2 **Station Blackout**

The licensing basis analysis remains applicable to Framatome fuel since station blackout is solely driven by decay heat. All other criteria are not fuel dependent. As discussed in Section 5.1.1, the decay heat is insignificantly impacted by the introduction of the ATRIUM 11 fuel design. Framatome fuel is designed to perform in a manner similar to and analogous with fuel of current and previous designs.

5.2 ***Event Changes Potentially Impacting Mass and Energy Release***

[

] No other plant design changes are planned during this transition to ATRIUM 11 fuel. Therefore the analysis of record is not impacted by the introduction of ATRIUM 11 fuel. In the future, plant modifications will be dispositioned for their impact on the licensing basis events and analyses will be updated as necessary.

5.3 ***Implementation of Transition Cycle Transient Limitation and Conditions***

Appendix A provides a listing of, and initial compliance with, the limitations and conditions (L&Cs) related to the AURORA-B AOO topical report, Reference 1. As noted in the Appendix, a few L&Cs require plant specific review which will be provided as part of the initial application (first transition licensing reports). The initial application cycle reload report is scheduled to be provided for information at a later date and will contain compliance with the 4 remaining, plant specific L&Cs 7, 11, 16, and 18a. The application of the L&Cs to the Monticello licensing evaluations are discussed below:

Limitation and Condition 7

> *As discussed in Section 3.6 of this SE, licensees should provide justification for the key plant parameters and initial conditions selected for performing sensitivity analyses on an event-specific basis. Licensees should further justify that the input values ultimately chosen for these key plant parameters and initial*

> *conditions will result in a conservative prediction of FoMs when performing calculations according to the AURORA-B EM described in ANP-10300P.*

As part of the initial preparations for licensing Monticello, Framatome will review the plant parameters document for the key parameters associated with the potentially limiting events. Framatome will also look for parameters that have a range of values that may be allowed for operational flexibility. Likewise, for initial conditions, Framatome will examine the range allowed during normal operation. This will include initial conditions such as power, flow, pressure, and inlet subcooling. Sensitivity studies will be performed for all of these key parameters/conditions for all FoM (MCPR, LHGR, and overpressure) and **[**

]

Limitation and Condition 11

> *AREVA will provide justification for the uncertainties used for the highly ranked plant specific PIRT parameters C12, R01, R02, and SL02 on a plant-specific basis, as described in Table 3.2 of this SE.*

The parameter C12 is the **[**

]

[

]

Limitation and Condition 16

> ***[ ]*** *is not sampled as part of the methodology, justification should be provided on a plant-specific basis that a conservative flow rate has been assumed* ***[ ]***

The **[ ]** is provided by Monticello in the plant parameters document. This flow accounts for **[ ]**. The AURORA-B model **[**

]

Limitation and Condition 18a

> *Plant-specific licensing applications shall describe and provide justification for the method for determining and applying conservative measures in future deterministic analyses for each FoM (e.g., biasing calculational inputs, postprocessing adjustments to calculated nominal results).*

For licensing calculations at Monticello, **[**

]

For the LHGRFACp evaluations **[**

]

6.0 REFERENCES

1. ANP-10300P-A Revision 1, *AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios,* Framatome Inc., January 2018.

2. ANP-10335P-A Revision 0, *ACE/ATRIUM 11 Critical Power Correlation*, Framatome Inc., May 2018.

3. ANP-3881P Revision 0, *Monticello ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report,* Framatome Inc., November 2020.

4. ANP-3924P Revision 0, *Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel*, Framatome Inc., June 2021.

5. ANP-3933P Revision 0, *Monticello ATWS-I Evaluation for ATRIUM 11 Fuel*, Framatome Inc, June 2021.

Appendix A LIMITATIONS FROM THE SAFETY EVALUATION FOR LTR ANP-10300P-A REVISION 1

Compliance to the limitations and conditions from Section 5.0 of the safety evaluation in ANP-10300P-A Revision 1, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios” (Reference 1) is discussed in the following table.

Appendix A *(Continued)*

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
1	AURORA-B may not be used to perform analyses that result in one or more of its CCDs (S-RELAP5, MB2-K, MICROBURN-B2, RODEX4) operating outside the limits of approval specified in their respective TRs, SEs, and plant-specific LARs. In the case of MB2-K, MB2-K is subject to the same limitations and conditions as MICROBURN-B2.	All methods are used within their limits of approval.
2	The regulatory limit contained in 10 CFR 50.46(b)(2) , requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. Because AURORA-B makes use of the Cathcart-Pawel oxidation correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness (See Section 3.4.3.1). Should the NRC staff position regarding the appropriate acceptance criterion for the Cathcart-Pawel correlation change, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.	**[** **]**

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
3	Parameter uncertainty distributions and their characterizing upper and lower *2σ* levels are presented in Table 3.6 and discussed in Section 3.6 of this SE. The distribution types will not be changed and the characterizing upper and lower *2σ* uncertainties will not be reduced without prior NRC approval. In the cases of the parameters **[** **]**, the respective methodologies discussed in Section 3.6.4.10 and Section 3.6.4.17 shall be used when determining the associated upper and lower *2σ* levels. The **[** **]** is subject to Limitation and Condition No. 4, below.	**[** **]**
4	As discussed in Section 3.3.1.2, before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are modeled in licensing analyses using AURORA-B, AREVA must justify that the AURORA-B EM can acceptably predict void fraction results for the new fuel designs within the **[** **]** prediction uncertainty bands. Otherwise, the prediction uncertainty bands should be appropriately expanded, and the **[** **]** should be appropriately updated utilizing the methodology discussed in Section 0 of this SE.	Justification of void fraction application to ATRIUM 11 fuel is provided in Section 5.1 of Reference 4.
5	As discussed in Section 3.3.2.4.4, before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are included in licensing analyses performed using the AURORA-B EF, AREVA must justify that the **[** **]** void-quality correlation within MICROBURN-B2 is valid for the new fuel designs at EPU and EFW conditions.	Justification of void fraction application to ATRIUM 11 fuel is provided in Section 5.1 of Reference 4.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
6	The *2σ* ranges **[** **]** until AREVA supplies additional justification (e.g., as part of a first-time application analysis) demonstrating an acceptable alternative for NRC review and approval. For **[** **]** will be utilized when performing licensing analyses to determine peak cladding temperature and maximum local oxidation. Should the NRC staff position regarding these uncertainties change as a result of additional justification, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.	**[** **]**
7	As discussed in Section 3.6 of this SE, licensees should provide justification for the key plant parameters and initial conditions selected for performing sensitivity analyses on an event-specific basis. Licensees should further justify that the input values ultimately chosen for these key plant parameters and initial conditions will result in a conservative prediction of FoMs when performing calculations according to the AURORA-B EM described in ANP-10300P.	See Section 5.3 for further discussion of this L&C. The justification for the key plant parameters and initial conditions will be thoroughly explored for the first application cycle for Monticello.
8	The sampling ranges for uncertainty distributions used in the non-parametric order statistics analyses will be truncated at no less than ±*6σ* **[** **]**	**[** **]**

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
9	For any highly ranked PIRT phenomena whose uncertainties are not addressed in a given non-parametric order statistics analysis via sampling , AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.2 of this SE. For any pertinent medium ranked PIRT phenomena whose uncertainties are not addressed in a given non-parametric order statistics analysis via sampling, AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.4 of this SE.	Framatome analyses comply with the requirements of Tables 3.2 and 3.4 of the SE.
10	The assumptions of **[** **]** will be used in the AURORA-B EM to ensure the uncertainty in SL03: **[** **]** is conservatively accounted for.	**[** **]**
11	AREVA will provide justification for the uncertainties used for the highly ranked plant-specific PIRT parameters C12, R01, R02, and SL02 on a plant-specific basis, as described in Table 3.2 of this SE.	See Section 5.3 for further discussion of this L&C. **[** **]**
12	When applying the AURORA-B EM to the **[** **]** , any changes to AURORA-B to enhance **[** **]** on a plant-specific basis without prior NRG review and approval are not approved as part of this SE, as described in Table 3.2 of this SE.	**[** **]**

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
13	The AURORA-B uncertainty methodology discussed in Section 3.6 of this SE may be used in licensing applications for the events listed in Section 3.1 of this SE, with the exception of three specific events identified in Section 3.6.2 of this SE : **[** **]**. These events are generally expected to be benign and hence nonlimiting. While the NRC staff's review concluded that the AURORA-B EM contains code models and correlations sufficient for simulating these events, the uncertainty methodology developed in the TR did not address certain important phenomena or conditions associated therewith. Therefore, while licensing applications may rely on nominal calculations with the AURORA-B EM for these events in the course of demonstrating that all regulatory limits are satisfied with significant margin, the existing uncertainty methodology may not be applied directly to these specific events.	**[** **]**
14	The scope of the NRC staff's approval for AURORA-B does not include the ABWR design.	Not applicable. Monticello is not an ABWR design.
15	For application to BWR/2s at EPU or EFW conditions, plant-specific justification should be provided for the applicability of AURORA-B, as discussed in Section 3 .1 of this SE.	Not applicable. Monticello is not a BWR/2
16	**[** **]** is not sampled as part of the methodology, justification should be provided on a plant-specific basis that a conservative flow rate has been assumed **[** **]**.	See Section 5.3 for further discussion of this L&C. **[** **]** will be conservatively modeled for the first application cycle for Monticello.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
17	If the AURORA-B EM calculates that the film boiling regime is entered during a transient or accident, AREVA must justify that the uncertainty associated with heat transfer predictions in the film boiling regime is adequately addressed.	[ ]
18	As discussed in Section 3.6.5 of this SE regarding conservative measures : a. Plant-specific licensing applications shall describe and provide justification for the method for determining and applying conservative measures in future deterministic analyses for each FoM (e.g., biasing calculational inputs, post-processing adjustments to calculated nominal results), and	See Section 5.3 for further discussion of this L&C. Compliance with this item will be demonstrated for the first application cycle for Monticello.
	b. If the 95/95 FoM for a given parameter calculated according to the defined conservative measures during a deterministic analysis shows a difference in magnitude exceeding 1σ from the corresponding value calculated in the most recent baseline full statistical analysis, AREVA must re-perform the full statistical analysis for the affected scenario and determine new conservative measures.	[ ]
19	As discussed in Section 3.6.5 of this SE, the following stipulations are necessary to ensure that the FoMs calculated by AREVA in accordance with ANP-10300P would satisfy the 95/95 criterion: a. AREVA will use multivariate order statistics when multiple FoMs are drawn from a single set of statistical calculations,	Framatome calculations will utilize the multivariate order statistics when a single transient is used to determine multiple figures of merits.
	b. AREVA will choose the sample size prior to initiating statistical calculations,	Framatome will choose the sample size prior to initiating statistical calculations.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
	c. AREVA will not arbitrarily discard undesirable statistical results, and	Framatome will not arbitrarily discard undesirable statistical results.
	d. AREVA will maintain an auditable record to demonstrate that its process for performing statistical licensing calculations has been executed in an unbiased manner.	Framatome will maintain an auditable record to demonstrate the process for performing statistical licensing calculations is being executed in an unbiased manner.
20	The implementation of any new methodology within the AURORA-B EM (i.e., replacement of an existing CCD) is not acceptable unless the AURORA-B EM with the new methodology incorporated into it has received NRC review and approval. An existing NRC-approved methodology cannot be implemented within the AURORA-B EM without NRC review of the updated EM.	The evaluation model is implemented as described in Reference 1. No CCD as described in Reference 1 is replaced.
21	NRC-approved changes that revise or extend the capabilities of the individual CCDs comprising the AURORA-B EM may not be incorporated into the EM without prior NRC approval.	Framatome regulatory procedures require use of NRC approved methodologies within the applicability defined for that methodology.
22	As discussed in Section 3.3.1.5 and Section 4.0 of this SE, the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EF. Other CPR correlations (existing and new) that would be used with the AURORA-B EM must be reviewed and approved by the NRC or must be developed with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel" (Reference 50). Furthermore, if transient thermal-hydraulic simulations are performed in the process of applying AREVA CPR correlations to co-resident fuel, these calculations should use the AURORA-B methodology.	Framatome regulatory procedures require use of NRC approved methodologies. The applicability of the ACE/ATRIUM 11 correlation for use in the AURORA-B AOO methodology is described in Reference 2.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
23	Except when prohibited elsewhere, the AURORA-B EM may be used with new or revised fuel designs without prior NRC approval provided that the new or revised fuel designs are substantially similar to those fuel designs already approved for use in the AURORA-B EM (i.e., thermal energy is conducted through a cylindrical ceramic fuel pellet surrounded by metal cladding, flow in the fuel channels develops into a predominantly vertical annular flow regime, etc.). New fuel designs exhibiting a large deviation from these behaviors will require NRC review and approval prior to their implementation in AURORA-B.	Framatome has no fuel designs that exhibit a large deviation from the behaviors described in this limitation and condition. If a fuel design is developed that is significantly different, this fuel design will be submitted to the NRC for approval.
24	Changes may be made to the AURORA-B EM in the **[** **]** areas discussed in Section 4.0 of this SE without prior NRC approval.	**[** **]**
25	The parallelization of individual CCDs may be performed without prior NRC approval as discussed in Section 4.0 of this SE.	**[** **]**
26	AREVA must continue to use existing regulatory processes for any code modifications made in the **[** **]** areas discussed in Section 4.0 of this SE.	**[** **]**

ENCLOSURE

ATTACHMENT 11b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3925P REPORT, REVISION 0

MONTICELLO ATRIUM 11 TRANSIENT DEMONSTRATION

JULY 2021

(3 pages follow)

AFFIDAVIT

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3925P Revision 0, "Monticello ATRIUM 11Transient Demonstration," dated July 2021 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: July 6, 2021

Alan B. Meginnis

ENCLOSURE

ATTACHMENT 12a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3934NP REPORT, REVISION 0

MONTICELLO LOCA ANALYSIS FOR ATRIUM 11 FUEL

JULY 2021

(75 pages follow)

Monticello LOCA Analysis for ATRIUM 11 Fuel

ANP-3934NP
Revision 0

July 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1.	All	Initial Issue

Contents

Tables

Figures

Nomenclature

ADS	automatic depressurization system
ADSVOOS	ADS valve out-of-service
BWR	boiling-water reactor
CFR	Code of Federal Regulations
CHF	critical heat flux
CMWR	core average metal-water reaction
DC	direct current
DEG	double-ended guillotine
DG	diesel generator
ECCS	emergency core cooling system
EFW	extended flow window
FHOOS	feedwater heaters out-of-service
HPCI	high-pressure coolant injection
ICF	increased core flow
ID	inside diameter
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MSIV	main stream isolation valve
MWR	metal-water reaction
NSSS	nuclear steam supply system
NRC	Nuclear Regulatory Commission, U.S.
OD	outside diameter
PCT	peak cladding temperature

Nomenclature (continued)

RCIC	reactor core isolation cooling
RDIV	recirculation discharge isolation valve
RWCU	reactor water cleanup
SE	Safety Evaluation
SF-ADS	single failure of ADS
SF-BATT	single failure of battery (DC) power
SF-DGEN	single failure of a diesel generator
SF-HPCI	single failure of the HPCI system
SF-LPCI	single failure of LPCI injection valve
SLO	single-loop operation
TLO	two-loop operation

1.0 Introduction

The results of a loss-of-coolant accident (LOCA) break spectrum and emergency core cooling system (LOCA-ECCS) analyses for Monticello are documented in this report. The purpose of the break spectrum analysis is to identify the break characteristics that result in the highest calculated peak cladding temperature (PCT) **[**

] during a postulated LOCA. The results provide the maximum average planar linear heat generation rate (MAPLHGR) limit for ATRIUM 11 fuel as a function of exposure for normal (two-loop) operation.

Variation in the following LOCA parameters is examined:

- Break location
- Break type (double-ended guillotine (DEG) or split)
- Break size
- Limiting ECCS single failure
- Axial power shape (top- or mid-peaked)
- Initial statepoint
- Fuel rod type

The analyses documented in this report are performed with LOCA Evaluation Models developed by Framatome*, and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by Framatome for LOCA analyses are collectively referred to as the AURORA-B LOCA Evaluation Model (References 1 – 3). The calculations described in this report are performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46.

Key model characteristics included in the report analyses are shown below. Other initial conditions used in the analyses are described in Section 4.0.

- Operation in the extended flow window (EFW) domain of Figure 1.1 is supported. **[**

]

* Framatome Inc. formerly known as AREVA Inc.

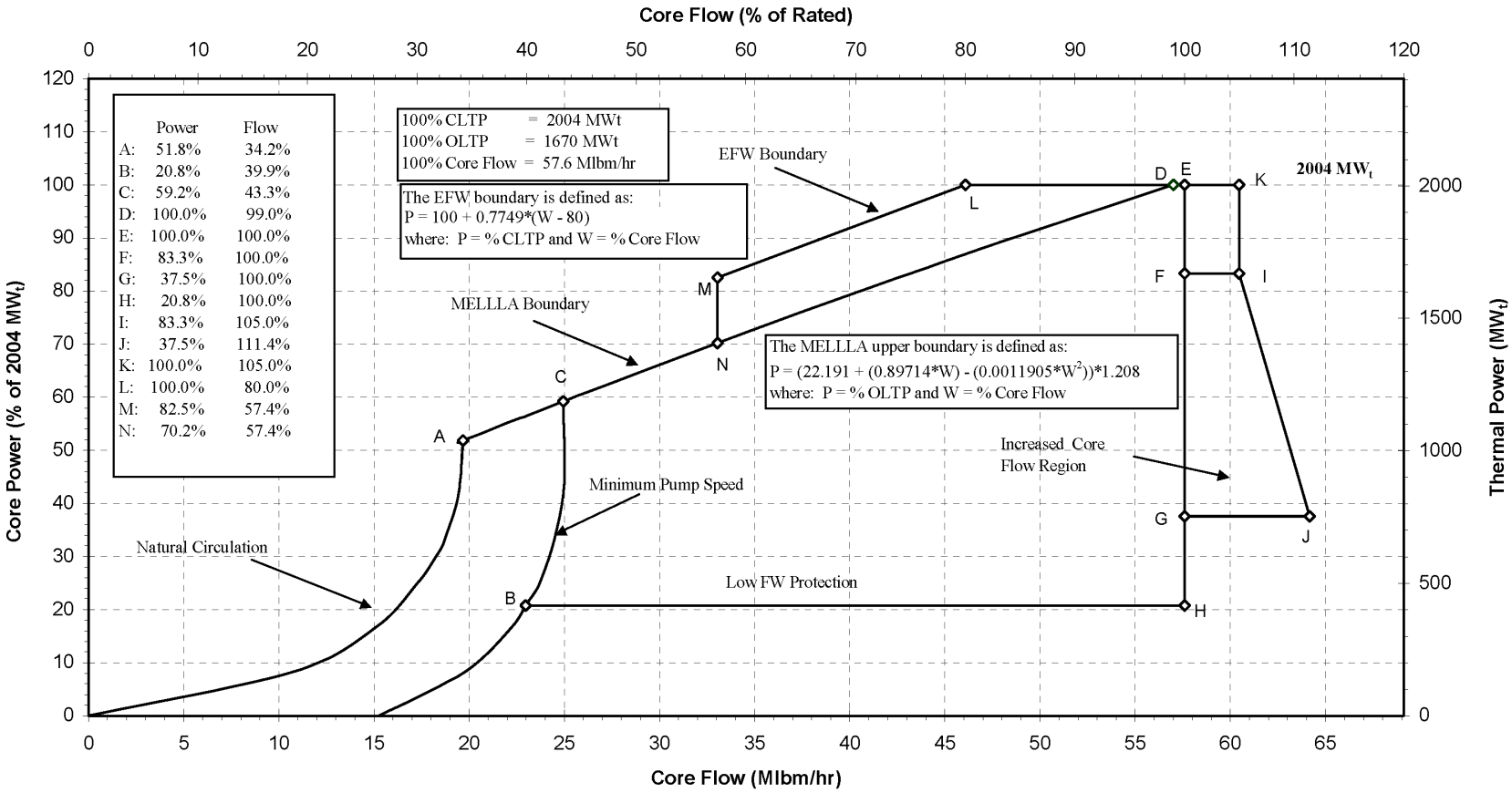
[ ]

- [ ]
- The core is composed entirely of ATRIUM 11 fuel.
- A 2.0% increase in initial core power to address the maximum uncertainty in monitoring reactor power, as per NRC requirements, is included.
- [ ] were assumed to be at the MAPLHGR limit shown in Figure 2.1.
- [ ]

The limiting break characteristics from the break spectrum study are used in analyses to determine the MAPLHGR limit and [ ] versus exposure. Even though the limiting break will not change with exposure, the value of PCT calculated for any given set of break characteristics is dependent on exposure and the corresponding MAPLHGR and [ ].

Single-loop operation (SLO) results are discussed in Section 7.0. Long term coolability is addressed in Section 8.0.

Figure 1.1 Monticello Power / Flow Map
EPU/EFW



2.0 Summary of Results

The LOCA analysis results presented in this report are applicable to Monticello. A more detailed discussion of results is provided in Sections 6.0 – 7.0.

The PCT and metal-water reaction (MWR) results, from the ATRIUM 11 fuel exposure-dependent analysis presented in Section 9.0, are presented below.

Parameter	ATRIUM 11*
Peak cladding temperature (°F)	2120 [ ]
Local cladding oxidation (max %)	11.22 [ ]
Total hydrogen generated (% of total hydrogen possible)	< 0.97

The MAPLHGR limit was determined by applying the AURORA-B LOCA Evaluation Model for the analysis of the limiting LOCA event. The exposure-dependent MAPLHGR limit for ATRIUM 11 fuel is shown in Figure 2.1. Exposure dependent results with the [ ] are presented in Section 9.0. The results of these calculations confirm that the LOCA acceptance criteria in the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below these limits.

The LOCA analysis results (i.e., the limiting break characteristics and exposure analysis) presented in this report are applicable for a full core of ATRIUM 11 fuel as well as transition cores containing ATRIUM 11 fuel. [ ]

* [ ]

[

]

The SLO LOCA analyses support operation with an ATRIUM 11 multiplier of 0.8 applied to the normal two-loop operation MAPLHGR limit. [

]

The long-term coolability evaluation confirms that the ECCS capacity is sufficient to maintain adequate cooling in an ATRIUM 11 core for an extended period after a LOCA.

Available ADS valves are presented in Table 5.1. No additional valves are assumed to be out-of-service (ADSVOOS). All analyses also support the [

]

The analysis supports operation in the EFW domain of the Monticello power/flow map shown in Figure 1.1.

Figure 2.1 MAPLHGR Limit for ATRIUM 11 Fuel Two-Loop Operation*

* A MAPLHGR multiplier of 0.8 is required for single-loop operation.

Figure 2.2 [ ] for ATRIUM 11 Fuel

[ ]

3.0 LOCA Description

3.1 *Accident Description*

The LOCA is described in the Code of Federal Regulations 10 CFR 50.46 as a hypothetical accident that results in a loss of reactor coolant from breaks in reactor coolant pressure boundary piping up to and including a break equivalent in size to a double-ended rupture of the largest pipe in the reactor coolant system. There is not a specifically identified cause that results in the pipe break. However, for the purpose of identifying a design basis accident, the pipe break is postulated to occur inside the primary containment before the first isolation valve.

For a boiling water reactor (BWR), a LOCA may occur over a wide spectrum of break locations and sizes. Responses to the break vary significantly over the break spectrum. The largest possible break is a double-ended rupture of a recirculation pipe; however, this is not necessarily the most severe challenge to the ECCS. A double-ended rupture of a main steam line causes the most rapid primary system depressurization, but because of other phenomena, steam line breaks are seldom limiting with respect to the event acceptance criteria (10 CFR 50.46). Because of these complexities, an analysis covering the full range of break sizes and locations is performed to identify the limiting break characteristics.

Regardless of the initiating break characteristics, the event response is conveniently separated into three phases: the blowdown phase, the refill phase, and the reflood phase. The relative duration of each phase is strongly dependent upon the break size and location. **[**

]

During the blowdown phase of a LOCA, there is a net loss of coolant inventory, an increase in fuel cladding temperature due to core flow degradation, and for the larger breaks, the core becomes fully or partially uncovered. There is a rapid decrease in pressure during the blowdown phase. During the early phase of the depressurization,

the exiting coolant provides core cooling. Consistent with the discussion presented in Reference 1, **[**

]

In the refill phase of a LOCA, the ECCS is functioning and there is a net increase of coolant inventory. During this phase the core sprays provide core cooling and, along with low-pressure and high-pressure coolant injection (LPCI and HPCI), supply liquid to refill the lower portion of the reactor vessel. In general, the core heat transfer to the coolant is less than the fuel decay heat rate and the fuel cladding temperature continues to increase during the refill phase.

In the reflood phase, the coolant inventory has increased to the point where the mixture level re-enters the core region. During the core reflood phase, cooling is provided above the mixture level by entrained reflood liquid and below the mixture level by pool boiling. Sufficient coolant eventually reaches the core hot node and the fuel cladding temperature decreases. **[**

]

3.2 *Acceptance Criteria*

A LOCA is a potentially limiting event that may place constraints on fuel design, local power peaking, and in some cases, acceptable core power level. During a LOCA, the normal transfer of heat from the fuel to the coolant is disrupted. As the liquid inventory in the reactor decreases, the decay heat and stored energy of the fuel cause a heatup of the undercooled fuel assembly. In order to limit the amount of heat that can contribute to the heatup of the fuel assembly during a LOCA, an operating limit on the MAPLHGR is applied to each fuel assembly in the core.

The Code of Federal Regulations prescribes specific acceptance criteria (10 CFR 50.46) for a LOCA event as well as specific requirements and acceptable features for Evaluation Models (10 CFR 50 Appendix K). The conformance of the AURORA-B LOCA Evaluation Models to Appendix K is described in Reference 1. The ECCS must be designed such that the plant response to a LOCA meets the following acceptance criteria specified in 10 CFR 50.46:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are commonly referred to as the PCT criterion, the local oxidation criterion, the hydrogen generation criterion, the coolable geometry criterion, and the long-term cooling criterion. A MAPLHGR limit is established for each fuel type to ensure that these criteria are met.

LOCA results are provided in Section 6.0 to identify the LOCA events which produce the highest PCT [ ] LOCA analysis results demonstrating that the PCT, local oxidation, and hydrogen generation (core wide oxidation) criteria are met are provided in Section 9.0. Compliance with these three criteria ensures that a coolable geometry is maintained. Long-term coolability criterion is discussed in Section 8.0.

4.0 LOCA Analysis Description

The Evaluation Model used for the break spectrum analysis is the AURORA-B LOCA analysis methodology described in Reference 1. The AURORA-B LOCA methodology employs two major computer codes to evaluate the system and fuel response during all phases of a LOCA. These are the S-RELAP5 and RODEX4 computer codes. A **[** **]** of the LOCA to determine the PCT and maximum local clad oxidation for **[** **]**

A complete analysis starts with the specification of fuel parameters using RODEX4 (Reference 3). RODEX4 is used to determine the **[** **]** The initial stored energy used in S-RELAP5 is **[** **]**

4.1 *Break Spectrum Analysis*

S-RELAP5 is used to calculate the thermal-hydraulic response during all phases of the LOCA using a **[** **]** The reactor vessel nodalization is shown in Figure 4.1 and the core nodalization is shown in Figure 4.2 consistent with those in the topical report approved by the NRC (Reference 1). The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and decay heat as required by Appendix K of 10 CFR 50. The clad swelling and rupture models from NUREG-0630 (Reference 2) have been incorporated into S-RELAP5.

The S-RELAP5 model is executed over a range of break locations, break sizes, break types, initial statepoints, axial shapes and assumed single-failures to determine the break that yields the highest PCT **[**

]

4.2 *Exposure Analysis*

The **[**

] from beginning-of-life to end-of-life **[** **]** increments to determine an exposure-dependent MAPLHGR limit and **[**

] Figures of merit including PCT, local cladding oxidation, and core-wide metal-water reaction are evaluated over the range of exposures to confirm the acceptability of the LOCA analysis with respect to 10 CFR 50.46 criteria. **[**

]

4.3 *Plant Parameters*

The LOCA analysis is performed using the plant parameters provided by the utility. Table 4.1 provides a summary of reactor initial conditions used in the break spectrum analysis. Table 4.2 lists selected reactor system parameters.

The LOCA analysis is performed for a full core of ATRIUM 11 fuel. Some of the key fuel parameters used in the analysis are summarized in Table 4.3.

4.4 *ECCS Parameters*

The ECCS configuration is shown in Figure 4.3. Table 4.4 – Table 4.7 provide the important ECCS characteristics assumed in the analysis. The ECCS is modeled as time-dependent junctions connected to the appropriate reactor locations: LPCS injects into the upper plenum, HPCI injects into the upper downcomer, and LPCI injects into the recirculation lines.

The flow through each ECCS valve is determined based on system pressure and valve position. Flow versus pressure for a fully open valve is obtained by linearly interpolating the pump capacity data provided in Table 4.4 – Table 4.6. No credit for ECCS flow is assumed until the ECCS injection valves open and the ECCS pumps reach rated speed.

The ADS valves are modeled as a junction connecting the reactor steam line to the suppression pool. The flow through the ADS valves is calculated based on pressure and valve flow characteristics. The valve flow characteristics are determined such that the calculated flow is equal to the rated capacity at the reference pressure shown in Table 4.7.

In the Framatome LOCA analysis model, ECCS initiation is assumed to occur when the water level drops to the applicable level setpoint. No credit is assumed for the start of HPCI, LPCS, or LPCI due to high drywell pressure. **[**

]

The recirculation discharge isolation valve (RDIV) parameters are shown in Table 4.8.

Table 4.1 Initial Conditions

Reactor power (% of rated)	102	102	[	]
[				]
Reactor power (MWt)	2044.1	2044.1	[	]
[				]
[				]
Steam flow rate (Mlb/hr)	8.5	8.5	[	]
Steam dome pressure (psia)	1048	1048	[	]
[				]
[				]
[				]
[				]
Rod average power distributions	Figure 4.4	Figure 4.5	Figure 4.6	Figure 4.7

* [ ]

Table 4.2 Reactor System Parameters

Parameter	Value
Vessel ID (in)	205
Number of fuel assemblies	484
Recirculation suction pipe area (ft^2)	3.679
1.0 DEG suction break area (ft^2)	7.358
Recirculation discharge pipe area (ft^2)	3.679
1.0 DEG discharge break area (ft^2)	7.358

Table 4.3 ATRIUM 11 Fuel Assembly Parameters

[ ]

* Does not include additional inner channel milling near the top of the channel.

Table 4.4 High-Pressure Coolant Injection Parameters

Parameter	Value
Coolant temperature (°F)	127
Initiating Signals and Setpoints	
Water level*	L2 (422.1 in)
High drywell pressure (psig)	Not credited
Time Delays	
Time for HPCI pump to reach rated speed and injection valve wide open (sec)	45
Delivered Coolant Flow Rate Versus Pressure	
Vessel to Torus ΔP (psid)	Flow Rate (gpm)
0	0
150	2700
1120	2700

* Relative to vessel zero.

Table 4.5 Low-Pressure Coolant Injection Parameters

Parameter	Value
Reactor pressure permissive for opening valves (psia)	350
Coolant temperature (°F)	90
Initiating Signals and Setpoints	
Water level*	L2 (422.1 in)
High drywell pressure (psig)	Not credited
Time Delays	
Total system delay from initiating signal until the system is ready to inject (sec)	53.2
LPCI injection valve stroke time (sec)	35

Delivered Coolant Flow Rate Versus Pressure

Vessel to Drywell ΔP (psid)	Flow Rate for 2 Pumps Injecting Into 1 Recirculation Loop (gpm)	Vessel to Drywell ΔP (psid)	Flow Rate for 4 Pumps Injecting Into 1 Recirculation Loop (gpm)
0	8,038	0	14,733
20	7,740	20	14,184
72	6,872	72	12,582
151	5,249	151	9,600
209	2,528	203	5,187
226	0	222	0

* Relative to vessel zero.

Table 4.6 Low-Pressure Core Spray Parameters

Parameter	Value
Reactor pressure permissive for opening valves (psia)	350
Coolant temperature (°F)	90
Initiating Signals and Setpoints	
Water level*	L2 (422.1 in)
High drywell pressure (psig)	Not credited
Time Delays	
Time for LPCS pumps to reach rated speed (maximum) (sec) †	38
LPCS injection valve stroke time (sec)	15
Delivered Coolant Flow Rate Versus Pressure	
Vessel to Drywell ΔP (psid)	Flow Rate for 1 Pump (gpm)
0	3,700
130	2,700
240	1,125
279	0

* Relative to vessel zero.

† Includes 15-second delay for diesel generator start. 1-second signal processing delay for water level trip L2 is assumed in parallel with diesel generator delay.

Table 4.7 Automatic Depressurization System Parameters

Parameter	Value
Number of valves installed	3
Number of valves available	3
Minimum flow capacity per valve (lbm/hr at 1080 psig)	814,500*
Initiating Signals and Setpoints	
Water level†	L2 (422.1 in)
Time Delays	
Delay time (from ADS initiating signal to time valves are open (sec)‡	138

* [ ]

† Relative to vessel zero.

‡ ADS time initiation occurs after L2 setpoint trip. ADS valves are opened after the timer has elapsed. [ ]

Table 4.8 Recirculation Discharge Isolation Valve Parameters

Parameter	Value
Reactor pressure permissive for closing valves – analytical (psig)	
Two-loop operation	None
Single-loop operation	900
Time Delays	
RDIV stroke time after pressure permissive and initiating signal (sec)	35

Figure 4.1 S-RELAP5 Vessel Model

Figure 4.2 S-RELAP5 Core Model

Figure 4.3 ECCS Schematic

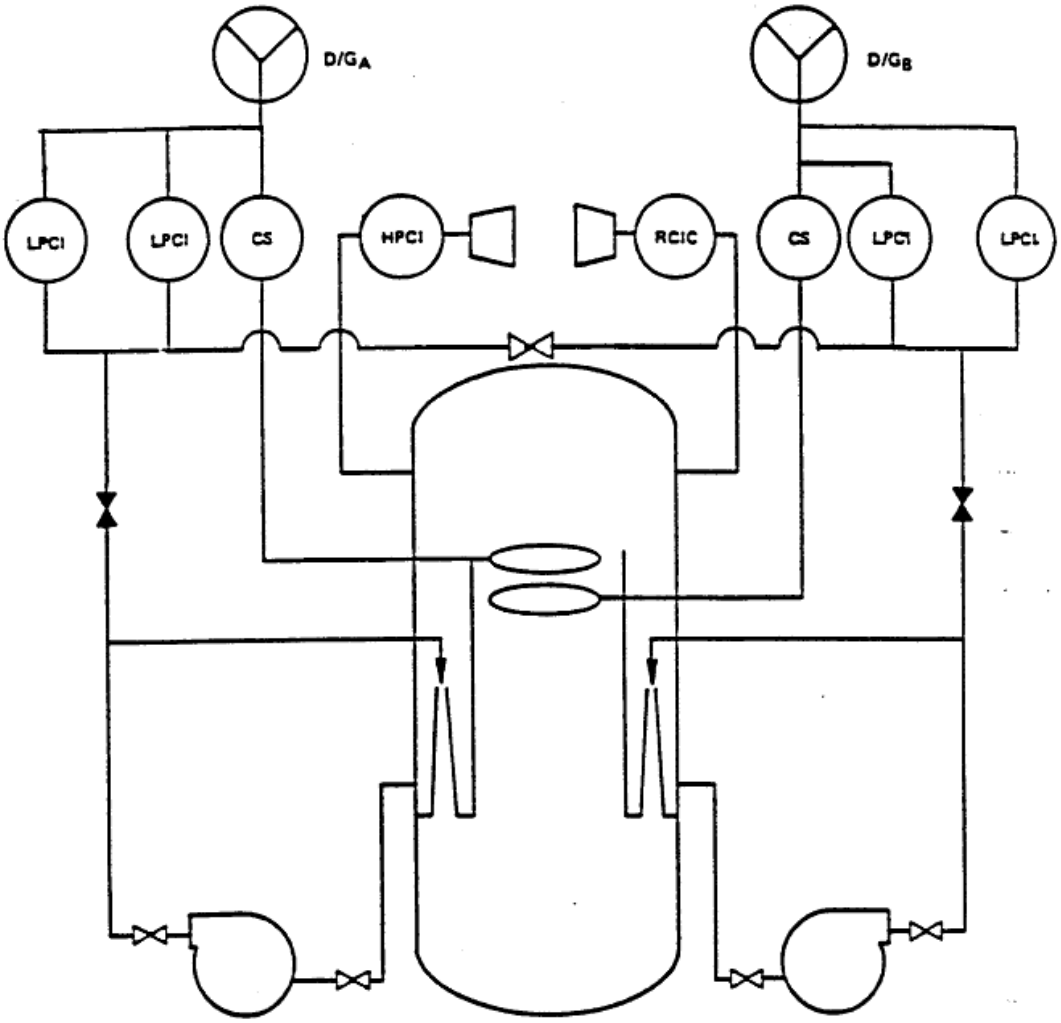


Figure 4.4 Rod Average Power Distributions for 102%P and [ ] Mid- and Top-Peaked

[ ]

Figure 4.5 Rod Average Power Distributions for 102%P and [ ] Mid- and Top-Peaked

[ ]

Figure 4.6 Rod Average Power Distributions for [ ] Mid- and Top-Peaked

[ ]

Figure 4.7 Rod Average Power Distributions for [ ] Mid- and Top-Peaked

[ ]

5.0 Break Spectrum Analysis Description

The objective of the LOCA break spectrum analyses is to ensure that the operating conditions, break location, break type, break size, and ECCS single failure that produce the maximum PCT **[ ]** are identified. The LOCA response scenario varies considerably over the spectrum of break locations. Potential break locations have been separated into two groups: recirculation line breaks and non-recirculation line breaks. The basis for the break locations and potentially limiting single failures analyzed in this report is described in the following sections.

5.1 *Limiting Single Failure*

Regulatory requirements specify that the LOCA analysis consider availability of offsite power supplies and that only safety grade systems and components are available. In addition, regulatory requirements also specify that the most limiting single failure of ECCS equipment must be assumed in the LOCA analysis. The term "most limiting" refers to the ECCS equipment failure that produces the greatest challenge to event acceptance criteria. The limiting single failure can be a common power supply, an injection valve, a system pump, or system initiation logic. The most limiting single failure may vary with break size and location. The potential limiting single failures (SF) identified in the USAR (Reference 5), are shown below:

- DC power (SF-BATT)
- Diesel generator (SF-DGEN)
- Low pressure coolant injection (SF-LPCI)
- High-pressure coolant injection system (SF-HPCI)
- Automatic depressurization system (SF-ADS)

The single failures and the available ECCS for each failure assumed in these analyses are summarized in Table 5.1. Other potential failures are not specifically considered because they result in as much or more ECCS capacity.

5.2 *Recirculation Line Breaks*

The response during a recirculation line LOCA is dependent on break size. The rate of reactor vessel depressurization decreases as the break size decreases. The high-pressure ECCS and ADS will assist in reducing the reactor vessel pressure to the pressure where the LPCI and LPCS flows start. For large breaks, rated LPCS and LPCI flow is generally reached before or shortly after the time when the ADS valves open so the ADS system is not required to mitigate the LOCA. ADS operation is an important emergency system for small breaks where it assists in depressurizing the reactor system faster, and thereby reduces the time required to reach rated LPCS and LPCI flow.

The two largest flow resistances in the recirculation piping are the recirculation pump and the jet pump nozzle. For breaks in the discharge piping, there is a major flow resistance in both flow paths from the reactor vessel to the break. For breaks in the suction piping, both major flow resistances are in the flow path from the vessel to the pump side of the break. As a result, pump suction side breaks experience a more rapid blowdown, which tends to make the event more severe (if ECCS capacity is equal). With LPCI Loop Selection Logic (LSL), all available LPCI flow is diverted to the intact loop and the RDIV is closed in that loop to direct flow to the core for break areas $\geq 0.4\ ft^2$. For break areas $< 0.4\ ft^2$, the broken loop is instead selected for LPCI flow and RDIV closure.

Two break types (geometries) are considered for the recirculation line break. The two types are the double-ended guillotine (DEG) break and the split break.

For a DEG break, the piping is assumed to be completely severed resulting in two independent flow paths to the containment. The DEG break is modeled by setting the break area (at both ends of the pipe) equal to the full pipe cross-sectional area and varying the discharge coefficient between 1.0 and 0.4. The range of discharge coefficients is used to cover uncertainty in the actual geometry at the break.

[ ]

[ ] The most limiting DEG break is determined by varying the discharge coefficient.

A split type break is assumed to be a longitudinal opening or hole in the piping that results in a single break flow path to the containment. Appendix K of 10 CFR 50 defines the cross-sectional area of the piping as the maximum split break area required for analysis.

Break types, break sizes, and single failures are analyzed for both suction and discharge recirculation line breaks.

Section 6.0 provides a description and results summary for breaks in the recirculation line.

5.3 *Non-Recirculation Line Breaks*

In addition to breaks in the recirculation line, breaks in other reactor coolant system piping must be considered in the LOCA break spectrum analysis. Although the recirculation line large breaks result in the largest coolant inventory loss, they do not necessarily result in the most severe challenge to event acceptance criteria. The double-ended rupture of a main steam line is expected to result in the fastest depressurization of the reactor vessel. Special consideration is required when the postulated break occurs in ECCS piping. Although ECCS piping breaks are small relative to a recirculation pipe DEG break, they disable an ECCS system and therefore, increase the postulated break severity. The following sections address potential LOCAs due to breaks in non-recirculation line piping.

Non-recirculation line breaks outside containment are inherently less challenging to fuel limits than breaks inside containment. For breaks outside containment, isolation or check valve closure will terminate break flow prior to the loss of significant liquid inventory and the core will remain covered. If high-pressure coolant inventory makeup cannot be reestablished, ADS actuation may become necessary. [ ]

[

] Although analyses of breaks outside containment may be required to address non-fuel related regulatory requirements, these breaks are not limiting relative to fuel acceptance criteria such as PCT.

5.3.1 Main Steam Line Breaks

A steam line break [

] The break results in high steam flow out of the broken line and into the containment. Prior to MSIV closure, a steam line break also results in high steam flow in the intact steam lines as they feed the break via the steam line manifold. A steam line break inside containment results in a rapid depressurization of the reactor vessel. Initially the break flow will be high quality steam; however, the rapid depressurization produces a water level swell that results in liquid discharge at the break. For steam line breaks, the largest break size is most limiting because it results in the most level swell and liquid loss out of the break.

[

]

5.3.2 Feedwater Line Breaks

[

]

5.3.3 HPCI Line Breaks

The HPCI injection line is connected to the feedwater line outside containment.

[

]

The HPCI steam supply line is connected to the main steam line inside containment.

[

]

5.3.4 LPCS Line Breaks

A break in the LPCS line is expected to have many characteristics similar to [

] However, some characteristics of the LPCS line break are unique and are not addressed in other LOCA analyses. Two important differences from other LOCA analyses are that the break flow will exit from the region inside the core shroud and the break will disable one LPCS system. The LPCS line break is assumed to occur just outside the reactor vessel. [

]

5.3.5 LPCI Line Breaks

The LPCI injection lines are connected to the larger recirculation discharge lines. A break in a LPCI line would result in the partial or full loss of LPCI injection. **[**

]

5.3.6 RCIC Line Breaks

The RCIC discharges to the feedwater line; **[**

]

The steam supply to the RCIC turbine comes from the main steam line from the reactor vessel; **[**

]

5.3.7 RWCU Line Breaks

The reactor water cleanup (RWCU) extraction line is connected to a recirculation suction line with an additional connection to the vessel bottom head. **[**

]

The RWCU return line is connected to the feedwater line; **[**

]

5.3.8 Shutdown Cooling Line Breaks

The shutdown cooling suction piping is connected to a recirculation suction line and the shutdown cooling return line is connected to a recirculation discharge line. **[**

]

5.3.9 Instrument Line Breaks

[

]

Table 5.1 Available ECCS for Recirculation Line Break LOCAs

Assumed Failure	Systems Remaining* † ‡	Disposition
DC Power (SF-BATT)	3 ADS + 1 LPCS + 2 LPCI	Analyze
LPCI Injection Valve (SF-LPCI)	3 ADS + 2 LPCS + 1 HPCI	Analyze
Diesel Generator (SF-DGEN)	3 ADS + 1 LPCS + 1 HPCI + 2 LPCI	Bounded by SF-BATT
HPCI System (SF-HPCI)	3 ADS + 2 LPCS + 4 LPCI	Bounded by SF-BATT
ADS Valve (SF-ADS)	2 ADS + 2 LPCS + 1 HPCI + 4 LPCI	Analyze

* Systems remaining, as identified in this table for recirculation line breaks, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed for recirculation breaks, less the ECCS in which the break is assumed.

† 2 LPCI refers to two LPCI pumps injecting into the intact loop, 4 LPCI refers to four LPCI pumps injecting into the intact loop.

‡ Loop selection logic directs all available LPCI flow to the intact loop for breaks ≥ 0.4 ft^2. For breaks < 0.4 ft^2, all LPCI flow is directed to the broken loop.

6.0 TLO Recirculation Line Break Spectrum Analyses

The largest diameter recirculation system pipes are the suction line between the reactor vessel and the recirculation pump and the discharge line between the recirculation pump and the riser manifold ring. LOCA analyses are performed for breaks in both of these locations with consideration for both DEG and split break geometries. The break sizes considered included DEG breaks with discharge coefficients from 1.0 to 0.4 and split breaks with areas ranging between the full pipe area and **[ ]** ft^2. As shown in Table 5.1, the limiting single failures considered in the recirculation line break analyses are SF-BATT, SF-LPCI, and SF-ADS.

[

]

6.1 *Break Spectrum Analysis Results*

The break spectrum analyses demonstrate that the recirculation line break case with the **[ ]** is the 1.0 DEG break in the pump suction piping with a single failure of SF-LPCI and a top-peaked axial power shape when operating at 102% rated core power and **[ ]** This case is presented in Table 6.1.

Table 6.2 provides a summary of the **[ ]** from the recirculation line break calculations for each of the single failures, state points, and axial power shapes. The event times for the 1.0 DEG break in the pump suction piping with a single failure of SF-LPCI and a top-peaked axial power shape when operating at 102% rated core power and **[ ]** are presented in Table 6.3 and plots of key parameters from the LOCA analyses of this case are provided in Figures 6.1 – 6.15.

Table 6.1 Break Spectrum Results* for TLO Recirculation Line Breaks

Break spectrum case resulting	1.0 DEG pump suction SF-LPCI Top-peaked axial 102%P/[ ]
[ ]	[ ]

* [ ]

Table 6.2 Summary of Break Spectrum [ ] for TLO Recirculation Line Breaks

[ ]

Table 6.3 Event Times for the [ ] from the TLO Recirculation Line Break Spectrum Analysis

Figure 6.1 [ ] from the TLO Recirculation Line Break Spectrum Analysis Upper Plenum Pressure

[ ]

Figure 6.2 [ ] from the TLO Recirculation Line Break Spectrum Analysis Total Break Flow Rate

[ ]

Figure 6.3 [ ] from the TLO Recirculation Line Break Spectrum Analysis Core Inlet Flow Rate

[ ]

Figure 6.4 [ ] from the TLO Recirculation Line Break Spectrum Analysis ADS Flow

[ ]

Figure 6.5 [ ] from the TLO Recirculation Line Break Spectrum Analysis HPCI Flow

[ ]

Figure 6.6 [ ] from the TLO Recirculation Line Break Spectrum Analysis LPCS Flow

[ ]

Figure 6.7 [ ] from the TLO Recirculation Line Break Spectrum Analysis LPCI Flow

Figure 6.8 [ ] from the TLO Recirculation Line Break Spectrum Analysis RDIV Flows

Figure 6.9 [ ] from the TLO Recirculation Line Break Spectrum Analysis Relief Valve Flow

[ ]

Figure 6.10 [ ] from the TLO Recirculation Line Break Spectrum Analysis Downcomer LOCA Water Level

[ ]

Figure 6.11 [ ] from the TLO Recirculation Line Break Spectrum Analysis Upper Plenum Liquid Level

[ ]

Figure 6.12 [ ] from the TLO Recirculation Line Break Spectrum Analysis Hot Channel Liquid Level

[ ]

Figure 6.13 [ ] from the TLO Recirculation Line Break Spectrum Analysis Core Bypass Liquid Level

[ ]

Figure 6.14 [ ] from the TLO Recirculation Line Break Spectrum Analysis Lower Plenum Liquid Level

[ ]

Figure 6.15 [ ] from the TLO Recirculation Line Break Spectrum Analysis Hot Channel Inlet Flow

[ ]

7.0 Single-Loop Operation LOCA Analysis

During SLO, the pump in one recirculation loop is not operating. A break may occur in either loop, but results from a break in the inactive loop would be similar to those from a two-loop operation break. If a break occurs in the inactive loop during SLO, the intact active loop flow to the reactor vessel would continue during the recirculation pump coastdown period and would provide core cooling similar to that which would occur in breaks during TLO. The system response would be similar to that resulting from an equal-sized break during two-loop operation. A break in the active loop during SLO results in a more rapid loss of core flow and earlier degraded core conditions relative to those from a break in the inactive loop. Therefore, only breaks in the active recirculation loop are analyzed.

A break in the active recirculation loop during SLO will result in an earlier loss of core heat transfer relative to a similar break occurring during two-loop operation. This occurs because there will be an immediate loss of jet pump drive flow. Therefore, fuel rod surface temperatures will increase faster in an SLO LOCA relative to a TLO LOCA. Also, the early loss of core heat transfer will result in higher stored energy in the fuel rods at the start of the heatup. The increased severity of an SLO LOCA can be reduced by applying an SLO multiplier to the two-loop MAPLHGR limit.

7.1 *SLO Analysis Modeling Methodology*

[

]

[ ] SLO is not considered for the EFW region since SLO is not allowed in the EFW region.

7.2 *SLO Analysis Results*

[

]

The SLO analyses are performed with a 0.8 multiplier applied to the two-loop MAPLHGR limit resulting in a maximum SLO MAPLHGR limit of **[ ]** kW/ft. **[ ]** The analyses are performed at maximum stored energy fuel conditions. The limiting SLO LOCA is the 1.0 DEG break in the pump suction piping with a single failure of SF-LPCI and a mid-peaked axial power shape when operating at **[ ]**

A comparison of the limiting SLO and the limiting two-loop results is provided in Table 7.1. The results in Table 7.1 show that the two-loop LOCA results bound the limiting SLO results when a 0.8 multiplier is applied to the two-loop MAPLHGR limit.

[ ]

Table 7.1 Single- and Two-Loop Operation PCT Summary

Operation	Limiting Case	PCT (°F)
Single-loop	1.0 DEG pump suction mid-peaked SF-LPCI	[ ]
Two-loop	1.0 DEG pump suction top-peaked SF-LPCI	[ ]

8.0 Long-Term Coolability

Long-term coolability addresses the issue of reflooding the core and maintaining a water level adequate to cool the core and remove decay heat for an extended time period following a LOCA. For non-recirculation line breaks, the core can be reflooded to the top of the active fuel and be adequately cooled indefinitely. For recirculation line breaks, the core will initially remain covered following reflood due to the static head provided by the water filling the jet pumps to a level of approximately two-thirds core height. Eventually, the heat flux in the core will not be adequate to maintain a two-phase water level over the entire length of the core. Beyond this time, the upper third of the core will remain wetted and adequately cooled by core spray. Maintaining water level at two-thirds core height with one core spray system operating is sufficient to maintain long-term coolability as demonstrated by the NSSS vendor (Reference 6). Since fuel temperatures during long-term cooling are low relative to the PCT and are not significantly affected by fuel design, this conclusion is applicable to ATRIUM 11 fuel. This LOCA analysis assesses conditions from the time of the initiation of the break to the time when long term cooling conditions can be established as demonstrated in Reference 6.

9.0 Exposure-Dependent LOCA Analysis Description and Results

Exposure-dependent LOCA results for ATRIUM 11 fuel are obtained by repeated analyses [ ] from the break spectrum analysis [ ]

Table 9.1 shows the exposure-dependent LOCA analysis results for the ATRIUM 11 fuel. The S-RELAP5 model is applied to obtain these results as described in Section 4.2. The analysis is performed at [ ] which ensures appropriate limits are applied up to the monitored maximum assembly average and rod average exposure limits. The MAPLHGR input is consistent with the data in Figure 2.1. [ ] Exposure-dependent fuel rod data is provided from RODEX4 results [ ] The impact of thermal conductivity degradation is addressed with RODEX4.

The ATRIUM 11 limiting PCT is 2120°F at [ ] exposure for the 1.0 DEG break in the pump suction piping with a single failure of SF-LPCI and a top-peaked axial power shape when operating at 102% rated core power and [ ]. The maximum local MWR of 11.22% occurred at [ ] exposure, [ ] Analysis results show that the hot rod average MWR is 0.97%. Since all other rods in the core are at lower power, the core average metal water reaction (CMWR) will be significantly less.

Figure 9.1 shows the cladding temperature of the ATRIUM 11 PCT rod as a function of time for the limiting PCT result from the exposure-dependent LOCA analysis. The maximum temperature of 2120°F occurs at [ ]. These results demonstrate the acceptability of the ATRIUM 11 MAPLHGR limit shown in Figure 2.1.

Table 9.1 ATRIUM 11 Exposure-Dependent LOCA Analysis Results

CMWR is < 0.97% at all exposures.*

* The rod average MWR for the hot rod is 0.97% which supports the conclusion that the CMWR is less than 0.97%.

Figure 9.1 Limiting [ ] PCT Exposure-Dependent LOCA Analysis

[ ]

10.0 Conclusions

The AURORA-B LOCA Evaluation Model was applied to confirm the acceptability of the ATRIUM 11 MAPLHGR limit and **[** **]** for Monticello. The following conclusions were made from the analyses presented in this report.

- The limiting PCT is obtained from Section 9.0 based on a recirculation line break of 1.0 DEG break in the pump suction piping with a single failure of SF-LPCI and a top-peaked axial power shape when operating at 102% of rated core power and **[** **]**.
- The limiting break analysis identified above satisfies all the acceptance criteria specified in 10 CFR 50.46. The analysis is performed in accordance with 10 CFR 50 Appendix K requirements.
- The multiplier applied to the MAPLHGR limit for SLO is 0.8 for ATRIUM 11 fuel. **[** **]** This multiplier ensures that a LOCA from SLO is less limiting than a LOCA from two-loop operation.
- The acceptance criteria of the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below the ATRIUM 11 MAPLHGR limit given in Figure 2.1 **[** **]**.
  - Peak PCT < 2200°F.
  - Local cladding oxidation thickness < 17%.
  - Total hydrogen generation < 1%.
  - Coolable geometry, satisfied by meeting peak PCT, local cladding oxidation, and total hydrogen generation criteria.
  - Core long-term cooling, satisfied by concluding core flooded to top of active fuel or core flooded to the jet pump suction elevation (Reference 6).
- The MAPLHGR limit and **[** **]** are applicable for ATRIUM 11 full cores as well as transition cores containing ATRIUM 11 fuel.

11.0 References

1. ANP-10332P-A Revision 0, *AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios,* Framatome, March 2019.

2. XN-NF-82-07(P)(A) Revision 1, *Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model*, Exxon Nuclear Company, November 1982.

3. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,* AREVA NP, February 2008.

4. Safety Evaluation by the Office of Nuclear Reactor Regulation, Licensing Topical Report NEDC-33006P, “General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus,” General Electric Hitachi Nuclear Energy America, LLC, October 2008 (ML801130008).

5. Monticello Updated Safety Analysis Report, Revision 37.

6. NEDO-20566A*, General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10CFR50 Appendix K*, September 1986.

Appendix A Limitations from the Safety Evaluation for LTR ANP-10332PA

Compliance to the limitations and conditions from Section 5 of the safety evaluation in ANP-10332PA, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios" (Reference 1) is discussed in the following table.

Appendix A *(Continued)*

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
1	The AURORA-B LOCA evaluation model shall be supported by an approved nodal core simulator and lattice physics methodology. Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall identify the nodal core simulator and lattice physics methods supporting the AURORA-B LOCA analysis and reference an NRC-approved TR confirming their acceptability for the intended application.	MICROBURN-B2 and the underlying cross section generation code, CASMO-4, are used for the nodal core simulator and lattice physics methodology from the following NRC-approved TR: EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2," Siemens Power Corporation, October 1999.
2	The full, stand-alone version of the RODEX4 code shall be used in accordance with an approved methodology to supply steady-state fuel thermal-mechanical inputs to the AURORA-B LOCA evaluation model.	The stand-alone version of RODEX4 is used to supply steady-state fuel thermal-mechanical input in accordance with the following NRC-approved methodology: BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP Inc., February 2008.
3	The AURORA-B LOCA evaluation model may not be used to perform analyses that result in any of its constituent components or supporting codes (i.e., S-RELAP5, RODEX4 kernel, RODEX4, core simulator and lattice physics methods) being operated outside approved limits documented in their respective TRs, SEs, code manuals, and plant-specific licensing applications.	The analyses are within the limits of the TRs, SEs, code manuals and plant-specific licensing applications.
4	TR ANP-10332P **[** **]**.	**[** **]** LOCA report.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
5	As discussed above in Section 2.1, the conclusions of this SE apply only to the use of the AURORA-B LOCA evaluation model for the purpose of demonstrating compliance with relevant regulatory requirements in effect at the time the NRC staff's technical review of ANP-10332P was completed (i.e., as of December 31, 2018).	The analyses only apply regulatory requirements in effect at the time the NRC staff's review was completed. They **[** **]**.
6	This SE does not constitute **[** **]** of the evaluation model.	The evaluation model **[** **]**.
7	**[** **]**.	The **[** **]**.
8	**[** **]**.	The **[** **]** in the analyses.
9	Safety analyses performed with the AURORA-B LOCA evaluation model **[** **]**.	**[** **]**.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
10	To ensure adequate conservatism in future plant-specific safety analyses, absent specific NRC staff approval for higher values, this SE limits **[** **]**.	A **[** **]**.
11	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[** **]**.	BWR fuel rods are **[** **]**.
12	The Appendix K lockout preventing the return to nucleate boiling **[** **]**.	The analyses **[** **]**.
13	**[** **]**.	**[** **]**.
14	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[** **]**.	Analyses **[** **]**.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
15	[ ].	The [ ].
16	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model [ ].	[ ].
17	To assure satisfaction of GDC 35 (or similar plant-specific design criterion), [ ].	[ ]
18	Safety analyses performed with the AURORA-B LOCA evaluation model [ ].	[ ].

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
19	Safety analyses for **[** **]**.	The application of AURORA-B LOCA **[** **]**. **[** **]**. Approximately **[** **]**.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
20	Simulations supporting plant safety analyses **[** **]**.	Simulations **[** **]**.
21	As discussed in Section 3.3.5.7, Framatome used a **[** **]**.	The **[** **]**.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
22	The NRC staff has not specifically reviewed any plant parameters in ANP-10332P or deemed them acceptable for use in plant safety analyses. Therefore, **[** **]**.	The licensee **[** **]**.
23	Safety analyses performed with the AURORA-B LOCA evaluation model shall include justification that **[** **]**.	A **[** **]**.
24	**[** **]**.	**[** **]**.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
25	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[** **]**	The **[** **]**.
26	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[** **]**.	The **[** **]**.
27	As discussed in Section 4.3 of this SE, new or modified Framatome **[** **]**.	The analyses **[** **]**.

ENCLOSURE

ATTACHMENT 12b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3934P REPORT, REVISION 0

MONTICELLO LOCA ANALYSIS FOR ATRIUM 11 FUEL

JULY 2021

(3 pages follow)

AFFIDAVIT

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3934P Revision 0, "Monticello LOCA Analysis for ATRIUM 11 Fuel," dated July 2021 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: July 9, 2021

Alan B. Meginnis

ENCLOSURE

ATTACHMENT 13a

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

ANP-3932NP REPORT, REVISION 0

APPLICATION OF BEO-III METHODOLOGY WITH PERIOD-BASED DETECTION ALGORITHM AT MONTICELLO

JUNE 2021

(25 pages follow)

framatome

Application of BEO-III Methodology with Period-Based Detection Algorithm at Monticello

ANP-3932NP
Revision 0

June 2021

0414-12-F04 (Rev. 004, 04/27/2020)

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

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Nomenclature

Acronym	Definition
2RPT	Two-Pump Recirculation Pump Trip
AOO	Anticipated Operational Occurrences
APRM	Average Power Range Monitor
BEO-III	Best-estimate Enhanced Option III
BSP	Backup Stability Protection
BWROG	Boiling Water Reactor Owners Group
CHDR	Channel Decay Ratio
D&S	Detect and Suppress
DR	Decay Ratio
EFW	Extended Flow Window
EM	Evaluation Model
F/I MCPR	Ratio of Final to Initial Minimum Critical Power Ratio
FoM	Figure of Merit
FWHOOS	Feedwater Heater Out Of Service
ICO	Independent Channel Oscillations
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
OLMCPR	Operating Limit Minimum Critical Power Ratio
OPRM	Oscillation Power Range Monitor
PBDA	Period Based Detection Algorithm
SAFDL	Specified Acceptable Fuel Design Limit
SLMCPR	Safety Limit Minimum Critical Power Ratio

Acronym	Definition
SLO	Single Loop Operation
USNRC	United States Nuclear Regulatory Commission
ΔCPR	Delta Critical Power Ratio

ABSTRACT

This report presents a plant-specific Best-estimate Enhanced Option III (BEO-III) evaluation for Monticello using the approved methodology from Reference 1. The method establishes the cycle-specific operating limit Minimum Critical Power Ratio (OLMCPR) based on statistical analyses of pump trip scenarios and evaluation of the time dependent local power range monitors (LPRMs) and core minimum critical power ratio (MCPR) to determine the most limiting event based on the Monticello Period Based Detection Algorithm (PBDA) detect and suppress (D&S) hardware response.

This report provides the plant-specific BEO-III evaluation results for Monticello. Backup Stability Protection (BSP) is discussed in Section 2.0. Section 3.0 presents the validation of RAMONA5-FA against Monticello specific data and Section 4.0 presents the ATRIUM 11 Equilibrium Cycle Sample Application. Finally Section 5.0 documents the compliance of the methodology to the NRC Limits and Conditions associated with the approved BEO-III methodology.

1.0 Introduction

The Best-estimate Enhanced Option-III methodology (BEO-III) was approved in Reference 1. Monticello utilizes the PBDA as the primary stability protection feature, which is the approved algorithm used in the BEO-III methodology. The PBDA algorithm tracks the number of confirmation counts, or successive oscillations which have a characteristic period which is within a small tolerance compared to the average of the previous periods. If the number of confirmation counts and the peak magnitude of the oscillation exceed the trip setpoints, the OPRM cell is tripped. If any of the OPRM cells associated with an OPRM channel is tripped, the OPRM channel is tripped. For Monticello, if two out of four OPRM channels are tripped, then a reactor trip occurs and the oscillations are suppressed by control rod insertion.

The calculation procedure for the approved BEO-III methodology is presented in Section 7.2 of Reference 1. This report presents a sample plant-specific analysis for Monticello following the approved BEO-III methodology. The sample analysis establishes the cycle-specific OLMCPR based on statistical analyses of pump trip scenarios and evaluation of the time dependent LPRMs and core MCPR to determine the most limiting event based on the Monticello PBDA D&S hardware response.

2.0 Backup Stability Protection

When the OPRM system is declared inoperable, Backup Stability Protection will be provided in accordance with Reference 2 as described in Section 7.3 of Reference 1. The resultant manual BSP exclusion regions on the power/flow map and associated operator guidance will be employed.

3.0 RAMONA5-FA Qualification for Monticello

During the Long Term Stability Solution Option III Implementation testing, Xcel collected OPRM data for Monticello between May 11, 2009 and May 16, 2009 (Cycle 25). The testing was performed prior to the EPU power uprate. The OPRM testing was performed at three power/flow operating state-points (82.92% power / 64.67% flow, 53.20% power / 50.61% flow, and 28.16% power / 37.33% flow). The operating state-points were converted to the current rated power and flow bases for Monticello. RAMONA5-FA statistical calculations of the OPRM amplitudes were performed and the calculated results were compared to the measured amplitudes. **[**

]

4.0 ATRIUM 11 Equilibrium Cycle Sample Application

The BEO-III methodology described in Reference 1 was applied to a Monticello equilibrium ATRIUM 11 core design. The analyses demonstrate that the PBDA hardware installed at the Monticello reactors can detect and suppress oscillations with a high confidence level for the ATRIUM 11 fuel design. All of the full power cycle exposures in the equilibrium cycle design depletion were assessed **[** **]** consistent with Section 7.2 of Reference 1.

The sampled parameters for the statistical analysis are consistent with those defined in Reference 1 and are presented in Table 4-1. As defined in Reference 1, there are **[** **]**

For this Monticello sample application, the analyses in this section are performed using a setpoint of **[** **]** These are representative values for the PBDA setpoints. The values used for licensing of actual core designs will be specified by the licensee.

4.1 *Limiting EFW 2RPT Scenario*

For the limiting EFW state point at rated core power and 80% of rated core flow, a sample size of **[ ]** trials was selected as a typical value from Table 7-1 of Reference 1. The statistical results for the core MCPR FoM are presented graphically in Figure 4-1. Based on a **[ ]** OLMCPR, the 95/95 MCPR of **[ ]** is shown to be well above the **[**

].

While the **[**

]

Therefore, ICO do not invalidate the assumption that the reactor protection system can detect and suppress the oscillations prior to violation of the specified acceptable fuel design limits.

4.1.1 Evaluation of Pump Coastdown

The pump coastdown can potentially impact the FoM by causing a reset in the PBDA confirmation counts that results in a delayed reactor trip. Given the PBDA setpoint of **[**

]

Table 4-1
Sampled Parameters for BEO-III ATRIUM 11 Statistical Analyses

Figure 4-1 Limiting EFW MCPR

Figure 4-2 Limiting EFW [ ]

4.2 *Potentially Limiting Scenarios*

The other pump trip scenarios identified in Section 7.2 of Reference 1 are the lowest core flow at rated core power within the MELLLA domain with the minimum feedwater temperature allowed by the licensee for the feedwater heater out-of-service (FWHOOS) or Final Feedwater Temperature Reduction (FFTR) scenario and the highest core power under single loop operation (SLO). Since Monticello is not licensed for any reduced feedwater temperature scenarios and the base case analysis from Section 4.1 conservatively considered the minimum feedwater temperature from the normal feedwater temperature range, the only potentially limiting scenario is the pump trip from SLO conditions. This analysis used **[ ]** trials in the statistical analysis consistent with the analysis in Section 4.1.

The limiting single pump trip under SLO was analyzed at 66.0% of rated power and 52.5% of rated flow. The statistical results for the core MCPR FoM are presented in Figure 4-3 for this state point. Based on an assumed **[ ]** power-dependent OLMCPR, the core MCPR FoM of **[**

]

Figure 4-3 SLO MCPR

Figure 4-4 SLO [ ]

4.3 *Tmin Confirmation*

The Monticello plant-specific D&S hardware specifies a minimum oscillation period of **[ ]** seconds. The EFW 2RPT scenario yielded a minimum period, Tmin, of **[ ]** seconds. The 1PT from the EFW corner produced **[**

]

The statistical analysis for the **[ ]** statepoint was repeated **[**

] The 95/95 minimum period was **[ ]** seconds. Figure 4-5 presents the oscillation period for each of the EFW 2RPT trials, and the limiting trial for the EFW 1RPT scenario at each exposure **[ ]** Based on these analyses the Tmin value of **[ ]** seconds is confirmed to bound the minimum oscillation period for any credible oscillations.

The use of **[**

] and is not required for any future cycle specific confirmation calculations.

Figure 4-5 EFW Pump Trip Calculation Periods

5.0 Compliance with ANP-10344P-A Limitations and Conditions

The Safety Evaluation for the Reference 1 topical report lists eight limitations and conditions that applications of the methodology must satisfy. The following section discusses how the plant specific methodology and analysis in this report comply with those limitations and conditions.

Limitation and Condition 1

> *MICROBURN-B2 is an integral component in the BEO-III methodology. Application of a new core simulator requires review and approval by the NRC.*

MICROBURN-B2 is the specified core simulator for the Monticello Application Methodology.

Limitation and Condition 2

> *Selected settings and modeling options, including core and vessel nodalization and time step control parameters, shall be defined consistently with the validation basis presented in Section 6.0.*

All settings and modeling options, including core and vessel nodalization and time step control parameters are consistent with the validation basis.

Limitation and Condition 3

[

]

[

]

Limitation and Condition 4

[

]

The **[**

] for protecting the SAFDL.

Limitation and Condition 5

> *Framatome must continue to use existing regulatory processes for any code modifications made to the RAMONA-5FA code. The existing regulatory processes do not allow changes to the RAMONA5-FA code that would substantively alter the BEO-III methodology, as described in ANP-10344P and supporting RAI responses, which the NRC staff relied upon as the basis for the finding of acceptability in this SE, without prior NRC review and approval.*

The RAMONA5-FA code utilized for the Monticello Application Methodology was evaluated as per existing regulatory processes. There are no changes that would alter the basis of the methodology.

Limitation and Condition 6

> *Plant-specific applications shall justify whether the recirculation pump coastdown behavior will have a significant impact on the final MCPR for the specific plant and conditions being analyzed. If so, the uncertainties in the recirculation pump coastdown response should be included in the statistical analyses or otherwise accounted for.*

The concern associated with this limitation and condition is that variations in the pump coastdown rate could lead to system resets that would cause the reactor trip to be delayed and allow the amplitude to grow to larger magnitudes. The assessment of **[**

] on the final MCPR for the ATRIUM 11 equilibrium cycle.

Limitation and Condition 7

> *If the 1RPT EFW event remains stable, additional analyses are required using* ***[*** ***]*** *to ensure that the lowest oscillation period remains above Tmin under any anticipated conditions.*

As presented in Section 4.3, the 1RPT from the minimum core flow at rated power in the EFW domain **[**

]

Limitation and Condition 8

> *After applying the* ***[***
>
> ***]*** *If trends are observed which indicate that the most limiting exposure point(s) may be outside the analyzed range of exposures, additional exposure points should be analyzed until reasonable assurance is attained that the limiting exposure point is analyzed*.

The **[** **]** Review of the FoM trend as a function of exposure indicates that there is a reasonable assurance that the limiting exposure points were analyzed.

6.0 References

1. ANP-10344P-A Revision 0 *Framatome Best-estimate Enhanced Option III Methodology*, Framatome Inc., March 2021.

2. OG02-0119-260, Backup Stability Protection (BSP) for Inoperable Option III Solution, GE Nuclear Energy, July 17, 2002.

ENCLOSURE

ATTACHMENT 13b

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST
APPLICATION TO ADOPT ADVANCED FRAMATOME METHODOLOGIES

AFFIDAVIT FOR

ANP-3932P REPORT, REVISION 0

APPLICATION OF BEO-III METHODOLOGY WITH PERIOD-BASED DETECTION ALGORITHM AT MONTICELLO

JUNE 2021

(3 pages follow)

AFFIDAVIT

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3932P Revision 0, "Application of BEO-III Methodology with Period-Based Detection Algorithm at Monticello," dated June 2021 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: June 4, 2021

Alan B. Meginnis