Enclosure 1 Contains Proprietary Information –
Withhold in Accordance with 10 CFR 2.390

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Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10 CFR 50.4

SUSQUEHANNA STEAM ELECTRIC STATION
SUBMITTAL OF UNIT 2 CYCLE 21 RELOAD
SAFETY ANALYSIS REPORT
PLA-7929

Docket No. 50-387
and 50-388

Reference:

1) Susquehanna letter to NRC, "Proposed Amendment to Licenses NPF-14 and NPF-22: Application of Advanced Framatome Methodologies and TSTF-535 (PLA-7783)," dated July 15, 2019 (ADAMS Accession No. ML19196A270).

2) NRC letter to Susquehanna, "Issuance of Amendment Nos. 278 and 260 to Allow Application of Advanced Framatome Atrium 11 Fuel Methodologies (EPID L-2019-LLA-0153)," dated January 21, 2021 (ADAMS Accession Nos. ML20164A181 (Proprietary) and ML20168B004 (Non-Proprietary)).

3) Susquehanna letter to NRC, "Submittal of Unit 2 Cycle 21 Reload Licensing Documents to Support License Amendment Requesting Application of Advanced Framatome Methodologies (PLA-7861)," dated April 9, 2020 (ADAMS Accession No. ML20100F921).

4) Susquehanna letter to NRC, "Submittal of Unit 2 Cycle 21 Safety Limit Minimum Critical Power Ratio Report to Support License Amendment Requesting Application of Advanced Framatome Methodologies (PLA-7881)," dated July 30, 2020 (ADAMS Accession No. ML20212L631).

5) Susquehanna letter to NRC, "Submittal of Unit 2 Cycle 21 Fuel Rod Design Report to Support License Amendment Requesting Application of Advanced Framatome Methodologies (PLA-7910)," dated December 10, 2020 (ADAMS Accession No. ML20345A170).

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), submitted, in Reference 1, a request for an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License numbers NPF-14 and NPF-22. The proposed amendment would revise TS 5.6.5.b to allow application of Advanced Framatome Methodologies for determining core operating limits in support of loading Framatome fuel type ATRIUM 11, revise the low pressure safety limit in TS 2.1.1.1 and TS 2.1.1.2, and remove the neutronic methods penalties on Oscillation Power Range Monitor amplitude setpoint and the pin power distribution uncertainty and bundle power correlation coefficient. The NRC approved the requested amendment in Reference 2.

In Enclosure 7 to Reference 1, Susquehanna committed to provide certain reload licensing documents related to Unit 2, Cycle 21 (i.e., the first cycle expected to be loaded with ATRIUM 11 fuel), within 15 days of their approval. References 3, 4, and 5 provided earlier reports upon their approval. This letter provides the final Unit 2 report: Enclosure 1 provides the Unit 2, Cycle 21, Reload Safety Analysis Report. The report is submitted for information only.

Information provided in Enclosure 1 is considered proprietary to Framatome. The proprietary information has been denoted therein by brackets. As owners of the proprietary information, Framatome has executed an affidavit for the document which identifies the information as proprietary, is customarily held in confidence, and should be withheld from public disclosure in accordance with 10 CFR 2.390. Enclosure 2 provides a non-proprietary version of Enclosure 1. The Framatome affidavit is included as Enclosure 3.

There are no new or revised regulatory commitments contained in this submittal. This submittal satisfies Regulatory Commitment 7783-5 as documented in Enclosure 7 to Reference 1.

Should you have any questions regarding this submittal, please contact Ms. Melisa Krick, Manager – Nuclear Regulatory Affairs, at (570) 542-1818.

K. Cimorelli

Enclosures:

1. Framatome Report ANP-3884P, Revision 1, “Susquehanna Unit 2 Cycle 21 Reload Safety Analysis,” **[Proprietary Information – Withhold from Public Disclosure in accordance with 10 CFR 2.390]**
2. Framatome Report ANP-3884NP, Revision 1, “Susquehanna Unit 2 Cycle 21 Reload Safety Analysis,” (Non-Proprietary Version)
3. Framatome Affidavit for ANP-3884P, Revision 1, “Susquehanna Unit 2 Cycle 21 Reload Safety Analysis”

Copy: NRC Region I
Mr. C. Highley, NRC Senior Resident Inspector
Ms. S. Goetz, NRC Project Manager
Mr. M. Shields, PA DEP/BRP (w/out Enclosure 1)

Enclosure 2 of PLA-7929

Framatome Report
ANP-3884NP, Revision 1

Susquehanna Unit 2 Cycle 21 Reload Safety Analysis

(Non-Proprietary Version)

Susquehanna Unit 2 Cycle 21 Reload Safety Analysis

ANP-3884NP
Revision 1

February 2021

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1.	1-2, 5-9, 5-11, 5-26, 5-28, 8-5 – 8-9, 8-11 – 8-13	Update maximum power level associated with one TCV/TSV closed from 70% to 75% of rated power.
2.	5.1.4.4	Added text regarding applicability of base case limits to one TCV/TSV closed up to 75% of rated power.

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Nomenclature

Acronym	Definition
AOO	anticipated operational occurrence
ARO	all rods out
ARTS	average power range monitor, rod block monitor and technical specification improvement program
ATWS	anticipated transient without scram
BOC	beginning of cycle
BPWS	banked position withdrawal sequence
BWR	boiling-water reactor
BWROG	Boiling-Water Reactor Owner's Group
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPR	critical power ratio
CRDA	control rod drop accident
CRWE	control rod withdrawal error
DIVOM	delta over initial CPR value versus oscillation magnitude
EFPH	effective full-power hours
EHA	equipment handling accident
EM	evaluation model
EOC	end of cycle
EOC-RPT	end of cycle recirculation pump trip
EOFP	end of full-power
EOOS	equipment out-of-service
EPU	extended power uprate
FHA	fuel handling accident
FoM	figure of merit
FSAR	Final Safety Analysis Report
FWCF	feedwater controller failure
FWT	feedwater temperature
HPCI	high-pressure coolant injection
ICF	increased core flow
ICPR	initial critical power ratio
IHPCIS	inadvertent startup of the HPCI pump

Nomenclature

(Continued)

Acronym	**Definition**
LEOFP	licensing basis end of full-power
LFWH	loss of feedwater heating
LHGR	linear heat generation rate
LHGRFAC	linear heat generation rate factor
$LHGRFAC_f$	flow-dependent linear heat generation rate factor
$LHGRFAC_p$	power-dependent linear heat generation rate factor
LOCA	loss-of-coolant accident
LPRM	local power range monitor
LRNB	load rejection without bypass
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
$MCPR_f$	flow-dependent minimum critical power ratio
$MCPR_p$	power-dependent minimum critical power ratio
MELLLA	maximum extended load line limit analysis
MSIV	main steam isolation valve
NRC	Nuclear Regulatory Commission, U.S.
OLMCPR	operating limit minimum critical power ratio
OOS	out-of-service
OPRM	oscillation power range monitor
P_{bypass}	lowest power at which direct scram and RPT on TSV position or TCV fast closure are applicable
PAPT	protection against power transient
PCT	peak clad temperature
PHE	peak hot excess reactivity
PRFDS	pressure regulator failure – downscale
PRFO	pressure regulator failure – open
PRM	power range monitor
PROOS	pressure regulator out-of-service
RBM	rod block monitor
RPT	recirculation pump trip
RPTOOS	EOC-RPT out-of-service
SLC	standby liquid control
SLMCPR	safety limit minimum critical power ratio
SLO	single-loop operation
SRV	safety/relief valve
SRVOOS	safety/relief valve out-of-service
SSES	Susquehanna Steam Electric Station

Nomenclature

(Continued)

Acronym	Definition
TBV	turbine bypass valve
TBVOOS	turbine bypass valve out-of-service
TCV	turbine control valve
TIP	traversing incore probe
TLO	two-loop operation
TSV	turbine stop valve
TTNB	turbine trip without bypass
ΔCPR	change in critical power ratio
ΔMCPR	change in minimum critical power ratio

1.0 INTRODUCTION

Reload licensing analyses results generated by Framatome Inc. (Framatome) are presented in support of Susquehanna Unit 2 Cycle 21. The analyses reported in this document were performed using methodologies previously approved for generic application to boiling water reactors and demonstrated in Reference 1 to be applicable for ATRIUM 11 fuel operating in the Extended Power Uprate (EPU) and maximum extended load line limit analysis (MELLLA) operating domain. The NRC technical limitations associated with the application of the approved methodologies have been satisfied by these analyses.

The Cycle 21 core consists of a total of 764 fuel assemblies, including 300 fresh SUS2-21 ATRIUM 11 assemblies, 308 once-burned SUS2-20 ATRIUM-10 assemblies, and 156 twice-burned SUS2-19 ATRIUM-10 assemblies. The licensing analysis supports the core design presented in Reference 2.

The Cycle 21 reload licensing analyses were performed for the potentially limiting events and analyses that were identified in the disposition of events. The results of the analyses are used to establish the Technical Specifications/COLR limits and ensure that the design and licensing criteria are met. The design and safety analyses are based on the design and operational assumptions and plant parameters provided by the utility (References 3 and 4). The results of the reload licensing analysis support operation for the power/flow map presented in Figure 1.1. This reload licensing also supports operation with the equipment out-of-service (EOOS) scenarios presented in Table 1.1.

Table 1.1 EOOS Operating Conditions*

Turbine bypass valves out-of-service (TBVOOS)
End of cycle recirculation pump trip out-of-service (RPTOOS)
One pressure regulator out-of-service (PROOS)
One turbine control valve or turbine stop valve closed (1 TCV/TSV closed)†
Single-loop operation (SLO)‡
Two safety/relief valves out-of-service (SRVOOS)
Up to 42% of the TIP channels out-of-service (100% available at startup)
Up to 50% of the LPRMs out-of-service

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* Each EOOS condition is supported in combination with SLO, 2 SRVOOS, up to 42% of the TIP channels out-of-service, and/or up to 50% of the LPRMs out-of-service.

† Operation with one TCV/TSV closed is supported at power levels ≤ 75% of rated.

‡ SLO is supported up to a maximum power level of 2,652 MWt.

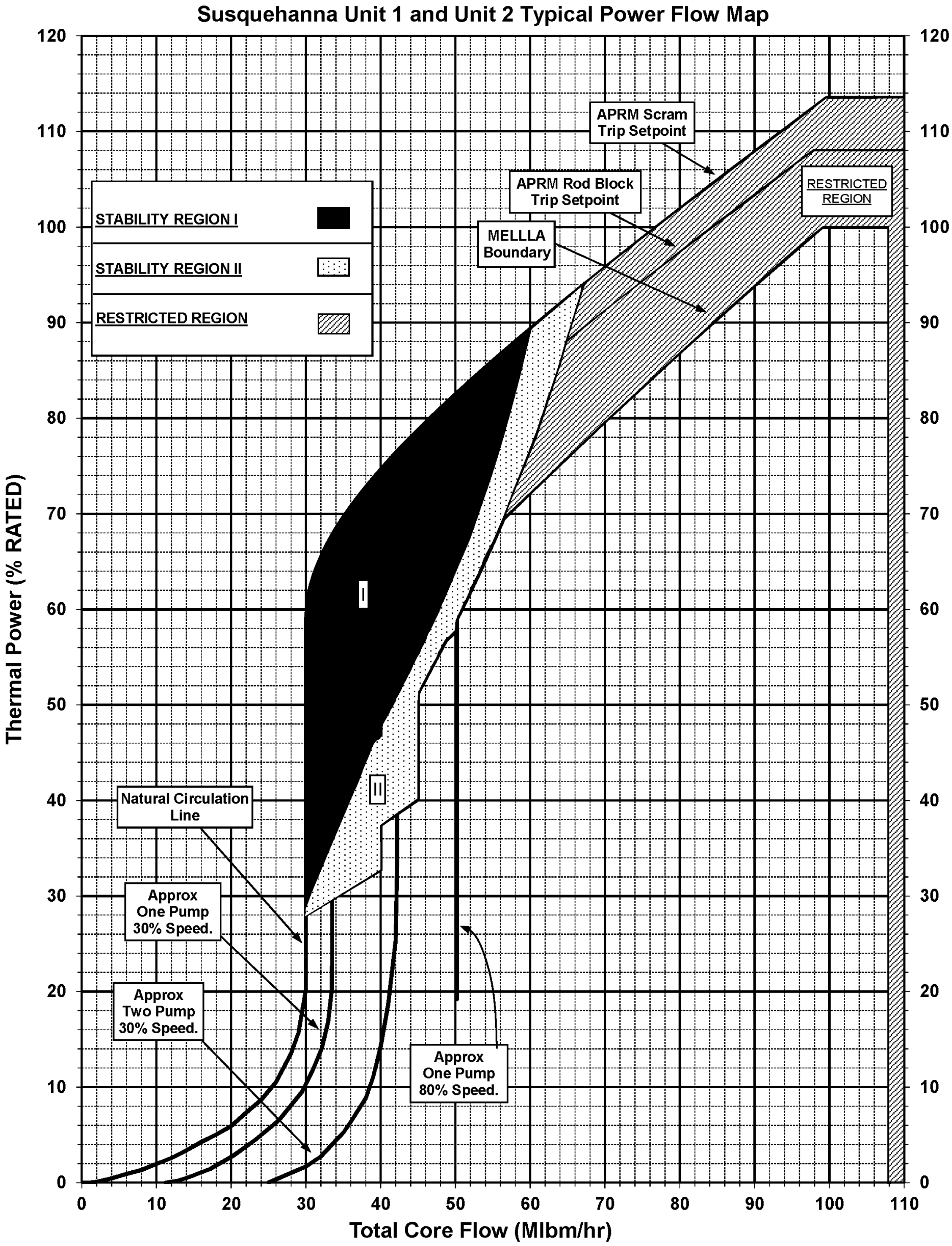

Figure 1.1 Susquehanna Unit 2 Power/Flow Map

2.0 DISPOSITION OF EVENTS AND PLANT MODELING SENSITIVITIES

2.1 *Disposition of Events for ATRIUM 11 Fuel Introduction*

A disposition of events to identify the limiting events which need to be analyzed to support operation at the Susquehanna Steam Electric Station (SSES) was performed for the introduction of ATRIUM 11 fuel. Events and analyses identified as potentially limiting were either evaluated generically for the introduction of ATRIUM 11 fuel or are performed on a cycle-specific basis.

The first step in the disposition of events 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 which must be met to ensure regulatory compliance and safe operation are also included. The SSES licensing basis is contained in the Final Safety Analysis Report (FSAR), the Technical Specifications, Core Operating Limits Reports (COLR), and other reload analysis reports.

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

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

  - 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.

The impacts of EOOS scenarios were also considered in the disposition of events.

A summary of the disposition of events is presented in Table 2.1 through Table 2.4. Table 2.1, Table 2.2, and Table 2.3 present a list of the events and analyses, the corresponding FSAR section, the disposition status, and any applicable comments. Table 2.4 presents a summary of the disposition of events for the EOOS scenarios. Note that operation in the increased core flow (ICF) and MELLLA regions of the power/flow map are included in the disposition results presented in Table 2.1, Table 2.2, and Table 2.3.

Table 2.1 Disposition of FSAR Chapter 15 Events Summary for ATRIUM 11 Fuel Introduction at Susquehanna

FSAR Section	Event Name	Disposition Status	Comments
15.1.1	Loss of Feedwater Heater (LFWH)	Address each reload	Potentially limiting AOO.
15.1.2	Feedwater Controller Failure (FWCF) - Maximum Demand	Address each reload	Potentially limiting AOO.
15.1.3	Pressure Regulator Failure Open (PRFO)	Address for initial reload.	Consequences of this event, relative to AOO thermal operating limits are non-limiting. If the core voiding due to the depressurization rate is large enough, the sensed vessel water level trip set point (L8) may be reached initiating a turbine and feedwater pump trip early in the transient. Turbine trip will initiate reactor scram and shut down the reactor. Thermal margins will be better than a typical turbine trip event because of the power reduction initially experienced due to increased core voids in this event. This event has been evaluated to ensure the current low MSIV pressure setpoint of 825 psig will protect the lower ACE/ATRIUM 11 pressure bound during the depressurization event.
15.1.4	Inadvertent Safety/Relief Valve Opening	No further analysis required	The event causes a mild depressurization. Thermal margins decrease only slightly through the transient and no fuel damage results from the transient. Consequences of this event are non-limiting.
15.1.5	Spectrum of steam system piping failures inside and outside containment in a PWR	Not applicable	Not applicable to BWR plants.
15.1.6	Inadvertent RHR Shutdown Cooling Operation	No further analysis required	This event is not considered credible for power operation.
15.2.1	Pressure Regulator Failure - Closed	Address each reload	Consequences of this event, relative to one pressure regulator out-of-service (PROOS) may be limiting; therefore, this equipment out-of-service will be addressed each cycle.

Table 2.1 Disposition of FSAR Chapter 15 Events Summary for ATRIUM 11 Fuel Introduction at Susquehanna *(Continued)*

FSAR Section	Event Name	Disposition Status	Comments
15.2.2	Generator Load Rejection with and without bypass (LRNB)	Address each reload	This event without bypass operable is a potentially limiting AOO. Load rejection with bypass operable is bound by the load rejection without bypass. The LRNB and TTNB events can be combined into a single event with the turbine control valves and turbine stop valves being assumed to start their closure at the same time. This assumption can provide a single analysis with consequences that are equal to or bounding of both events.
15.2.3	Turbine Trip with and without bypass (TTNB)	Address each reload	This event without bypass operable is a potentially limiting AOO. Turbine trip with bypass operable is bound by the turbine trip with no bypass. The LRNB and TTNB events can be combined into a single event with the turbine control valves and turbine stop valves being assumed to start their closure at the same time. This assumption can provide a single analysis with consequences that are equal to or bounding of both events.
15.2.4	MSIV Closures	No further analyses required	Consequences of this event, relative to thermal operating limits, are bound by the generator load rejection and turbine trip events which have much faster valve closure times (FSAR Sections 15.2.2 and 15.2.3, respectively).
15.2.5	Loss of Condenser Vacuum	No further analyses required	This transient is similar to a normal turbine trip with bypass operable. Consequences of this event are bound by the turbine trip event (FSAR Section 15.2.3).
15.2.6	Loss of AC Power	No further analysis required	The loss of AC power long-term water level response is bound by the loss of feedwater flow event (FSAR Section 15.2.7). If complete connection with the external grid is lost, the reactor will experience a generator load rejection. The coastdown of the recirculation pumps initiated at time = 0 will reduce the severity of the event, compared to the generator load rejection event, by reducing core power. Therefore the consequences of this event are bound by the LRNB event (FSAR Section 15.2.2).

Table 2.1 Disposition of FSAR Chapter 15 Events Summary for ATRIUM 11 Fuel Introduction at Susquehanna *(Continued)*

FSAR Section	Event Name	Disposition Status	Comments
15.2.7	Loss of Feedwater Flow	No further analyses 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 above L1. The MSIVs do not close and the system pressure remains low. Either the HPCI or RCIC system is capable of maintaining adequate core coverage and will provide inventory control. The long term water level transient is dependent upon the decay heat which is **[** **]**. This is a benign event.
15.2.8	Feedwater Line Break	No further analyses required	See FSAR 15.6.6.
15.2.9	Failure of RHR Shutdown Cooling	No further analysis required	Consequences of this event are benign.
15.3.1	Recirculation Pump Trip	No further analyses required	For the single pump trip event, MCPR remains approximately at the operating limit, thus the fuel thermal margins are not violated. Level swell is not sufficient to cause turbine trip and scram. This is a benign event. The two pump trip does not directly initiate a scram. The vessel swell due to the rapid flow coastdown is expected to reach the high level trip thereby shutting down the main turbine and feed pump turbines, and indirectly initiating scrams as a result of the main turbine trip. Thus this event is bounded by the turbine trip with no bypass event (FSAR Section 15.2.3).
15.3.2	Recirculation Flow Control Failure - Decreasing Flow	No further analyses required	The consequences of this event cannot be more severe than the single or two Recirculation Pump Trip events (FSAR Section 15.3.1).
15.3.3	Recirculation Pump Seizure	Address each reload	While this event is classified as an accident, it will be conservatively analyzed using the AOO acceptance criteria.
15.3.4	Recirculation Pump Shaft Break	No further analyses required	The consequences of this accident are bounded by the effects of the pump seizure event (FSAR Section 15.3.3).

Table 2.1 Disposition of FSAR Chapter 15 Events Summary for ATRIUM 11 Fuel Introduction at Susquehanna *(Continued)*

FSAR Section	Event Name	Disposition Status	Comments
15.4.1	Rod Withdrawal Error – Low Power	No further analysis required	Consequences of this event are benign.
15.4.2	Rod Withdrawal Error – At Power	Address each reload	This event is a potentially limiting AOO.
15.4.3	Control Rod Maloperation (System Malfunction or Operator Error)	No further analyses required	This event is addressed in FSAR Sections 15.4.1 and 15.4.2.
15.4.4	Abnormal Startup of Idle Recirculation Pump	No further analyses required	Consequences of this event are non-limiting.
15.4.5	Recirculation Flow Controller Failure with Increasing Flow	Address each reload	The slow run-up event determines the $MCPR_f$ limits.
15.4.6	Chemical and Volume Control System Malfunctions	Not applicable	Not applicable to BWRs.
15.4.7	Misplaced Bundle Accident	Address each reload	Potentially limiting accident.
15.4.8	Spectrum of Rod Ejection Assemblies	Not applicable	Not applicable to BWRs.
15.4.9	Control Rod Drop Accident (CRDA)	Address each reload	Potentially limiting accident.
15.5.1	Inadvertent HPCI Startup	Address each reload	This event is a potentially limiting AOO.
15.5.2	Chemical Volume Control System Malfunction (or Operator Error)	Not applicable	Not applicable to BWRs.
15.5.3	BWR Transients Which Increase Reactor Coolant Inventory	No further analysis required	These events are discussed in FSAR Sections 15.1 and 15.2.
15.6.1	Inadvertent Safety Relief Valve Opening	No further analysis required	This event is discussed in FSAR Section 15.1.4.
15.6.2	Instrument Line Break	No further analysis required	This event is bound by the consequences of the limiting LOCA.
15.6.3	Steam Generator Tube Failure	Not applicable	Not applicable to BWRs.

Table 2.1 Disposition of FSAR Chapter 15 Events Summary for ATRIUM 11 Fuel Introduction at Susquehanna *(Continued)*

FSAR Section	Event Name	Disposition Status	Comments
15.6.4	Steam System Piping Break – Outside Containment	No further analysis required	The main steam line break is considered in the identification of the spectrum of loss of coolant accident events and is bounded by the limiting loss of coolant accident scenario (FSAR Section 15.6.5).
15.6.5	Loss-of-Coolant Accidents – Inside Containment	Address for initial reload	Potentially limiting accident.
15.6.6	Feedwater Line Break – Outside Containment	No further analysis required	The feedwater line break is considered in the identification of the spectrum of loss of coolant accident events and is bounded by the limiting loss of coolant accident scenario (FSAR Section 15.6.5).
15.7.1	Gaseous Radwaste System Leak or Failure	No further analysis required	Core and system performance bound by other events.
15.7.2	Liquid Radwaste System Failure	No further analysis required	Bound by other events.
15.7.3	Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure	No further analysis required	This event does not affect the core or NSSS safety performance.
15.7.4	Fuel and Equipment Handling Accidents	Address for initial reload	Consequences of the refueling accident are evaluated to confirm the acceptance criteria are satisfied.
15.7.5	Spent Fuel Cask Drop Accident	No further analysis required	This event is not credible for a single failure.
15.8	Anticipated Transient Without Scram	Address each reload	ATWS overpressurization analyses need to be addressed each reload. **[** **]** (Reference 1). Peak cladding temperature and oxidation are bound by LOCA.

Table 2.1 Disposition of FSAR Chapter 15 Events Summary for ATRIUM 11 Fuel Introduction at Susquehanna *(Continued)*

FSAR Section	Event Name	Disposition Status	Comments
15.9	Station Blackout	No further analysis required	[ ]

Table 2.2 Disposition of Additional Items and Special Events Summary for ATRIUM 11 Fuel Introduction at Susquehanna

FSAR Section	Description	Disposition Status	Comments
---	Backup Stability Protection (BSP)	Address each reload	Analyses are required to support operation when the OPRM system is inoperable.
4.3	Control Blade Shutdown Margin	Address each reload	Fuel and core design dependent.
4.4.1.5 4.4.2.2 4.4.2.9	MCPR Safety Limit	Address each reload	Analyses are performed to confirm or establish the SLMCPR each cycle.
4.4.4.6	Thermal-Hydraulic Stability Analysis	Address each reload	A delta critical power ratio over initial critical power ratio versus oscillation magnitude (DIVOM) calculation is performed on a cycle specific basis. In addition, the CPR response to a two recirculation pump trip calculated by Framatome is used by Susquehanna to perform an OPRM setpoint calculation on a cycle-specific basis.
5.2.2	Overpressure Protection	Address each reload	ASME overpressurization analysis is required each reload.
9.3.5	Standby Liquid Control System	Address each reload	Analysis performed each reload to verify adequate SLC shutdown capacity.
9.5.1	Appendix R Fire Safe Shutdown	Address for initial reload	Susquehanna has elected to address this item.

Table 2.3 Disposition of License and Safety Design Basis Summary for ATRIUM 11 Fuel Introduction at Susquehanna

FSAR Section	Criteria	Disposition Status	Comments
4.2	Fuel System Design	Address for initial reload	Reference 5.
4.3	Nuclear Design	Address each reload	References 2 and 6.
4.4	Thermal and Hydraulic Design	Address each reload	References 7 and 8 as well as this report.
6.2.5	Combustible Gas Control in Containment	No further analysis needed	A design basis LOCA hydrogen release is no longer defined in 10 CFR 50.44 or Regulatory Guide 1.7 Revision 3. Therefore, LOCA hydrogen production is governed by 10 CFR 50.46 (see FSAR Section 15.6.5).
9.1.1	New Fuel Storage	Address for initial reload	Reference 9.
9.1.2	Spent Fuel Storage	Address for initial reload	Reference 10.

Table 2.4 Disposition of Operating Flexibility and EOOS Options on Limiting Events

Option	Affected Event Name	Comments
Backup Pressure Regulator Out-of-Service (PROOS)	PRFDS	Pressure regulator failure –closed analyses with PROOS will be evaluated each reload.
EOC-RPT Inoperable (RPTOOS)	FWCF LRNB TTNB Recirculation Pump Seizure IHPCIS	The FWCF with RPTOOS will be evaluated each reload. The combined LRNB/TTNB event with RPTOOS will be evaluated each reload. If a high water level turbine trip occurs during the pump seizure event, RPTOOS will need further analysis. If a turbine trip occurs in the base case IHPCIS analysis, RPTOOS will need further analysis.
Turbine Bypass Out-of-Service (TBVOOS)	FWCF Recirculation Pump Seizure Slow flow run up IHPCIS	The FWCF with TBVOOS will be evaluated each reload. If a high water level turbine trip occurs during the pump seizure event, TBVOOS will need further analysis. Slow flow run up with TBVOOS will be evaluated each reload. If a turbine trip occurs in the base case IHPCIS analysis, TBVOOS will need further analysis.
One TCV/TSV closed	FWCF Slow flow run up	FWCF analyses will be performed for the initial reload to determine the maximum power level supported for this condition. Slow flow run up with 1 TCV/TSV closed will be evaluated each reload.
Single Loop Operation	Recirculation Pump Seizure LOCA SLMCPR	SLO pump seizure event is a potentially limiting event relative to AOO acceptance criteria and will be evaluated each cycle. The impact of SLO should be addressed for the initial cycle in the break spectrum analyses. SLO SLMCPR will be evaluated each reload, including the impact on uncertainties associated with LPRMs out of service, TIP channels out of service and the LPRM calibration interval.

Table 2.4 Disposition of Operating Flexibility and EOOS Options on Limiting Events *(Continued)*

Option	Affected Event Name	Comments
ICF/MELLLA	ASME Overpressure FWCF LRNB TTNB IHPCIS LOCA ATWS SLMCPR Slow flow runup	These events need to be performed to cover the range of flows for ICF and MELLLA.

2.2 *Plant Specific Modeling Sensitivities*

As part of the initial application of the AURORA-B AOO methodology to a plant, justification must be provided to ensure that conservative plant parameters are being used. This requirement is defined in Limitation and Conditions 7 and 11 of the Reference 11 safety evaluation. In particular, these limitations and conditions state:

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.*

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.*

In order to comply with these requirements, a set of sensitivity studies was performed. Separate sensitivity studies were performed for each of the three figures of merit that were required to license SUS2-21: ΔMCPR (Table 2.5), transient nodal power (Table 2.6 for RODEX4 based and Table 2.7 for RODEX2 based), and overpressure (Table 2.8). These sensitivity studies address the key parameters required for licensing with the exception of C12 which is described below. In addition to these sensitivity studies, licensing calculations will also look at a wide range of core exposures and flow rates to ensure that the conservative statepoints have been analyzed.

Uncertainties associated with PIRT parameters R01, R02, and SL02 were evaluated for the initial transition. **[**

]

[ ]

Limitation and Condition 16 of the Reference 11 SE, given below, also requires a plant specific justification.

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 Susquehanna and accounts for [ ]. The AURORA-B model [ ]

Table 2.5 Plant Parameter Sensitivity Results for ΔMCPR

Table 2.5 Plant Parameter Sensitivity Results for ΔMCPR *(Continued)*

Table 2.5 Plant Parameter Sensitivity Results for ΔMCPR *(Continued)*

Table 2.5 Plant Parameter Sensitivity Results for ΔMCPR *(Continued)*

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Table 2.6 Plant Parameter Sensitivity Results for Transient Nodal Power

Table 2.6 Plant Parameter Sensitivity Results for Transient Nodal Power *(Continued)*

Table 2.6 Plant Parameter Sensitivity Results for Transient Nodal Power *(Continued)*

Table 2.6 Plant Parameter Sensitivity Results for Transient Nodal Power *(Continued)*

Table 2.7 Plant Parameter Sensitivity Results for Heat Flux Ratio

Table 2.7 Plant Parameter Sensitivity Results for Heat Flux Ratio *(Continued)*

Table 2.7 Plant Parameter Sensitivity Results for Heat Flux Ratio *(Continued)*

Table 2.8 Plant Parameter Sensitivity Results for Overpressure

Table 2.8 Plant Parameter Sensitivity Results for Overpressure *(Continued)*

2.3 *Inadvertent Startup of the HPCI Pump Modeling Considerations*

The approval of the AURORA-B methodology for the inadvertent startup of the HPCI pump (IHPCIS) event included a Limitation and Condition 12 that required that **[**

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3.0 MECHANICAL DESIGN ANALYSIS

The mechanical design analyses for ATRIUM-10 and ATRIUM 11 fuel assemblies are presented in the applicable mechanical design reports:

• Reload SUS2-19 ATRIUM-10 Fuel Assemblies	Reference 12
• Reload SUS2-20 ATRIUM-10 Fuel Assemblies	Reference 13
• Reload SUS2-21 ATRIUM 11 Fuel Assemblies	References 5 and 7

The maximum exposure limits for the reload fuel are:

ATRIUM-10 Maximum Exposure GWd/MTU	
Assembly Average	Full-Length Fuel Rod
54.0	62.0

ATRIUM 11 Maximum Exposure GWd/MTU	
Assembly Average	Full-Length Fuel Rod
57.0	62.0

The ATRIUM-10 and ATRIUM 11 LHGR limits are presented in Section 8.2. The fuel cycle design analysis (Reference 2) has verified the ATRIUM-10 and ATRIUM 11 fuel assemblies remain within licensed burnup limits.

4.0 THERMAL-HYDRAULIC DESIGN ANALYSIS

4.1 *Thermal-Hydraulic Design and Compatibility*

The results of the thermal-hydraulic characterization and compatibility analyses are presented in the thermal-hydraulic design report (Reference 14). The analysis results demonstrate the thermal-hydraulic design and compatibility criteria are satisfied for the Susquehanna Unit 2 transition core consisting of ATRIUM-10 and ATRIUM 11 fuel assemblies.

4.2 *Safety Limit MCPR Analysis*

The safety limit MCPR (SLMCPR) is defined as the minimum value of the critical power ratio which ensures less than 0.1% of the fuel rods in the core are expected to experience boiling transition during normal operation or an anticipated operational occurrence (AOO). The SLMCPR for all fuel in the SUS2-21 core was determined using the methodology described in Reference 15. The analysis was performed with a power distribution that conservatively represents expected reactor operating states that could both exist at the MCPR operating limit and produce a MCPR equal to the SLMCPR during an AOO.

The SUS2-21 SLMCPR analysis used the SPCB critical power correlation described in Reference 16 for the ATRIUM-10 fuel. The ACE/ATRIUM 11 critical power correlation, described in Reference 17, was applied to the ATRIUM 11 fuel assemblies.

In the Framatome methodology, the effects of channel bow on the critical power performance are accounted for in the SLMCPR analysis. Reference 15 discusses the application of a realistic channel bow model.

The fuel- and plant-related uncertainties used in the SLMCPR analysis are presented in Table 4.1. The radial power uncertainty used in the analysis includes the effects of

- Up to 42% traversing in-core probes (TIP) out-of-service (OOS) (100% TIP available at startup unless the affected LPRMs are bypassed).
- Up to 50% local power range monitors (LPRM) OOS (LPRM substitution model on or off).
- An LPRM calibration interval of up to 2,500 effective full-power hours (EFPH).

For TLO, analyses were performed for the minimum and maximum core flow conditions associated with rated power (99% and 108%).

The analysis results support a TLO SLMCPR of 1.06 and a SLO SLMCPR of 1.07. Consistent with the approved Susquehanna Unit 2 Technical Specification SLMCPR values, the Cycle 21 operating limits are based on SLMCPR values of 1.08 for TLO and 1.11 for SLO. Table 4.2 presents a summary of the analysis results including the SLMCPR and the percentage of rods expected to experience boiling transition.

4.3 *Core Hydrodynamic Stability*

4.3.1 Stability Exclusion Region Analysis

The NRC approved STAIF stability methodology (References 1, 18, and 19) was used in the core hydrodynamic stability analyses performed in support of Cycle 21. The power/flow state points used for this analysis were selected to help define 0.8 (regional) or 0.85 (global) and 1.0 decay ratio lines.

For the Cycle 21 analysis, the effects of cycle exposure, radial power distribution, and axial power distribution were considered. The analyses were performed using the design basis step-through control rod patterns for Cycle 21 and supports variations in FWT ranging from −10°F to +5°F. Operation is based on the expected previous cycle energy of 2,646 GWd.

The maximum STAIF decay ratio results shown in Table 4.3 were generated to assist Susquehanna in establishing administrative stability region boundaries, which are consistent with guidelines established by the BWROG and with commitments to the NRC. The results provided in Table 4.3 are based on the Cycle 21 core design from beginning of cycle (BOC) to licensing basis EOFP (36,735 MWd/MTU core average exposure and 2,558 GWd cycle energy). Operation in coastdown beyond this core average exposure is supported up to 37,856 MWd/MTU (maximum cycle energy of 2,711 GWd). These decay ratio results may be used to define constant decay ratio curves of 0.80 (regional), 0.85 (global) and 1.0 for Cycle 21. **[**

] As specified in Reference 3, these decay ratio results assume a flow runback from nominal conditions only.

4.3.2 DIVOM Analysis

The bases for the DIVOM calculation are identified below:

- The DIVOM analyses were performed in accordance with References 1 and 20.
- Neutronics input to RAMONA5-FA for Cycle 21 was generated using MICROBURN-B2 for the exposure points listed in Table 4.4. Any significant changes in operation from the cycle step-through will have to be evaluated for their impact on DIVOM.
- The DIVOM calculation utilized the latest version of RAMONA5-FA which includes bounds checking consistent with the approved SPCB and ACE / ATRIUM 11 topical reports, References 16 and 17.
- The base power/flow state point is the power calculated by MICROBURN-B2 based on an eigenvalue search and 30 Mlbm/hr (natural circulation).
- The Xenon level used was the equilibrium Xenon at 100% of rated power and 99 Mlbm/hr.

Key results of the DIVOM analysis:

- The DIVOM points for all exposures and state points are included in Figure 4.1 – Figure 4.6.
- The DIVOM limiting exposure was found to be 0 MWd/MTU (Figure 4.1). Both global and regional decay ratios decreased below 0.8 when core flow was increased to 35 Mlbm/hr.
- The 6,700 MWd/MTU case was rerun with 5% higher power at natural circulation flow. The DIVOM points are provided in Figure 4.6.
- No channel instabilities were observed for the DIVOM analyses.

Comparison of the Cycle 21 limiting DIVOM curve to that for Unit 2 Cycle 20 (Reference 21) shows the slopes are similar.

The core MCPR values for the 100% power / 99 Mlbm/hr and low power / flow state points are reported in Table 4.5.

Table 4.1 Fuel- and Plant-Related Uncertainties for Safety Limit MCPR Analyses

Parameter	Uncertainty
Plant-Related Uncertainties	
Feedwater flow rate	1.76%
Feedwater temperature	0.76%
Core Pressure	0.50%
Total core flow rate	
Two-loop	2.50%
Single-loop	6.00%
Fuel-Related Uncertainties	
[	
	]

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* **[** **]**

† **[**

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Table 4.2 Results Summary for Safety Limit MCPR Analyses

Power (% Rated)	Flow (% Rated)	Minimum Supported SLMCPR*	Percentage of Rods in Boiling Transition
100	108	TLO - 1.06	0.0686
100	99	TLO - 1.06	0.0633
67.1	52	SLO - 1.07	0.0791

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* The OLMCPR shown in Table 8.1 through Table 8.5 were developed assuming a TLO SLMCPR of 1.08 and a SLO SLMCPR of 1.11.

Table 4.3 Data to Define Constant Decay Ratio Curves

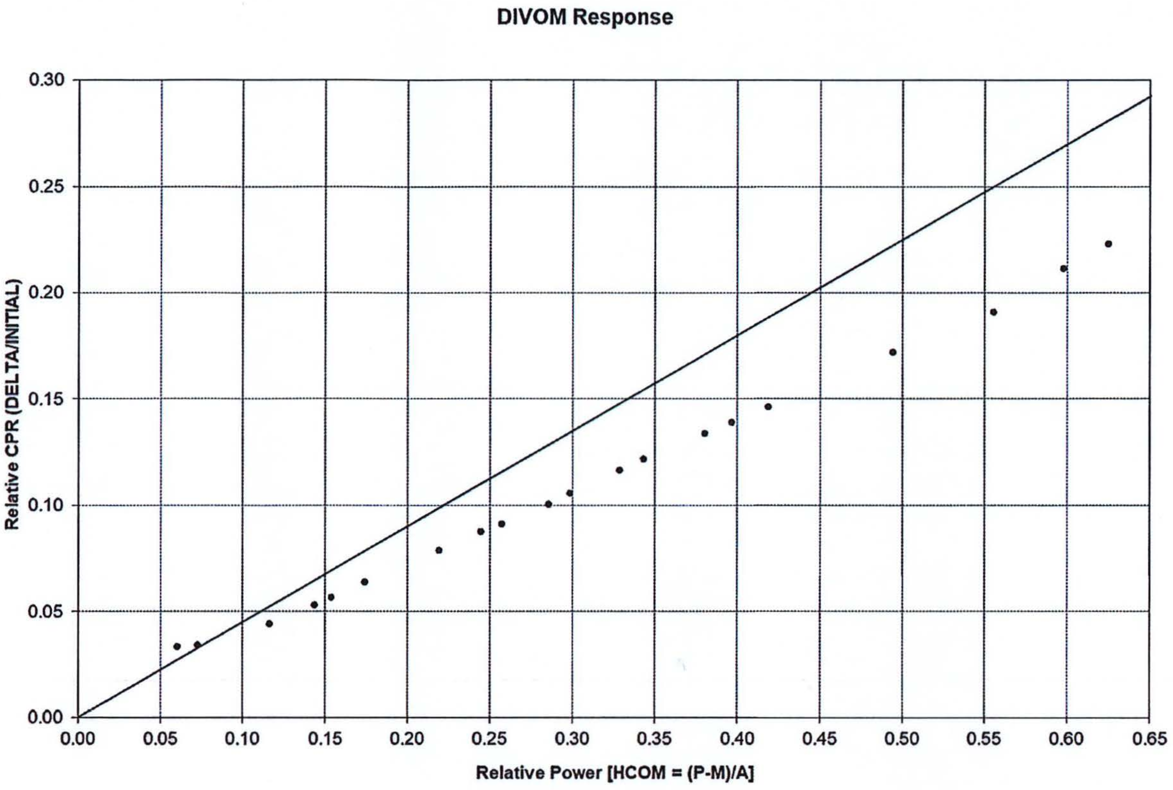
% of Rated Power / Flow	Global Decay Ratio	Regional Decay Ratio
28.0% power / 30 Mlbm/hr	0.480	0.392
35.5% power / 30 Mlbm/hr	0.847	0.607
37.5% power / 30 Mlbm/hr	0.999	0.673
54.6% power / 40 Mlbm/hr	0.861	0.700
56.5% power / 40 Mlbm/hr	0.944	0.764
58.9% power / 45 Mlbm/hr	0.806	0.617
66.2% power / 45 Mlbm/hr	0.983	0.827
67.9% power / 50 Mlbm/hr	0.836	0.677
80.0% power / 50 Mlbm/hr	0.991	0.933
68.5% power / 50.5 Mlbm/hr	0.832	0.678
82.9% power / 50.5 Mlbm/hr	0.997	0.975
86.6% power / 56.0 Mlbm/hr	0.839	0.794

Table 4.4 DIVOM Exposure Points

Exposure MWd/MTU
0
6,700
13,500 (PHE)
16,000
18,340.2

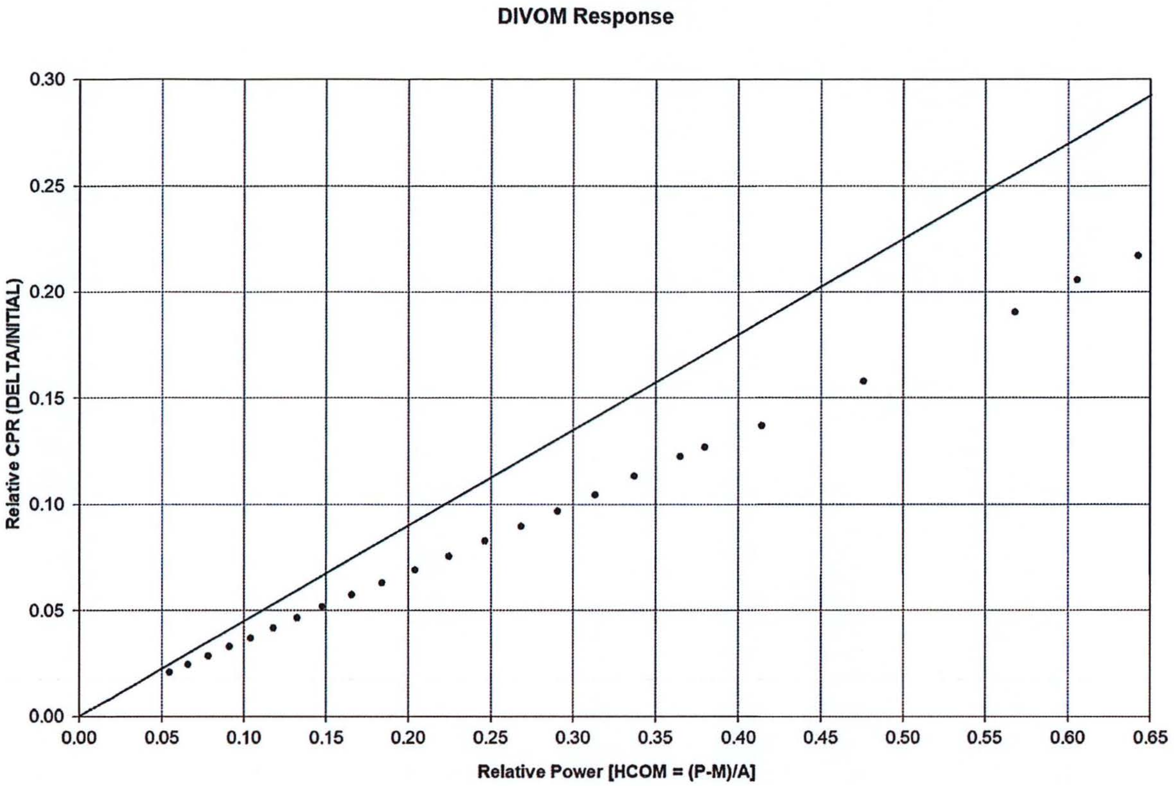
Table 4.5 Core MCPR Results for OPRM Setpoint Analysis for DIVOM Analyses

Cycle Point	100% Power 99 Mlbm/hr	Low Power Low Flow
0 MWd/MTU at 47.7/30	1.615	1.868
6,700 MWd/MTU at 46.0/30	1.647	1.942
13,500 MWd/MTU at 44.9/30	1.523	1.791
16,000 MWd/MTU at 43.7/30	1.564	1.980
18,340 MWd/MTU at 44.6/30	1.513	1.872
0 MWd/MTU at 56.0/35	1.615	1.716
6,700 MWd/MTU at 51.0/30	1.647	1.740



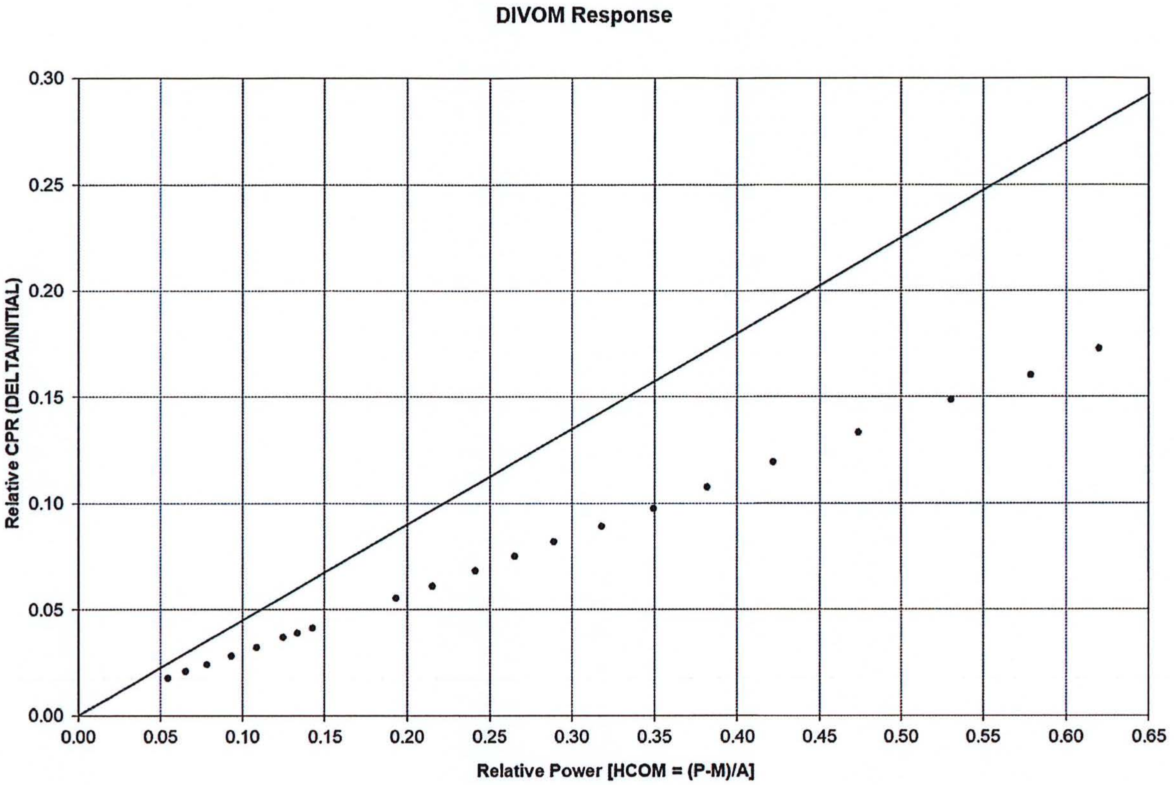
(P-M)/A	ΔCPR/ICPR	(P-M)/A	ΔCPR/ICPR
0.0603	0.0334	0.2985	0.1055
0.0727	0.0339	0.3289	0.1165
0.1165	0.0440	0.3430	0.1215
0.1439	0.0531	0.3803	0.1336
0.1538	0.0564	0.3969	0.1390
0.1743	0.0639	0.4191	0.1460
0.2197	0.0788	0.4944	0.1718
0.2449	0.0874	0.5555	0.1907
0.2571	0.0911	0.5979	0.2111
0.2855	0.1005	0.6249	0.2231

Figure 4.1 0 MWd/MTU
DIVOM Response



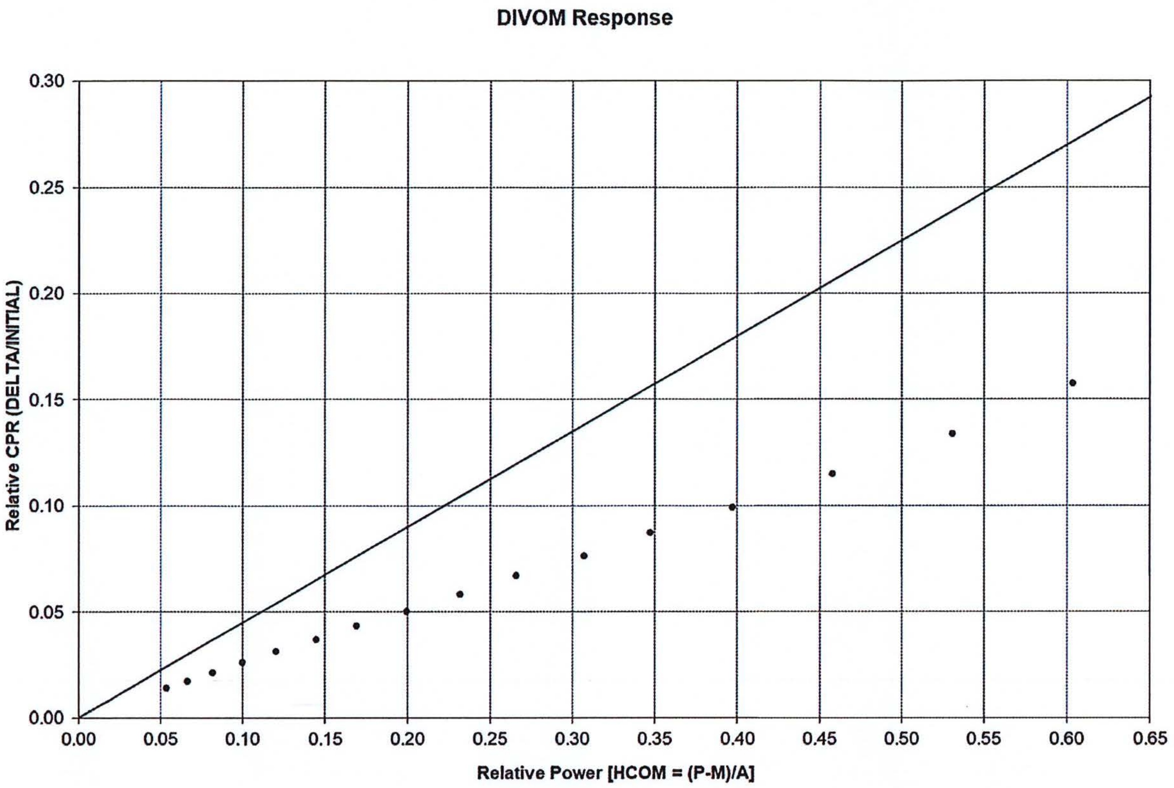
(P-M)/A	ΔCPR/ICPR	(P-M)/A	ΔCPR/ICPR
0.0546	0.0207	0.2467	0.0825
0.0660	0.0243	0.2684	0.0894
0.0782	0.0283	0.2907	0.0967
0.0910	0.0327	0.3133	0.1044
0.1041	0.0370	0.3372	0.1132
0.1178	0.0418	0.3648	0.1226
0.1325	0.0466	0.3800	0.1267
0.1479	0.0517	0.4141	0.1369
0.1657	0.0572	0.4760	0.1579
0.1845	0.0628	0.5680	0.1902
0.2041	0.0690	0.6055	0.2056
0.2250	0.0755	0.6427	0.2167

Figure 4.2 6,700 MWd/MTU DIVOM Response



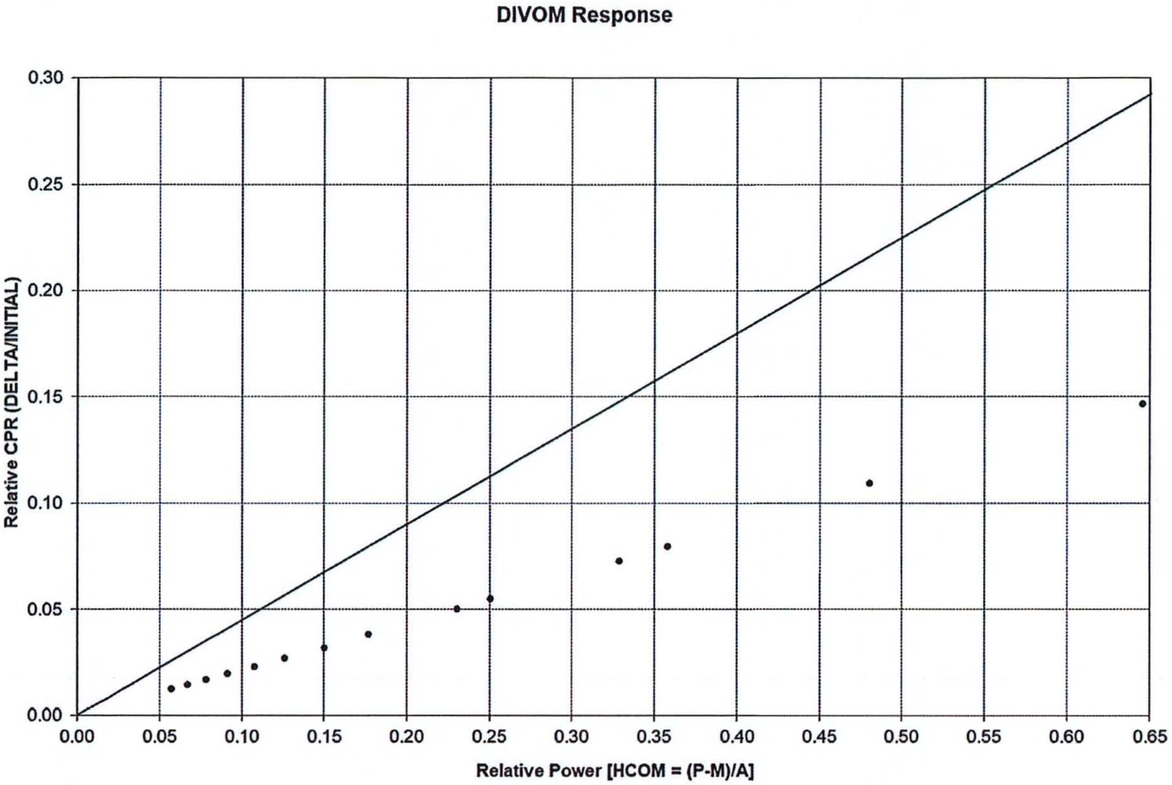
(P-M)/A	ΔCPR/ICPR	(P-M)/A	ΔCPR/ICPR
0.0546	0.0174	0.2657	0.0752
0.0656	0.0207	0.2890	0.0819
0.0785	0.0240	0.3178	0.0890
0.0930	0.0279	0.3492	0.0975
0.1085	0.0321	0.3817	0.1076
0.1248	0.0367	0.4219	0.1191
0.1333	0.0389	0.4738	0.1334
0.1427	0.0413	0.5303	0.1487
0.1934	0.0553	0.5788	0.1603
0.2156	0.0610	0.6202	0.1727
0.2412	0.0681		

**Figure 4.3 13,500 MWd/MTU
DIVOM Response**



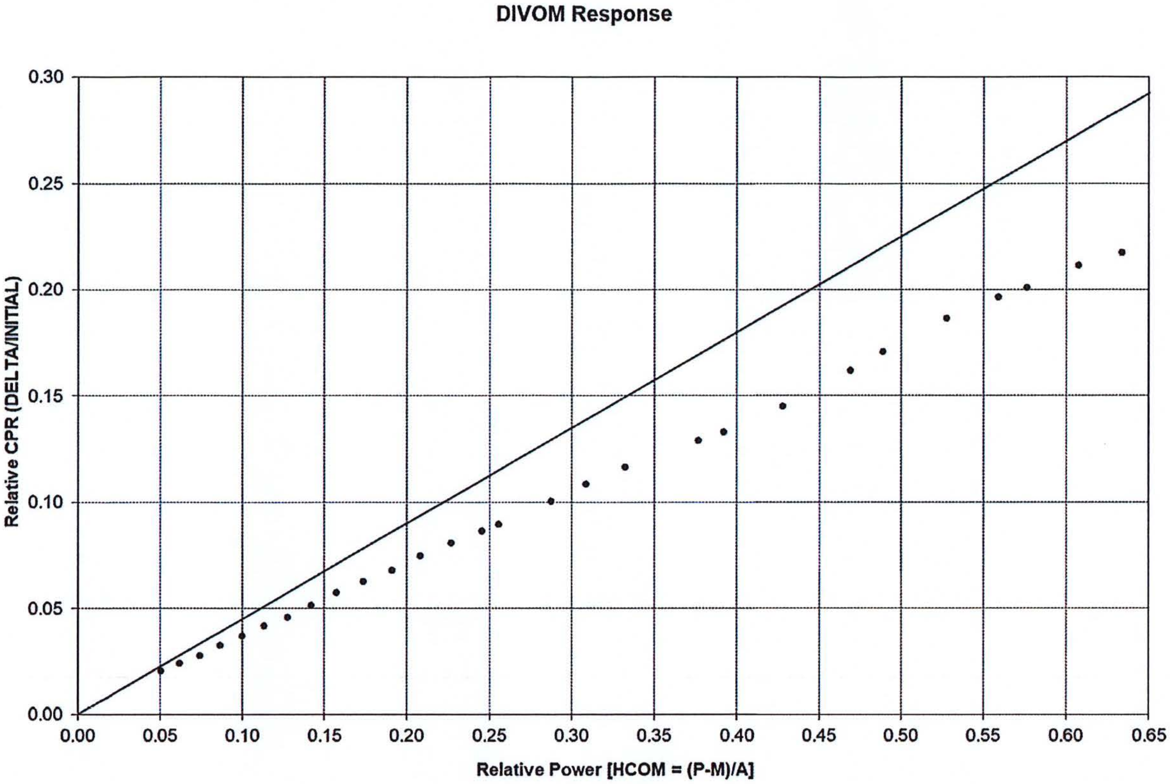
(P-M)/A	ΔCPR/ICPR	(P-M)/A	ΔCPR/ICPR
0.0535	0.0138	0.2319	0.0583
0.0663	0.0173	0.2660	0.0670
0.0818	0.0211	0.3071	0.0763
0.0999	0.0259	0.3476	0.0873
0.1201	0.0311	0.3973	0.0992
0.1445	0.0368	0.4577	0.1147
0.1693	0.0434	0.5310	0.1338
0.1999	0.0501	0.6035	0.1573

**Figure 4.4 16,000 MWd/MTU
DIVOM Response**



(P-M)/A	ΔCPR/ICPR	(P-M)/A	ΔCPR/ICPR
0.0572	0.0124	0.1773	0.0379
0.0668	0.0144	0.2306	0.0502
0.0781	0.0168	0.2504	0.0549
0.0914	0.0196	0.3286	0.0727
0.1078	0.0227	0.3579	0.0795
0.1264	0.0269	0.4808	0.1091
0.1501	0.0318	0.6455	0.1466

**Figure 4.5 18,340.2 MWd/MTU
DIVOM Response**



(P-M)/A	ΔCPR/ICPR	(P-M)/A	ΔCPR/ICPR	(P-M)/A	ΔCPR/ICPR
0.0506	0.0204	0.1913	0.0679	0.3921	0.1331
0.0620	0.0239	0.2084	0.0745	0.4281	0.1450
0.0741	0.0275	0.2268	0.0808	0.4692	0.1617
0.0865	0.0324	0.2454	0.0865	0.4891	0.1707
0.0999	0.0370	0.2559	0.0895	0.5277	0.1862
0.1132	0.0417	0.2876	0.1003	0.5593	0.1966
0.1276	0.0459	0.3088	0.1083	0.5765	0.2010
0.1422	0.0515	0.3324	0.1166	0.6075	0.2111
0.1578	0.0572	0.3768	0.1289	0.6340	0.2171
0.1741	0.0625				

**Figure 4.6 6,700 MWd/MTU at 51.0P/30F
DIVOM Response**

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

The AURORA-B methodology (Reference 11) is used with the Framatome THERMEX methodology (Reference 24) for the generation of thermal limits. 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.

The AURORA-B AOO methodology (Reference 11) 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.

The SPCB critical power correlation (Reference 16) is used to evaluate the thermal margin for the ATRIUM-10 fuel. The ACE/ATRIUM 11 critical power correlation (Reference 17) is used in the thermal margin evaluations for the ATRIUM 11 fuel.

5.1 *System Transients*

The reactor plant parameters for the system transient analyses were provided by the utility. Analyses have been performed to determine power-dependent MCPR limits that protect operation throughout the power/flow domain shown in Figure 1.1.

At Susquehanna, direct scram on turbine stop valve (TSV) position and turbine control valve (TCV) fast closure are bypassed at power levels less than 26% of rated (P_{bypass}). Scram will occur when the high pressure or high neutron flux scram setpoint is reached. Reference 25 indicates MCPR and LHGR limits only need to be monitored at power levels greater than or equal to 23% of rated, which is the lowest power analyzed for this report.

The limiting exposure for rated power pressurization transients is typically at end of full power (EOFP) when the control rods are fully withdrawn. Analyses were performed at cycle exposures prior to EOFP to ensure that the operating limits provide the necessary protection. The licensing basis EOFP (LEOFP) analysis was performed at EOFP + 450 MWd/MTU (core average exposure of 36,735 MWd/MTU). Analyses were performed to support coastdown operation to a core average exposure of 37,856 MWd/MTU. The licensing basis exposures used to develop the limits are presented in Table 5.1.

All pressurization transients assumed two of the lowest setpoint safety relief valves (SRV) were inoperable. This basis supports operation with 2 SRV out-of-service. Furthermore, the SRV were conservatively modeled in safety mode.

The Susquehanna Unit 2 turbine bypass system includes five bypass valves. However, for base case analyses in which credit is taken for turbine bypass operation, only four of the turbine bypass valves are assumed operable.

Variations in feedwater temperature of -10 to +5°F from the nominal feedwater temperature and variation of -15 to +10 psi in dome pressure are considered base case operation, not an EOOS condition. Sensitivities were performed to determine the limiting conditions (Section 2.2) and the transient analyses were performed with the limiting feedwater and dome pressure in the allowable ranges.

The results of the system pressurization transients are sensitive to the scram speed used in the calculations. To take advantage of average scram speeds faster than those associated with the Technical Specifications requirements, scram speed-dependent $MCPR_p$ limits are provided. The realistic average scram insertion times and the maximum allowable average scram insertion times used in the analyses are presented in Table 5.2. The realistic average scram $MCPR_p$ limits can only be applied if the scram speed test results meet their respective insertion times. System transient analyses were performed to establish $MCPR_p$ limits for realistic average scram and maximum allowable average scram insertion times. The Susquehanna Unit 2 Technical Specifications (Reference 25) allow for operation with up to 13 “slow” and 1 stuck control rod. One additional control rod is assumed to fail to scram. The realistic average scram and maximum allowable average scram analyses were performed to conservatively account for the effect of the slow and stuck rods on scram reactivity. For transient events below P_{bypass} without direct scram, the results are relatively insensitive to scram speed, and only maximum allowable average scram analyses are performed.

5.1.1 Load Rejection Without Bypass / Turbine Trip Without Bypass

The generator load rejection without bypass (LRNB) and the turbine trip without bypass (TTNB) events were conservatively combined as one event. The combined LRNB/TTNB event causes closure of the turbine stop valves (TSV) and fast closure of the turbine control valves (TCV). The closure of the TSV and TCV creates a compression wave that travels 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 and TCV causes a reactor scram and a recirculation pump trip (RPT) 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.

The LRNB/TTNB analyses were performed for a range of power and flow conditions to support generation of the thermal limits. Table 5.3 and Table 5.4 present the limiting transient event and results as a function of power used to generate the base case operating limits for realistic and maximum allowable average scram insertion times, respectively. Figure 5.1 through Figure 5.3 present the responses of various reactor and plant parameters for the limiting LRNB/TTNB transient initiated at maximum core power and maximum allowable average scram insertion time.

5.1.2 Feedwater Controller Failure (FWCF)

The increase in feedwater flow due to a failure of the feedwater controller 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 (L8). The initial water level is conservatively assumed to be at the low level alarm (L4) to delay the high-level trip and maximize the core inlet subcooling that result from the FWCF. Reaching the high water level setpoint will trip the main turbine and the reactor feed pump turbines. The main turbine trip causes the TSV to close in order to prevent damage to the turbine from excessive liquid inventory in the steam line. The valve closures create a compression wave that travels to the core causing a void collapse and subsequent rapid power excursion. The closure of the TSV initiates a reactor scram and an RPT. In addition to the TSV closure, the TCV also close in the fast closure mode. Four of the five installed turbine bypass valves (TBV) are assumed operable and provide pressure relief. The core power excursion is mitigated in part by the pressure relief, but the primary mechanisms for termination of the event are reactor scram and revoiding of the core.

The maximum feedwater flow demand assumes two feedwater pump operation (16.5 Mlbm/hr, Reference 4) below 50% rated power and three feedwater pump operation for 50% rated power and above. Analyses performed for SUS2-21 have demonstrated ΔMCPRs and LHGRFAC$_p$ multipliers calculated for the FWCF transients initiated at 50% power or greater to establish operating limits for two-loop operation can be conservatively used to establish operating limits for single-loop operation. In order to bound the effects of single-loop operation, the feedwater maximum flow runout for the two-loop operation cases is increased from 16.5 Mlbm/hr to 19 Mlbm/hr for core flows ≤ 52 Mlbm/hr (the max SLO core flow) and powers < 50%. Above 50% power, the FWCF event is bound by SLO pump seizure by a significant margin so no adjustment is required.

The FWCF analyses were performed for a range of power and flow conditions to support generation of the thermal limits. Table 5.3 and Table 5.4 present the limiting transient event and results as a function of power used to generate the base case operating limits for realistic and maximum allowable average scram insertion times, respectively. Figure 5.4 through Figure 5.6 present the responses of various reactor and plant parameters for the limiting FWCF transient initiated at maximum core power and maximum allowable average scram insertion time.

5.1.3 Inadvertent Startup of the HPCI Pump

The inadvertent startup of the HPCI system results in the injection of cold water to the reactor vessel 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 the 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 case, the event is similar to an LFWH event. At low power, the HPCI flow can become more than the steam flow, and the water level can increase until the high water level setpoint (L8) is reached. In this case, the event is similar to an FWCF.

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

]

The IHPCIS analyses were performed for a range of power and flow conditions to support generation of the thermal limits. Table 5.3 and Table 5.4 present the limiting transient event and results as a function of power used to generate the base case operating limits for realistic and maximum allowable average scram insertion times, respectively. Figure 5.7 through Figure 5.9 present the responses of various reactor and plant parameters for the limiting IHPCIS transient initiated at maximum core power.

5.1.4 Equipment Out-of-Service Scenarios

The equipment out-of-service (EOOS) scenarios supported for Susquehanna Unit 2 Cycle 21 are presented in Table 1.1.

5.1.4.1 TBVOOS

Operation with TBVOOS means that the fast opening capability of two or more of the turbine bypass valves cannot be assured, thereby reducing the pressure relief capacity during fast pressurization transients. While the base case LRNB/TTNB event is analyzed assuming the turbine bypass valves out-of-service, operation with TBVOOS has an adverse effect on the FWCF event. Analyses of the FWCF event with TBVOOS were performed to establish the TBVOOS operating limits. Table 5.5 and Table 5.6 present the limiting transient event and results as a function of power used to generate the TBVOOS operating limits for maximum allowable and realistic average scram insertion times, respectively.

5.1.4.2 RPTOOS

Operation with RPTOOS means the EOC-RPT trip on TSV position or TCV fast closure cannot be assured. The function of the EOC-RPT feature is to reduce the severity of the core power excursion caused by the pressurization transient. The RPT accomplishes this by helping revoid the core, thereby reducing the magnitude of the reactivity insertion resulting from the pressurization transient. RPT on high dome pressure is allowed. Operation with RPTOOS has an adverse effect on the LRNB/TTNB and FWCF events. Analyses of the LRNB/TTNB and FWCF events with RPTOOS were performed to establish the RPTOOS operating limits. Table 5.7 and Table 5.8 present the limiting transient event and results as a function of power used to generate the RPTOOS operating limits for maximum allowable and realistic average scram insertion times, respectively.

5.1.4.3 One PROOS

Two pressure regulators are provided to maintain primary system pressure control. They independently sense pressure and compare it to separate setpoints. The backup pressure regulator has a higher setpoint so the TCV are controlled by the primary regulator. Downscale failure of a pressure regulator (TCVs closing) is a non-limiting event as long as the backup pressure regulator is in service because the backup system will control the pressure. Therefore, the pressure regulator failure downscale (PRFDS) event with the backup pressure regulators operable is not analyzed.

If the PRFDS event occurs when the backup pressure regulator is inoperable, the TCVs will close in the servo or normal operating mode. Since the TCV closure is not a fast closure, there is no direct scram initiated by the closure. Closure of the TCV will result in an increase in reactor pressure and an increase in core power. Scram will be initiated either by high neutron flux (at high core power) or by high dome pressure (at lower core power). Similarly, RPT will only occur when/if initiated by high dome pressure. Table 5.9 and Table 5.10 present the limiting transient event and results as a function of power used to generate the PROOS operating limits for realistic and maximum allowable average scram insertion times, respectively.

5.1.4.4 One TCV/TSV Closed

With one of the turbine control valves or turbine stop valves assumed closed, the available TCVs will be further opened when operating at a given power level. In addition, the highest attainable power is decreased because of the decreased total steam flow capacity of the TCVs/TSVs. At the same initial power with the valves further open, TCV/TSV closure events such as the LRNB/TTNB are less severe because the pressurization occurs over a longer period of time. While the FWCF event is not affected during the turbine stop valve closure portion of the event it may be impacted during the overcooling phase. At a certain power level, the TCVs will be in the full open position with no ability to accommodate any additional steam flow increase during the overcooling phase. For the base case FWCF event, 4 of the 5 main turbine bypass valves are available and will open to accommodate an increase in steam flow that exceeds the capacity of the TCVs. However, if bypass valves open during the overcooling phase, the FWCF event will be more severe because when the TSV close on high water level, the remaining steam relieving capacity of the bypass valves is reduced. Analyses show at power levels ≤ 70% of rated, the steam flow will not exceed the capacity of the TCVs during the overcooling phase of the FWCF.

The base case results provided in Table 5.3 and Table 5.4 for power levels ≤ 70% of rated are applicable to operation with one TCV/TSV closed. Additional analyses were performed at 75%

of rated power to demonstrate the FWCF results with one TCV/TSV closed are bound by the base case limits provided in Table 8.1 and Table 8.2.

5.1.4.5 Single-Loop Operation

Single-loop operation is only supported up to a maximum core flow of 52 Mlbm/hr which corresponds to a maximum power level of 2,652 MWt at the MELLLA boundary. In SLO, the two-loop operation limiting ΔMCPRs and LHGRFAC multipliers for LRNB/TTNB, FWCF (as discussed in Section 5.1.2), IHPCIS, and PRFDS remain applicable. The power-dependent LHGR multipliers established for TLO are applicable during SLO. The power-dependent MCPR operating limits for SLO are established by adding the limiting power-dependent ΔMCPR for TLO to the SLMCPR for SLO (see Section 4.2).

While the recirculation pump seizure event is classified as an accident, the event is evaluated to protect the AOO acceptance criteria. Pump seizure is a postulated accident where an operating recirculation pump suddenly stops rotating. This results in a rapid decrease in core flow, a decrease in the fuel rod heat transfer rate and a decrease in the critical power ratio. The rapid decrease in the core flow causes an increase in core voiding, a decrease in core power, and an increase in the water level. If the water level reaches the high water level (L8) trip setpoint, the TSV and TCV will close in order to protect the turbine from excessive moisture. Vessel pressure will increase following closure of the valves and the TBV and/or SRVs may open to control pressure. Scram is initiated by closure of the valves. Seizure of one recirculation pump during two-loop operation and seizure of the active recirculation pump during single-loop operation are postulated. The pump seizure accident is non-limiting for ATRIUM 11 fuel during TLO (see Table 5.11) but more severe during SLO. Table 5.12 presents the ATRIUM 11 results of the SLO pump seizure analysis. For ATRIUM-10, the pump seizure event was analyzed by another vendor. The results are applicable for realistic and maximum allowable average scram insertion times as well as the remaining EOOS presented in Table 1.1. Figure 5.10 through Figure 5.12 present the responses of various reactor and plant parameters for the limiting SLO pump seizure transient. The event is a power decrease event so there is no threat to the transient thermal-mechanical limits during this event. The results for the state point analyzed are limiting for lower power levels and core flows.

LOCA is more severe when initiated during SLO. Therefore, a reduced MAPLHGR limit is established for SLO (see Section 6.1).

5.1.4.6 Two SRVOOS

As noted earlier, all pressurization transient analyses were performed with two of the lowest setpoint SRVs assumed inoperable. Therefore, the base case operating limits support operation with two SRVOOS. The EOOS operating limits also support operation with two SRVOOS.

5.2 *Loss of Feedwater Heating*

A loss of feedwater heating (LFWH) transient can occur when a steam extraction line to a feedwater heater is closed. This produces a gradual drop in the temperature of the feedwater entering the reactor vessel. This results in an increase in core inlet subcooling, which reduces voids, thereby increasing core power and shifting the axial power distribution toward the bottom of the core. As a result of the axial power shift and increase core power, voids begin to build up at the bottom region of the core, acting as negative feedback to the increased subcooling effect. This 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, so no pressurization occurs.

In support of the first transition to ATRIUM 11 in Unit 2 Cycle 21, a proposed bounding LFWH evaluation was performed for the Susquehanna units using Framatome's approved generic methodology (Reference 26) assuming a **[ ]** decrease in feedwater temperature. **[**

] The LFWH event is a non-limiting event with respect to ΔCPR. The LFWH results are presented in Table 5.13.

The LFWH results for power levels ≤ 75% of rated are also applicable to the one TCV/TSV closed condition.

5.3 *Control Rod Withdrawal Error*

The control rod withdrawal error (CRWE) transient is an inadvertent reactor operator initiated withdrawal of a control rod. This withdrawal increases local power and core thermal power, lowering the core MCPR. The CRWE transient is typically terminated by control rod blocks initiated by the rod block monitor (RBM). The CRWE analysis has been performed at rated and off-rated conditions assuming an ARTS rod block monitoring system. This analysis assumes that one RBM channel and not more than one-half of the LPRM detectors in the second RBM channel are bypassed or inoperable. The CRWE analysis assumes xenon free conditions and assumes that the plant will be operating in a standard A or B sequence control rod pattern.

The RBM equipment for ARTS includes a filter to reduce signal noise levels. Generic RBM setpoint reduction values for filter time lag effects have been determined based on representative CRWE calculations using a control rod withdrawal rate of 3.0 inches/sec ± 20% (3.6 inches/sec maximum) and a signal delay time constant of 0.50 ± 0.05 seconds (0.55 seconds maximum). The following are the recommended ARTS RBM setpoint reduction factors for use with filtered RBM signals.

- For 106% < RBM setpoint ≤ 108%, subtract 0.6%.
- For 108% < RBM setpoint ≤ 116%, subtract 0.8%.
- For 116% < RBM setpoint ≤ 124%, subtract 1.0%.
- For 124% < RBM setpoint ≤ 127%, subtract 1.2%.

The analysis supports plant operation with or without operational turbine bypass valves.

5.3.1 Base Case Operation

Table 5.14 and Table 5.16 list the limiting MCPRs for the CRWE analyses performed with turbine bypass valves operable for the powers analyzed. The MCPR values are based on a SLMCPR of 1.08. For other values of SLMCPR, the CRWE MCPR values can be adjusted by the difference in the SLMCPR and 1.08.

The realistic average scram insertion time $MCPR_p$ limits are provided in Table 8.1 and Table 8.3. In the high power range, the $MCPR_p$ limits in Table 8.1 support an RBM trip setpoint of 117% or less. In the intermediate power range, the $MCPR_p$ limits in Table 8.1 and Table 8.3 support an RBM trip setpoint of 122% or less. In the lower power range, the $MCPR_p$ limits in Table 8.1 and Table 8.3 support an RBM trip setpoint of 127% or less.

The base case CRWE will not exceed 135% of the steady-state LHGR limit; therefore the PAPT LHGR will not be violated.

Based on the evaluation of previous cycles of operation (SUS1-19 and SUS2-18) and the results for SUS2-21, the conclusion that the base case CRWE results presented in Table 5.14 and Table 5.16 for power levels ≤ 75% power are also applicable to the one TCV/TSV closed condition can be generically applied for all future cycles of operation for the Susquehanna units.

5.3.2 Turbine Bypass Valves Inoperable

Table 5.15 and Table 5.17 list the limiting MCPRs for the CRWE analyses performed with turbine bypass valves inoperable for the powers analyzed. The MCPR values are based on a SLMCPR of 1.08. For other values of SLMCPR, the CRWE MCPR values can be adjusted by the difference in the SLMCPR and 1.08.

For operating powers of 95% and below, the CRWE results are not impacted by the turbine bypass valves. For power above 95%, a conservative ΔCPR adder of **[**

] was applied to the CRWE MCPR values (with bypass) to account for turbine bypass valves out-of-service.

The CRWE $LHGRFAC_p$ values for turbine bypass valves inoperable shown in Table 5.18 are independent of the high power range RBM setpoint.

5.3.3 Rod Block Monitor System Operational Requirements

The bounding power dependent CRWE MCPR values where the ARTS RBM trip is not required to be in service have been selected from representative unblocked MCPR results. The RBM system operational requirements with turbine bypass valves in and out of service are shown in Table 5.19.

5.4 *Slow Flow Runup Analysis*

Flow-dependent MCPR ($MCPR_f$) limits and LHGR multipliers ($LHGRFAC_f$) are established to support operation at off-rated core flow conditions. The limits are based on the CPR and heat flux changes experienced by the fuel during slow flow excursions. The slow flow excursion event assumes a failure of the recirculation flow control system such that the core flow increases slowly to the maximum flow physically permitted by the equipment (121 Mlbm/hr). An uncontrolled increase in flow creates the potential for a significant increase in core power and heat flux. Operation with one TCV or TSV closed causes a larger increase in pressure and power during the flow excursion which results in a steeper flow runup path. A conservatively steep flow runup path was used in the analysis. The slow flow runup analyses were performed to support operation in all the EOOS scenarios.

XCOBRA is used to calculate the change in critical power ratio during a two-loop flow runup to the maximum flow rate. The MCPR_f limit is set such that the increase in core power, resulting from the maximum increase in core flow, assures the TLO safety limit MCPR is not violated. Calculations were performed for a range of initial flow rates to determine the corresponding MCPR values that put the limiting assembly on the safety limit MCPR at the high flow condition at the end of the flow excursion.

MCPR_f limits that provide the required protection are presented in Table 8.5. The MCPR_f limits are applicable for all exposures.

Flow runup analyses were performed with CASMO-4/MICROBURN-B2 to determine LHGRFAC_f multipliers for ATRIUM-10 and ATRIUM 11 fuel. The analysis assumes that the recirculation flow increases slowly along the limiting rod line to the maximum flow physically permitted by the equipment. A series of flow excursion analyses were performed at several exposures throughout the cycle starting from different initial power/flow conditions. Xenon is assumed to remain constant during the event. The LHGRFAC_f multipliers are established to provide protection against fuel centerline melt and overstraining of the cladding during a flow runup. LHGRFAC_f multipliers are presented in Table 8.10.

The maximum flow during a flow excursion in single-loop operation is much less than the maximum flow during two-loop operation. Therefore, the MCPR_f limits and LHGRFAC_f multipliers for two-loop operation are applicable for SLO.

5.5 *Licensing Axial Power Shape*

A hard bottom burn cycle step-through is used by Framatome for plant transient analyses to bound actual operation. A 450 MWd/MTU exposure window was added to the design EOFP exposure to account for possible variations in hot target eigenvalues and provide additional margin for licensing compliance. This conservative licensing axial power profile is given in Table 5.20. Cycle 21 operation is considered to be in compliance when:

- The integrated normalized power generated in the bottom 7 nodes from the projected EOFP solution at the state conditions provided in Table 5.20 is greater than the integrated normalized power generated in the bottom 7 nodes in the licensing basis axial power profile, and the individual normalized power from the projected EOFP solution is greater than the corresponding normalized power from the licensing basis axial power profile for at least 6 of the 7 bottom nodes (i.e., the normalized power in any one of the 7 bottom nodes from the projected EOFP solution may be less than the corresponding node from the licensing basis axial power profile).*
- The projected EOFP condition occurs at a core average exposure less than or equal to the licensed EOFP provided in Table 5.20.

The MCPR limits established in this report support full-power operation up to a core average exposure of 36,735 MWd/MTU. Operation in coastdown beyond this core average exposure is supported up to 37,856 MWd/MTU (cycle energy of 2,711 GWd).

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* This comparison should be made periodically during the cycle to ensure compliance. The projection to EOFP should start from the latest actual operating conditions (POWERPLEX or core follow) and the state conditions listed in Table 5.20 used for the comparison.

Table 5.1 Exposure Basis for Susquehanna Unit 2 Cycle 21 Transient Analyses

Cycle Exposure at End of Interval (MWd/MTU)	Core Average Exposure (MWd/MTU)	Comments
0	17,942	Beginning of cycle
18,790	36,735	Design basis rod patterns to EOFP + 450 MWd/MTU – LEOFP
19,912	37,856*	Maximum licensing core exposure – including coastdown

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* Maximum licensing core exposure is defined as the exposure corresponding to the cycle generation of 2,711 GWd.

Table 5.2 Scram Speed Insertion Times

Control Rod Postion (notch)	Maximum Allowable (sec)	Realistic (sec)
48 *(full-out)*	0.000	0.000
48	0.245	0.237
45	0.565	0.507
39	0.883	0.641
25	1.941	1.526
5	3.472	2.725
0 *(full-in)*	3.862	3.025

Table 5.3 Base Case Limiting Transient Event
Realistic Average Scram Insertion Time

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.32	LRNB/TTNB	0.32	LRNB/TTNB
90	0.31	FWCF	0.37	LRNB/TTNB
80	0.40	IHPCIS	0.43	FWCF
70	0.52	IHPCIS	0.51	IHPCIS
60	0.56	FWCF	0.64	FWCF
50	0.71	FWCF	0.74	FWCF
40	0.76	FWCF	0.69	FWCF
26	1.12	FWCF	0.90	FWCF
26 at > 50%F below P_{bypass}	1.64	FWCF	1.44	FWCF
23 at > 50%F below P_{bypass}	1.61	FWCF	1.52	FWCF
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.87	FWCF	1.44	FWCF

Table 5.4 Base Case Limiting Transient Event Maximum Allowable Average Scram Insertion Time

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.43	LRNB/TTNB	0.44	LRNB/TTNB
90	0.41	FWCF	0.48	FWCF
80	0.48	FWCF	0.59	FWCF
70	0.56	FWCF	0.62	FWCF
60	0.74	FWCF	0.65	FWCF
50	0.97	FWCF	0.81	FWCF
40	1.00	FWCF	0.76	FWCF
26	1.34	FWCF	1.00	FWCF
26 at > 50%F below P_{bypass}	1.64	FWCF	1.44	FWCF
23 at > 50%F below P_{bypass}	1.61	FWCF	1.52	FWCF
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.87	FWCF	1.44	FWCF

Table 5.5 TBVOOS Limiting Transient Event
Realistic Average Scram Insertion Time

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.41	FWCF TBVOOS	0.40	FWCF TBVOOS
90	0.38	FWCF TBVOOS	0.40	FWCF TBVOOS
80	0.40	IHPCIS	0.45	FWCF TBVOOS
70	0.52	IHPCIS	0.51	IHPCIS
60	0.61	FWCF TBVOOS	0.69	FWCF TBVOOS
50	0.72	FWCF TBVOOS	0.79	FWCF TBVOOS
40	0.79	FWCF TBVOOS	0.76	FWCF TBVOOS
26	1.15	FWCF TBVOOS	0.96	FWCF TBVOOS
26 at > 50%F below P_{bypass}	2.07	FWCF TBVOOS	1.94	FWCF TBVOOS
23 at > 50%F below P_{bypass}	2.22	FWCF TBVOOS	2.18	FWCF TBVOOS
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.93	FWCF TBVOOS	1.54	FWCF TBVOOS

Table 5.6 TBVOOS Limiting Transient Event
Maximum Allowable Average Scram Insertion Time

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.52	FWCF TBVOOS	0.54	FWCF TBVOOS
90	0.48	FWCF TBVOOS	0.53	FWCF TBVOOS
80	0.51	FWCF TBVOOS	0.59	FWCF TBVOOS
70	0.60	FWCF TBVOOS	0.66	FWCF TBVOOS
60	0.78	FWCF TBVOOS	0.70	FWCF TBVOOS
50	1.00	FWCF TBVOOS	0.83	FWCF TBVOOS
40	1.04	FWCF TBVOOS	0.82	FWCF TBVOOS
26	1.34	FWCF	1.04	FWCF TBVOOS
26 at > 50%F below P_{bypass}	2.07	FWCF TBVOOS	1.94	FWCF TBVOOS
23 at > 50%F below P_{bypass}	2.22	FWCF TBVOOS	2.18	FWCF TBVOOS
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.93	FWCF TBVOOS	1.54	FWCF TBVOOS

Table 5.7 RPTOOS Limiting Transient Event Realistic Average Scram Insertion Time

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.41	LRNB/TTNB RPTOOS	0.41	LRNB/TTNB RPTOOS
90	0.41	LRNB/TTNB RPTOOS	0.42	LRNB/TTNB RPTOOS
80	0.40	IHPCIS	0.44	FWCF RPTOOS
70	0.52	IHPCIS	0.51	IHPCIS
60	0.58	FWCF RPTOOS	0.67	FWCF RPTOOS
50	0.71	FWCF	0.77	FWCF RPTOOS
40	0.76	FWCF	0.72	FWCF RPTOOS
26	1.12	FWCF	0.92	FWCF RPTOOS
26 at > 50%F below P_{bypass}	1.64	FWCF	1.44	FWCF
23 at > 50%F below P_{bypass}	1.61	FWCF	1.52	FWCF
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.87	FWCF	1.44	FWCF

**Table 5.8 RPTOOS Limiting Transient Event
Maximum Allowable Average Scram Insertion Time**

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.55	LRNB/TTNB RPTOOS	0.64	LRNB/TTNB RPTOOS
90	0.57	LRNB/TTNB RPTOOS	0.73	LRNB/TTNB RPTOOS
80	0.54	LRNB/TTNB RPTOOS	0.70	LRNB/TTNB RPTOOS
70	0.57	FWCF RPTOOS	0.79	FWCF RPTOOS
60	0.80	FWCF RPTOOS	0.70	FWCF RPTOOS
50	0.97	FWCF RPTOOS	0.82	FWCF RPTOOS
40	1.03	FWCF RPTOOS	0.79	FWCF RPTOOS
26	1.34	FWCF	1.03	FWCF RPTOOS
26 at > 50%F below P_{bypass}	1.64	FWCF	1.44	FWCF
23 at > 50%F below P_{bypass}	1.61	FWCF	1.52	FWCF
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.87	FWCF	1.44	FWCF

Table 5.9 PROOS Limiting Transient Event
Realistic Average Scram Insertion Time

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.32	LRNB/TTNB	0.32	LRNB/TTNB
90	0.37	PRFDS	0.37	LRNB/TTNB
80	0.41	PRFDS	0.43	FWCF
70	0.52	IHPCIS	0.51	IHPCIS
60	0.77	PRFDS	0.68	PRFDS
50	0.86	PRFDS	0.82	PRFDS
40	1.04	PRFDS	0.87	PRFDS
26	1.32	PRFDS	1.13	PRFDS
26 at > 50%F below P_{bypass}	1.64	FWCF	1.44	FWCF
23 at > 50%F below P_{bypass}	1.61	FWCF	1.52	FWCF
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.87	FWCF	1.44	FWCF

Table 5.10 PROOS Limiting Transient Event Maximum Allowable Average Scram Insertion Time

Core Power (% of rated)	ATRIUM-10 ΔMCPR	Limiting Event	ATRIUM 11 ΔMCPR	Limiting Event
100	0.43	LRNB/TTNB	0.44	LRNB/TTNB
90	0.50	PRFDS	0.48	FWCF
80	0.54	PRFDS	0.59	FWCF
70	0.60	PRFDS	0.62	FWCF
60	0.77	PRFDS	0.69	PRFDS
50	0.97	FWCF	0.82	PRFDS
40	1.06	PRFDS	0.89	PRFDS
26	1.34	FWCF	1.13	PRFDS
26 at > 50%F below P_{bypass}	1.64	FWCF	1.44	FWCF
23 at > 50%F below P_{bypass}	1.61	FWCF	1.52	FWCF
26 at ≤ 50%F below P_{bypass}	1.76	FWCF	1.33	FWCF
23 at ≤ 50%F below P_{bypass}	1.87	FWCF	1.44	FWCF

Table 5.11 Limiting ΔMCPR Results for TLO Pump Seizure

Power / Flow (% of rated)	ATRIUM 11
100 / 99	0.29

Table 5.12 Limiting ΔMCPR Results for SLO Pump Seizure

Power / Flow (% of rated)	ATRIUM 11
67.2 / 52	0.77

Table 5.13 Limiting ΔCPR Results for Loss of Feedwater Heating

Core Power (% of rated)	ATRIUM 11
100	0.146
90	0.176
80	0.210
70	0.244
60	0.278
50	0.312
40	0.346
30	0.381

Table 5.14 ATRIUM-10 Control Rod Withdrawal Error Main Turbine Bypass Operable MCPR Versus RBM Setpoint for MCPR Safety Limit = 1.08*

High Power Range			Intermediate Power Range			Low Power Range		
RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR
108	100	1.29	113	85	1.36	118	65	1.58
	85	1.30		65	1.46		30	1.64
109	100	1.29	114	85	1.47	119	65	1.58
	85	1.32		65	1.58		30	1.67
110	100	1.30	115	85	1.47	120	65	1.58
	85	1.33		65	1.58		30	1.67
111	100	1.32	116	85	1.47	121	65	1.58
	85	1.34		65	1.58		30	1.67
112	100	1.32	117	85	1.47	122	65	1.58
	85	1.34		65	1.58		30	1.67
113	100	1.34	118	85	1.47	123	65	1.58
	85	1.36		65	1.58		30	1.67
114	100	1.43	119	85	1.47	124	65	1.58
	85	1.47		65	1.58		30	1.67
115	100	1.43	120	85	1.47	125	65	1.58
	85	1.47		65	1.58		30	1.67
116	100	1.43	121	85	1.47	126	65	1.58
	85	1.47		65	1.58		30	1.67
117	100	1.43	122	85	1.47	127	65	1.58
	85	1.47		65	1.58		30	1.67

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* Operation with one TCV/TSV closed is only supported at power levels ≤ 75% of rated.

Table 5.15 ATRIUM-10 Control Rod Withdrawal Error Main Turbine Bypass Inoperable MCPR Versus RBM Setpoint for MCPR Safety Limit = 1.08

High Power Range			Intermediate Power Range			Low Power Range		
RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR
108	>95	1.52	113	85	1.36	118	65	1.58
	85	1.30		65	1.46		30	1.64
109	>95	1.54	114	85	1.47	119	65	1.58
	85	1.32		65	1.58		30	1.67
110	>95	1.55	115	85	1.47	120	65	1.58
	85	1.33		65	1.58		30	1.67
111	>95	1.56	116	85	1.47	121	65	1.58
	85	1.34		65	1.58		30	1.67
112	>95	1.56	117	85	1.47	122	65	1.58
	85	1.34		65	1.58		30	1.67
113	>95	1.58	118	85	1.47	123	65	1.58
	85	1.36		65	1.58		30	1.67
114	>95	1.69	119	85	1.47	124	65	1.58
	85	1.47		65	1.58		30	1.67
115	>95	1.69	120	85	1.47	125	65	1.58
	85	1.47		65	1.58		30	1.67
116	>95	1.69	121	85	1.47	126	65	1.58
	85	1.47		65	1.58		30	1.67
117	>95	1.69	122	85	1.47	127	65	1.58
	85	1.47		65	1.58		30	1.67

Table 5.16 ATRIUM 11 Control Rod Withdrawal Error Main Turbine Bypass Operable MCPR Versus RBM Setpoint for MCPR Safety Limit = 1.08*

High Power Range			Intermediate Power Range			Low Power Range		
RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR
108	100	1.29	113	85	1.37	118	65	1.46
	85	1.30		65	1.39		30	1.66
109	100	1.30	114	85	1.40	119	65	1.46
	85	1.31		65	1.43		30	1.66
110	100	1.31	115	85	1.40	120	65	1.46
	85	1.32		65	1.43		30	1.66
111	100	1.35	116	85	1.43	121	65	1.46
	85	1.37		65	1.46		30	1.69
112	100	1.35	117	85	1.43	122	65	1.46
	85	1.37		65	1.46		30	1.69
113	100	1.35	118	85	1.43	123	65	1.46
	85	1.37		65	1.46		30	1.69
114	100	1.36	119	85	1.43	124	65	1.46
	85	1.40		65	1.46		30	1.69
115	100	1.39	120	85	1.43	125	65	1.46
	85	1.40		65	1.46		30	1.69
116	100	1.39	121	85	1.43	126	65	1.46
	85	1.43		65	1.46		30	1.69
117	100	1.39	122	85	1.43	127	65	1.46
	85	1.43		65	1.46		30	1.69

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* Operation with one TCV/TSV closed is only supported at power levels ≤ 75% of rated.

Table 5.17 ATRIUM 11 Control Rod Withdrawal Error Main Turbine Bypass Inoperable MCPR Versus RBM Setpoint for MCPR Safety Limit = 1.08

High Power Range			Intermediate Power Range			Low Power Range		
RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR	RBM Trip Setpoint (%)	Core Power (% rated)	MCPR
108	>95	1.51	113	85	1.37	118	65	1.46
	85	1.30		65	1.39		30	1.66
109	>95	1.52	114	85	1.40	119	65	1.46
	85	1.31		65	1.43		30	1.66
110	>95	1.53	115	85	1.40	120	65	1.46
	85	1.32		65	1.43		30	1.66
111	>95	1.58	116	85	1.43	121	65	1.46
	85	1.37		65	1.46		30	1.69
112	>95	1.58	117	85	1.43	122	65	1.46
	85	1.37		65	1.46		30	1.69
113	>95	1.58	118	85	1.43	123	65	1.46
	85	1.37		65	1.46		30	1.69
114	>95	1.61	119	85	1.43	124	65	1.46
	85	1.40		65	1.46		30	1.69
115	>95	1.61	120	85	1.43	125	65	1.46
	85	1.40		65	1.46		30	1.69
116	>95	1.64	121	85	1.43	126	65	1.46
	85	1.43		65	1.46		30	1.69
117	>95	1.64	122	85	1.43	127	65	1.46
	85	1.43		65	1.46		30	1.69

Table 5.18 ATRIUM-10 and ATRIUM 11 CRWE LHGRFAC$_p$ Multipliers Main Turbine Bypass Inoperable

Thermal Power (% of Rated)	LHGRFAC$_p$
0 - 95	1.0
95 - 100	0.90

Table 5.19 RBM System
Operational Requirements

Thermal Power (% of Rated)	TBV in Service MCPR*	TBVOOS MCPR*
≥ 30 and < 90	< 1.71 for TLO < 1.76 for SLO	< 1.71 for TLO < 1.76 for SLO
≥ 90 and < 95	< 1.47 for TLO NA for SLO	< 1.47 for TLO NA for SLO
≥ 95	< 1.47 for TLO NA for SLO	< 1.68 for TLO NA for SLO

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* The MCPR values shown correspond to an SLMCPR of 1.08 for TLO and 1.11 for SLO.

Table 5.20 Licensing Basis Core Average Axial Power Profile

State Conditions for Power Shape Evaluation	
Power, MWt	3952
Dome pressure, psia	1050.4
Inlet subcooling, Btu/lbm	24.4
Flow, Mlbm/hr	108
Control state	ARO
Core Average Exposure, MWd/MTU	36,735

Licensing Axial Power Profile
(Normalized)

Node	Power
Top 25	0.292
24	0.826
23	1.062
22	1.243
21	1.339
20	1.383
19	1.400
18	1.398
17	1.380
16	1.409
15	1.403
14	1.355
13	1.303
12	1.242
11	1.160
10	1.108
9	1.039
8	0.945
7	0.841
6	0.735
5	0.626
4	0.537
3	0.477
2	0.387
Bottom 1	0.112

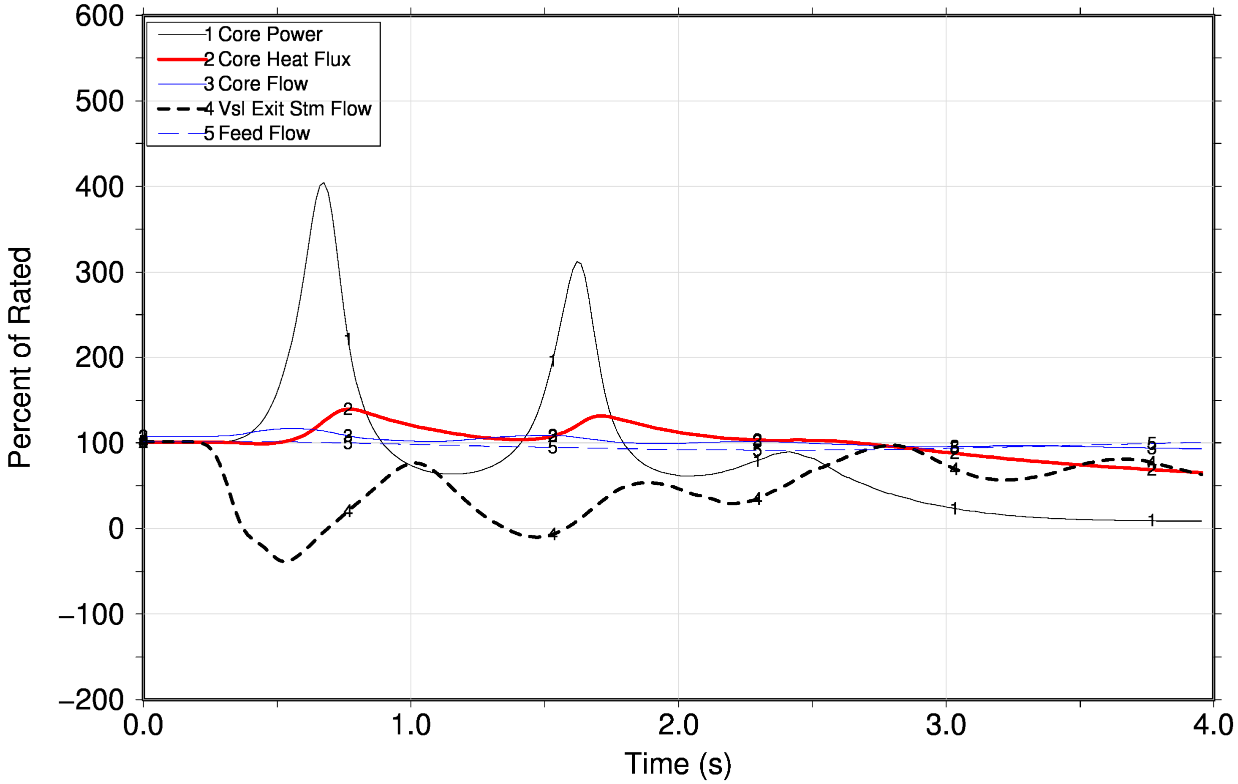

Figure 5.1 Load Rejection/Turbine Trip at 100P/108F Maximum Allowable Average Scram Insertion Time Key Parameters

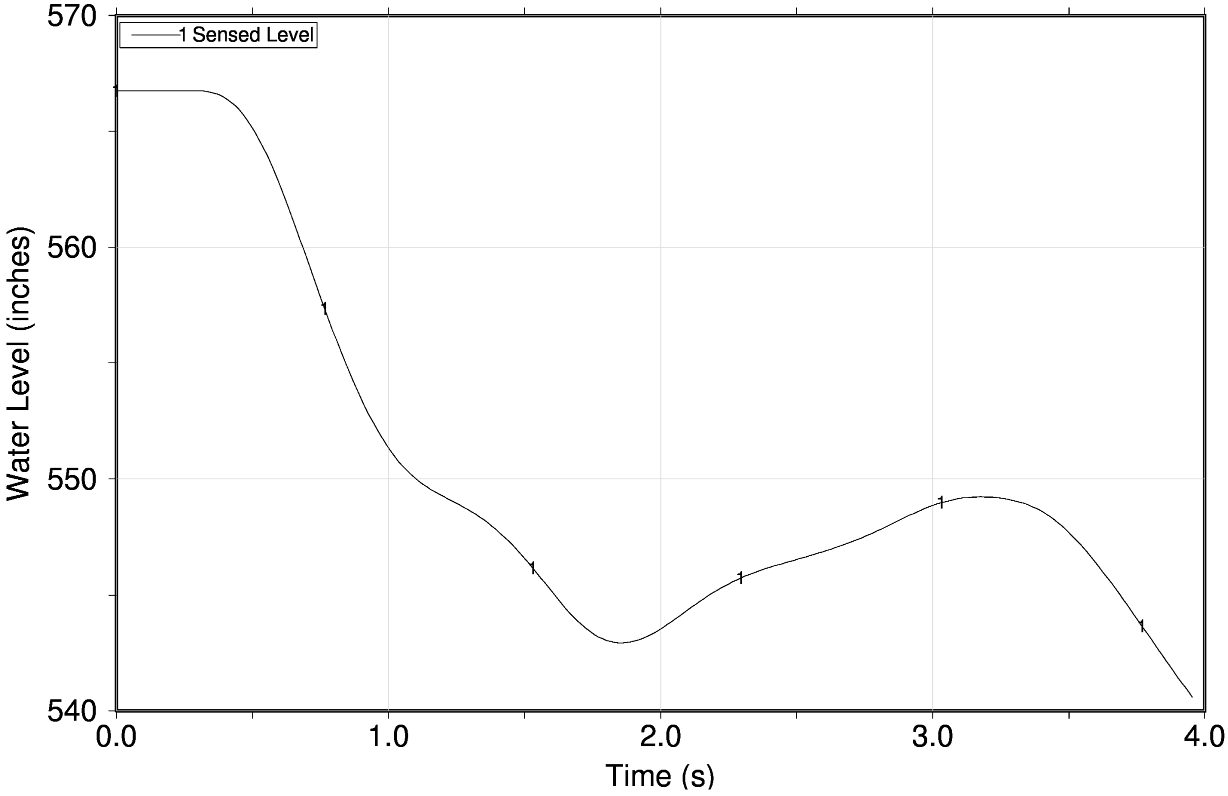

**Figure 5.2 Load Rejection/Turbine Trip at 100P/108F
Maximum Allowable Average Scram Insertion Time
Sensed Water Level**

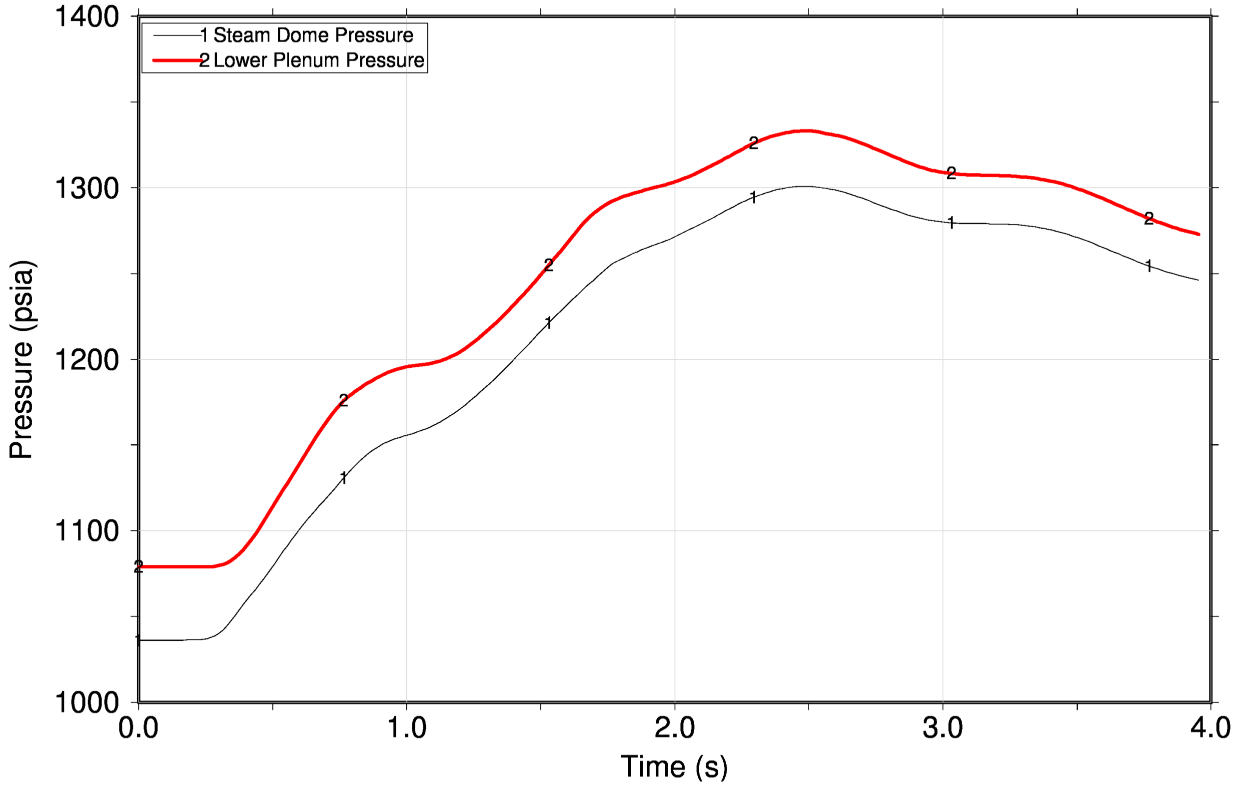

Figure 5.3 Load Rejection/Turbine Trip at 100P/108F Maximum Allowable Average Scram Insertion Time Vessel Pressures

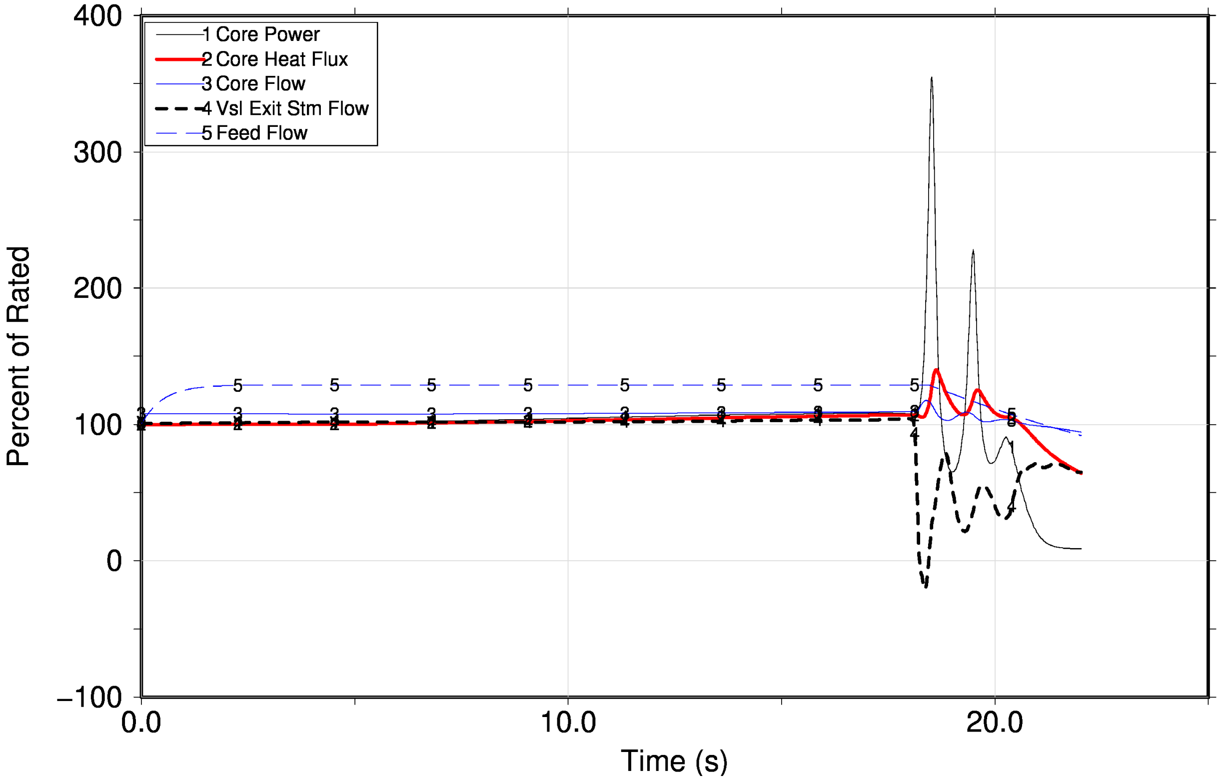

Figure 5.4 Feedwater Controller Failure at 100P/108F Maximum Allowable Average Scram Insertion Time Key Parameters

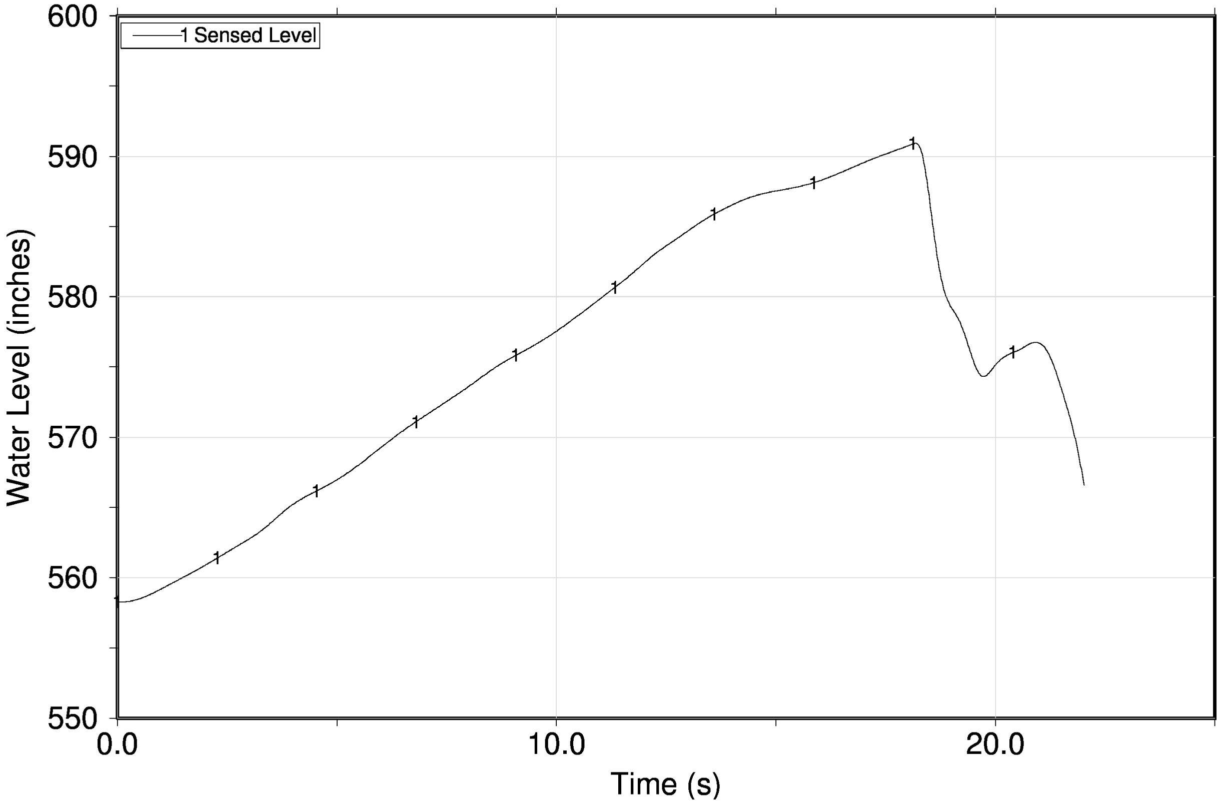

**Figure 5.5 Feedwater Controller Failure at 100P/108F
Maximum Allowable Average Scram Insertion Time
Sensed Water Level**

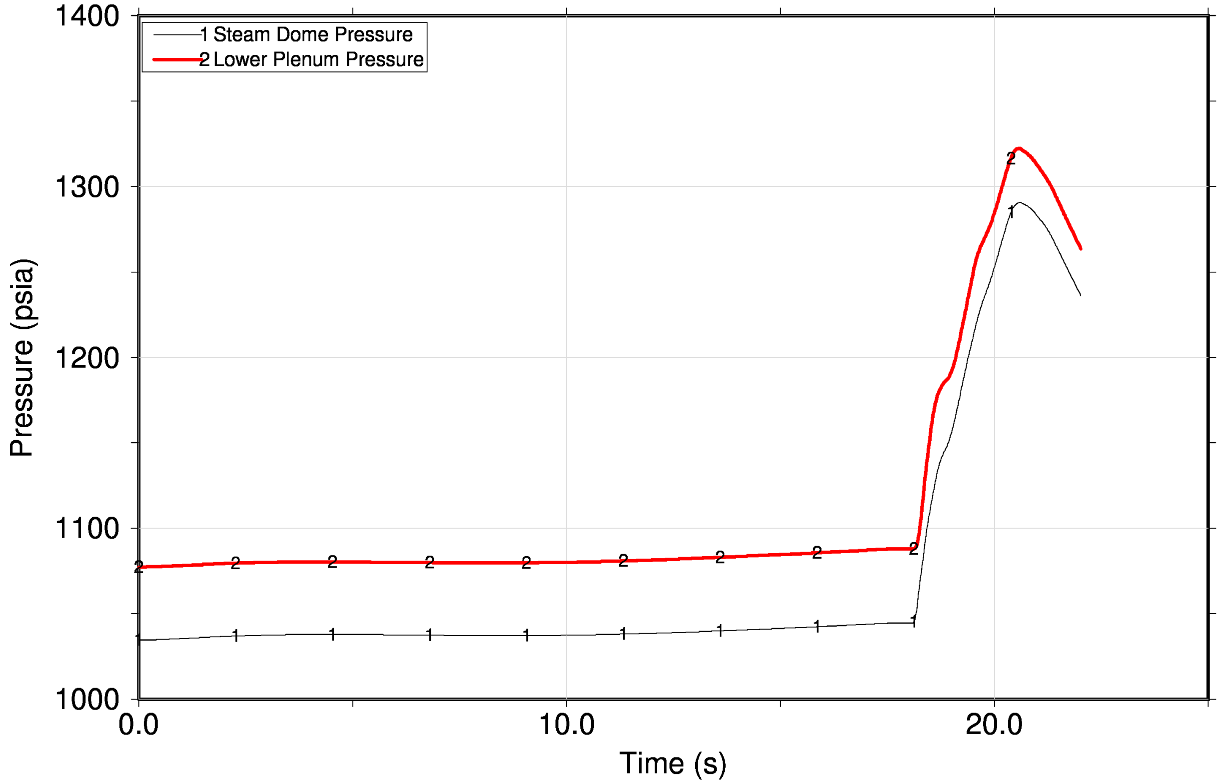

Figure 5.6 Feedwater Controller Failure at 100P/108F Maximum Allowable Average Scram Insertion Time Vessel Pressures

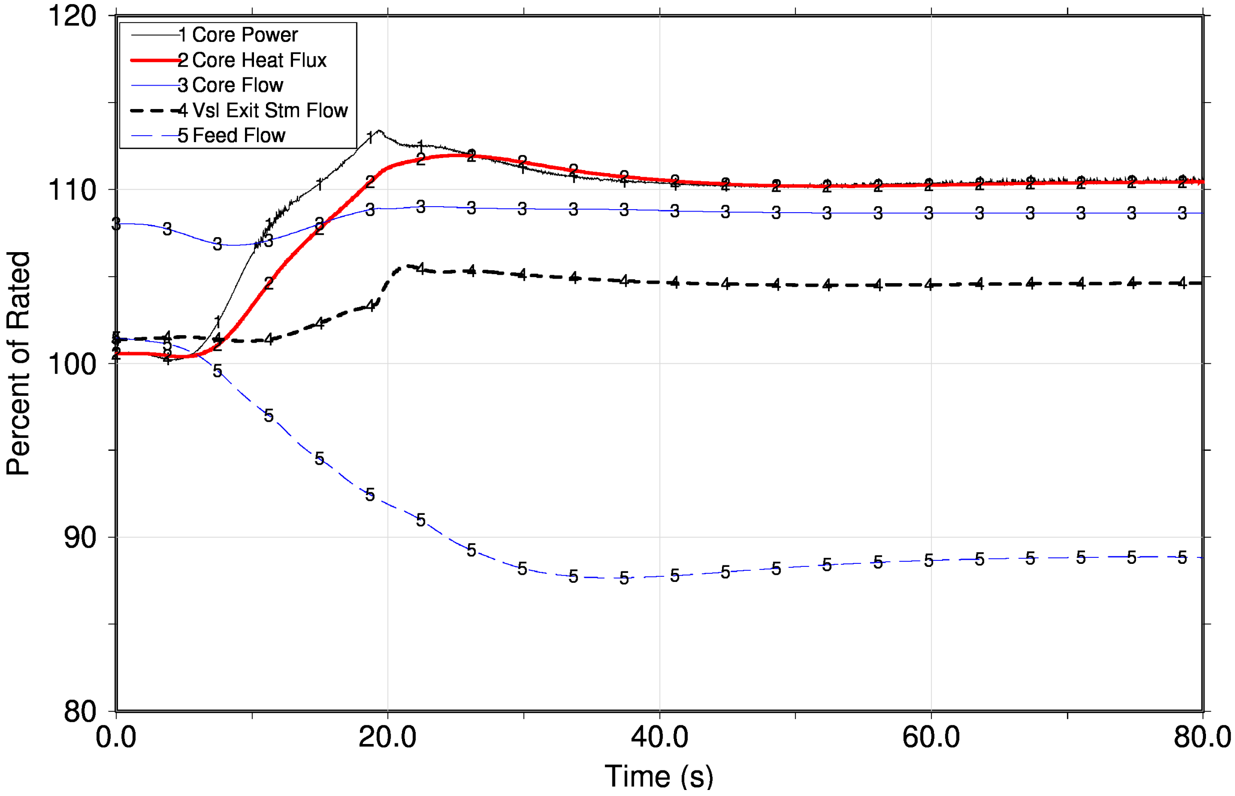

Figure 5.7 Inadvertent Startup of the HPCI Pump at 100P/108F
Key Parameters

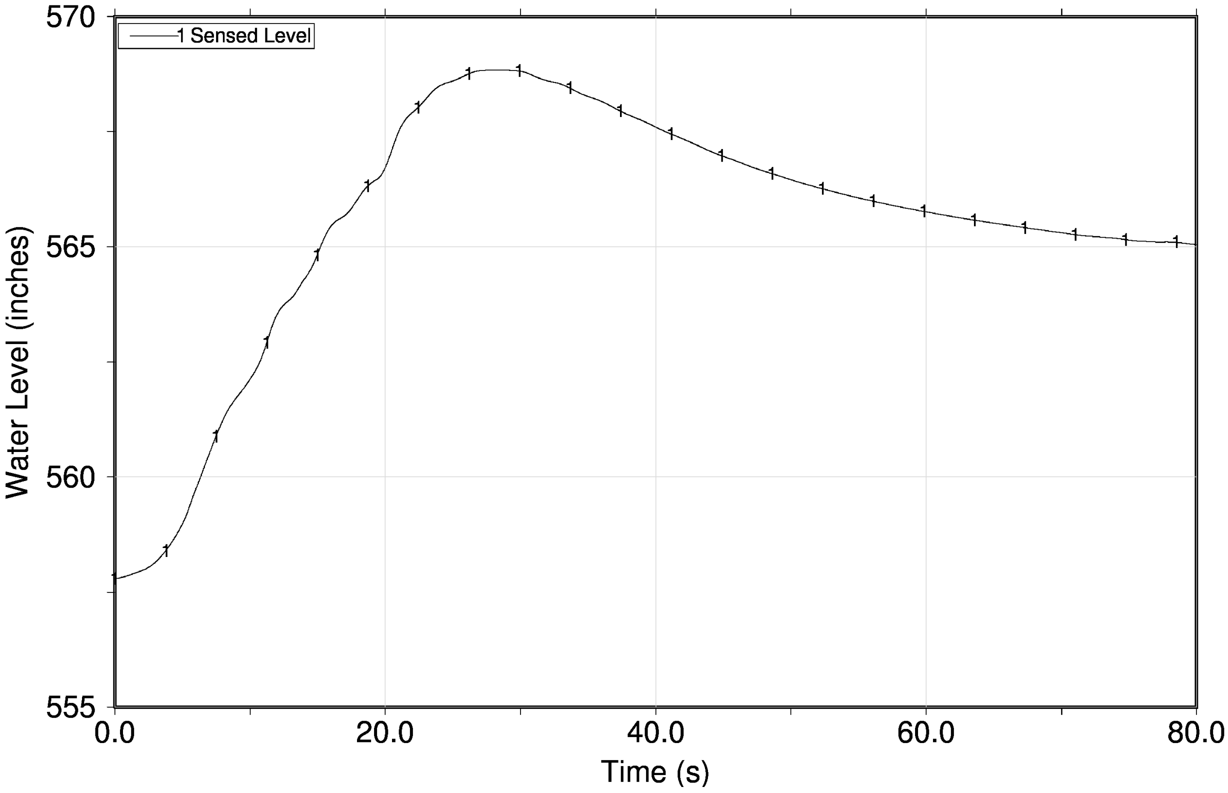

Figure 5.8 Inadvertent Startup of the HPCI Pump at 100P/108F Sensed Water Level

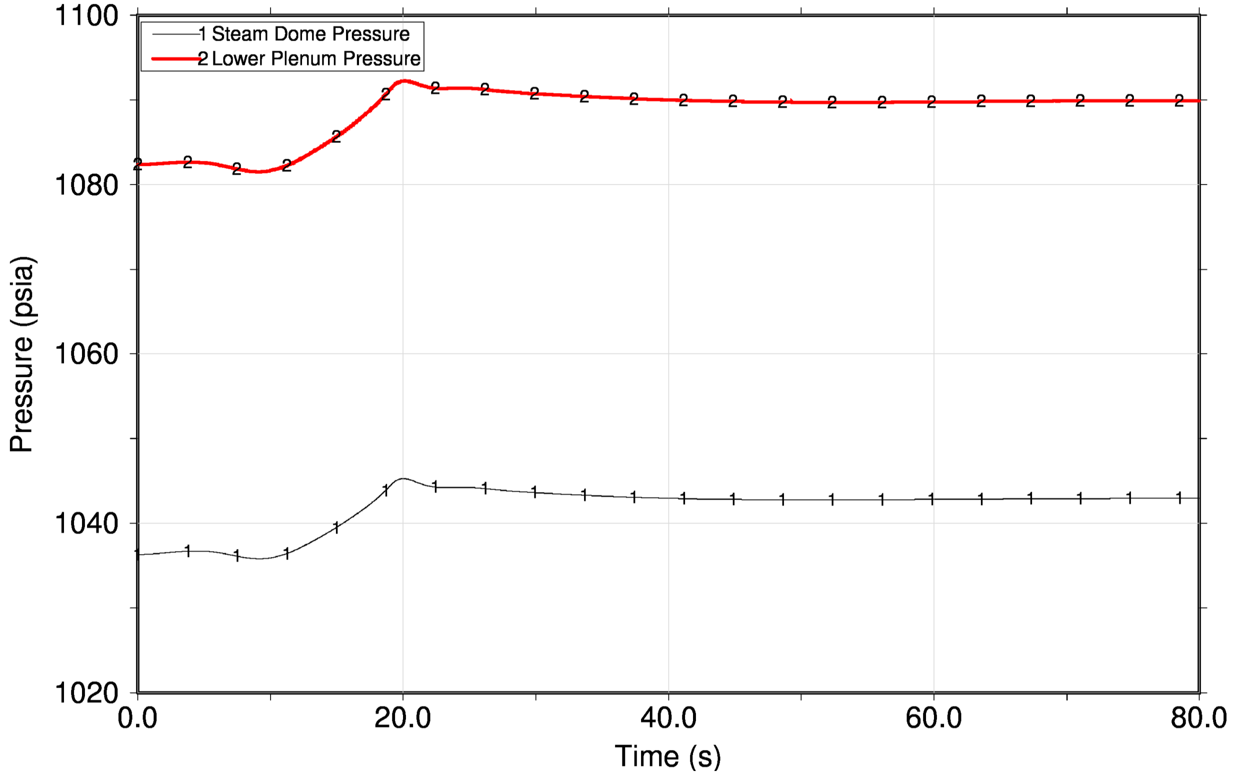

Figure 5.9 Inadvertent Startup of the HPCI Pump at 100P/108F Vessel Pressures

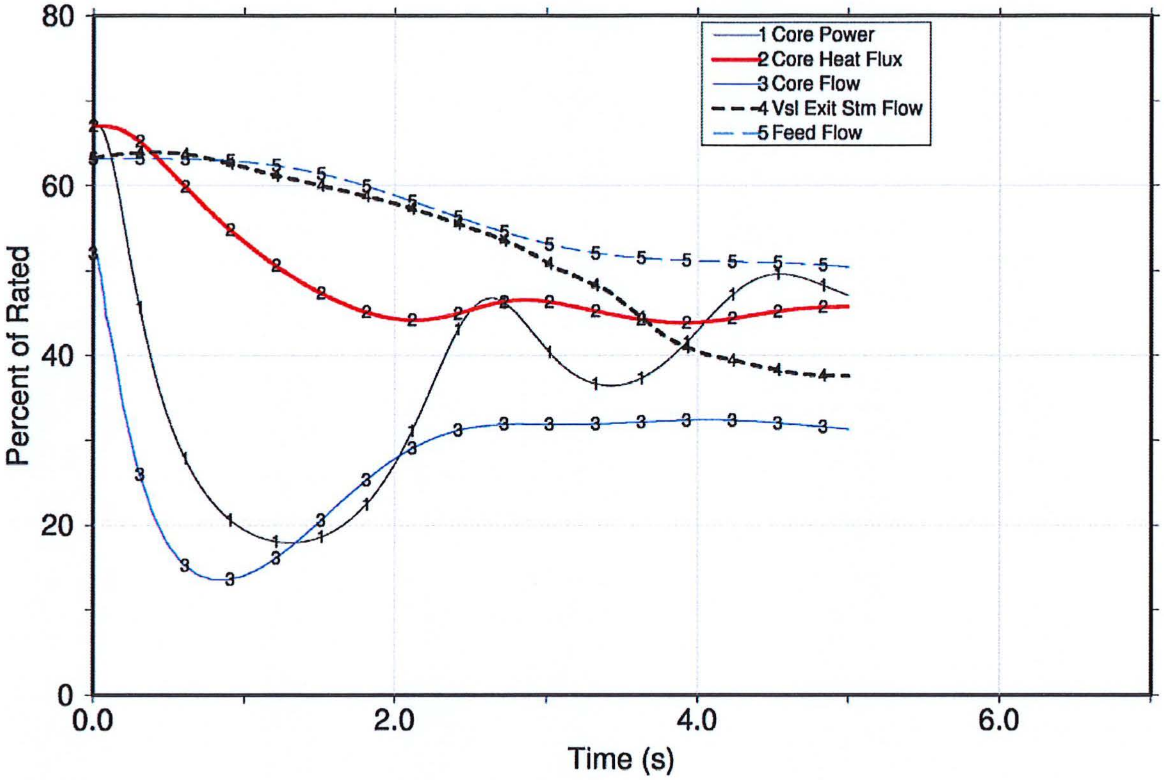

Figure 5.10 SLO Pump Seizure at 67.2P/52F Maximum Allowable Average Scram Insertion Time Key Parameters

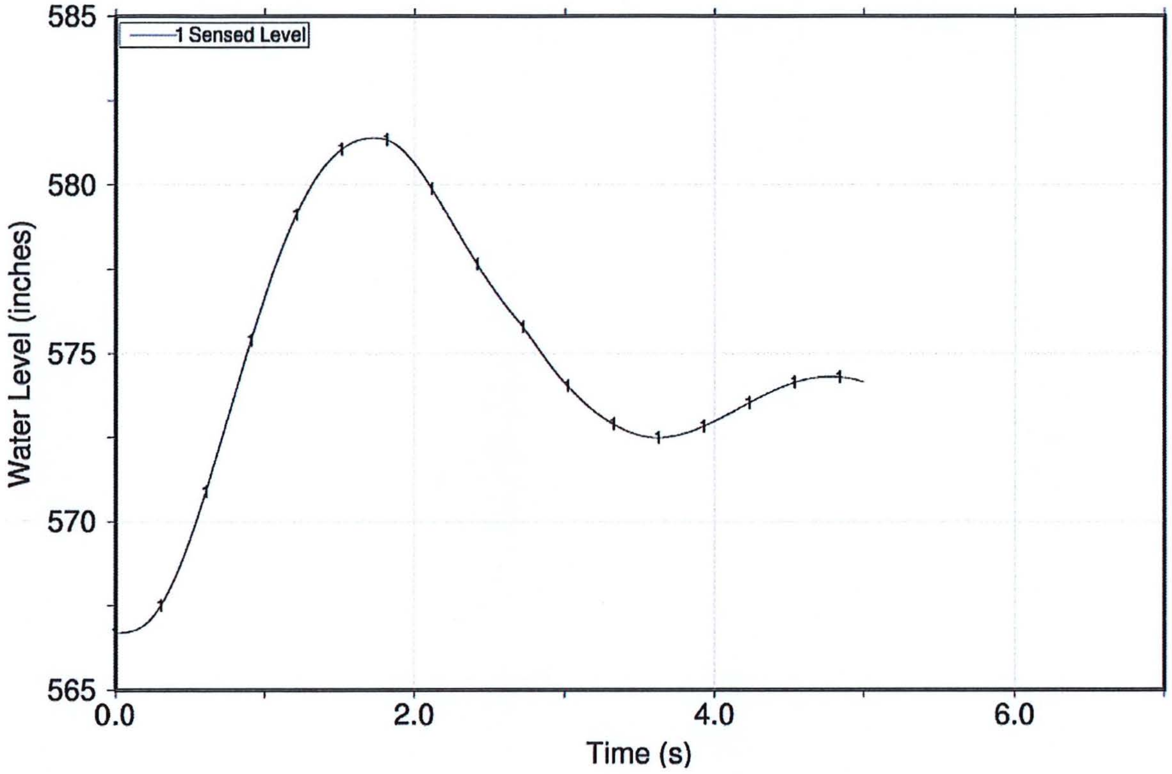

Figure 5.11 SLO Pump Seizure at 67.2P/52F
Maximum Allowable Average Scram Insertion Time
Sensed Water Level

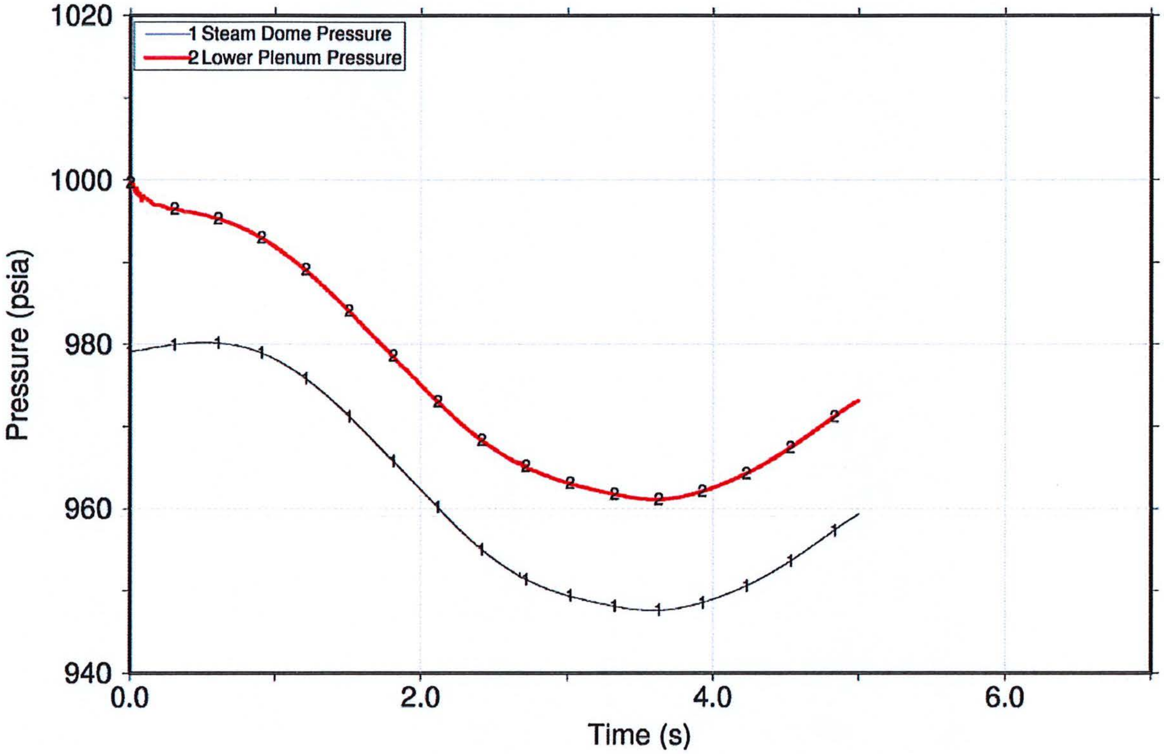

Figure 5.12 SLO Pump Seizure at 67.2P/52F Maximum Allowable Average Scram Insertion Time Vessel Pressures

6.0 POSTULATED ACCIDENTS

6.1 *Loss-of-Coolant Accident (LOCA)*

The results of the ATRIUM-10 LOCA analysis are presented in References 27 and 28 and provide a PCT of 1,844°F, and supplemented by Reference 29. The peak local metal water reaction is 0.80% and the core wide metal water reaction is < 0.2%. The ATRIUM-10 SLO MAPLHGR multiplier is 0.80. The cycle-specific OLMCPRs bound those assumed in References 27 and 28.

The results of the ATRIUM 11 LOCA analysis are presented in Reference 30 and provide a PCT of 1,784°F, and supplemented by Reference 29. The peak local metal water reaction is 4.64% and the core wide metal water reaction is 0.30%. The ATRIUM 11 SLO MAPLHGR multiplier is 0.80. The cycle-specific OLMCPRs bound those assumed in Reference 30. The ATRIUM 11 LOCA analyses are based on the **[**

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6.2 *Control Rod Drop Accident*

Framatome performed a Cycle 21 specific control rod drop accident analysis using the AURORA-B CRDA methodology (Reference 31) assuming the criteria of DG-1327 (Reference 32). The analysis supports startup with all allowable BPWS group combinations for A or B sequences, as defined in Reference 33. The maximum number of inoperable / slow control rods (per the technical specifications) are assumed and no intermediate banking will be credited for the second group of rod drops. The inoperable rod locations for this analysis are determined as part of the analysis as described in ANP-3771P (Reference 34). The following results are based on the Cycle 21 core design provided in Reference 2.

The final results show all cases analyzed pass requirements.

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Assembly	Melt	High Temp	PCMI	Combined
21B148	0	9	9	9
21B149	2	16	12	16
21B150	0	9	9	9
Total	2	34	30	34
Dose Equivalent			Ratio	Rods
			1.61	55

6.3 *Fuel Handling and Equipment Handling Accident*

Framatome has performed a review of the fuel and equipment handling accident for the ATRIUM 11 fuel assembly and determined that up to 182 fuel rods could fail if an ATRIUM-10 fuel assembly were dropped onto other ATRIUM 11 assemblies in the reactor core. In addition, it was determined that up to 203 fuel rods could fail if an ATRIUM 11 fuel assembly were dropped onto other ATRIUM 11 assemblies in the reactor core. It was also determined that up to 455 fuel rods could fail if a fuel assembly and the handling equipment were dropped onto other assemblies in the reactor core. These failed ATRIUM 11 fuel rod values should be used in the alternate source term dose rate calculation performed by Susquehanna for the fuel and equipment handling accident.

The FHA/EHA analysis is fuel type dependent and does not need to be repeated on a cycle specific basis.

6.4 *Fuel Loading Error (Infrequent Event)*

There are two types of fuel loading errors possible in a BWR: the mislocation of a fuel assembly in a core position prescribed to be loaded with another fuel assembly, and the misorientation of a fuel assembly with respect to the control blade. As described in Reference 35, the fuel loading error is characterized as an infrequent event. The acceptance criteria are that the offsite dose consequences due to the event shall not exceed a small fraction of the 10 CFR 50.67 limits.

6.4.1 Mislocated Fuel Bundle

Framatome has performed fuel mislocation error analyses for SUS2-21. This analysis evaluated the impact of a mislocated assembly against potential fuel rod failure mechanisms due to increased LHGR and reduced CPR. Based on this analysis, the offsite dose criteria (a small fraction of 10 CFR 50.67) is conservatively satisfied. A dose consequence evaluation is not necessary since no rod approached the fuel centerline melt or 1% strain limits, and less than 0.1% of the fuel rods are expected to experience boiling transition which could result in a dryout induced failure.

6.4.2 Misoriented Fuel Bundle

Framatome has performed fuel assembly misorientation analysis for SUS2-21. The analysis was performed assuming the limiting assembly was loaded in the worst orientation (rotated 180°) and depleted through the cycle without operator interaction. The analysis demonstrates the small fraction of 10 CFR 50.67 offsite dose criteria is conservatively satisfied. A dose consequence evaluation is not necessary since no rod approached the fuel centerline melt or 1% strain limits and less than 0.1% of the fuel rods are expected to experience boiling transition.

7.0 SPECIAL ANALYSES

7.1 *ASME Overpressurization Analysis*

The ASME overpressurization analysis is performed to demonstrate compliance with the ASME Boiler and Pressure Vessel Code. The analysis shows the Susquehanna Unit 2 safety valves have sufficient capacity and performance to prevent the reactor vessel pressure from reaching the safety limit of 110% of the design pressure.

Maximum system pressures could potentially result from closure of the main steam isolation valves (MSIV), closure of the TCVs, or closure of the TSVs. 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. For Susquehanna Unit 2 Cycle 21, **[**

] MSIV closure, TCV closure, and TSV closure runs were first performed for 102% power and 108 Mlbm/hr flow and 99 Mlbm/hr flow at the highest Cycle 21 exposure where rated power operation can be attained. **[**

] The analyses were performed with the following assumptions:

- No credit for direct scram on MSIV or TSV valve position or TCV fast closure (scram is delayed until the second safety-grade signal for high neutron flux or high dome pressure).
- Maximum Allowable Average Scram Insertion Time
- No credit for RPT on TSV position or TCV motion (RPT delayed until high dome pressure signal).
- No credit for opening of the turbine bypass valves.
- No credit for the SRVs opening at the relief setpoints (open at safety setpoints).
- The 2 lowest setpoint SRVs were assumed inoperable.
- Initial dome pressure at 1,064.7 psia.
- A fast MSIV closure time of 2 seconds was used for the MSIV closure case.

Results of the limiting MSIV closure overpressurization analyses are presented in Table 7.1. Figure 7.1 – Figure 7.3 show the response of various reactor plant parameters during the MSIV closure event resulting in the maximum vessel pressure. The maximum pressure of 1,342 psig occurs in the lower plenum. The maximum dome pressure is 1,309 psig. The results demonstrate the maximum vessel pressure limit of 1,375 psig and dome pressure limit of 1,325 psig are not exceeded.

7.2 *ATWS Event Evaluation*

7.2.1 ATWS Overpressurization Analysis

The anticipated transient without scram (ATWS) overpressurization analysis is performed to demonstrate 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). For Susquehanna Unit 2 Cycle 21, [ ] MSIV closure and pressure regulator failure open (PRFO) runs were first performed at 100% power at 108% and 99% flow. For the PRFO event, 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 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. For the MSIV closure event, the event is initiated by a fast closure of the MSIVs. This results in a pressurization wave that causes a decrease in core voids which results in an increase in core power and pressure.

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The analyses were performed with the following assumptions:

- The analytical limit ATWS-RPT setpoint and function were assumed.
- The 2 SRVs with the lowest relief setpoints were assumed inoperable.
- All scram functions were disabled.
- Initial dome pressure was set to the nominal pressure with a -15 psi uncertainty (1,035.4 psia).

- An MSIV closure time of 2.0 seconds is used for the MSIV closure event. An MSIV closure time of 5.0 seconds is used for the PRFO event.

Results of the limiting ATWS overpressurization analyses are presented in Table 7.2. Figure 7.4 – Figure 7.6 show the response of various reactor plant parameters during the MSIV closure event resulting in the maximum vessel pressure. The maximum pressure of 1,479 psig occurs in the lower plenum. The maximum dome pressure is 1,459 psig. The results demonstrate the ATWS maximum vessel pressure limit of 1,500 psig is not exceeded.

7.2.2 Long-Term Evaluation

Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This in turn may impact the amount of steam discharged to the suppression pool and containment. This impact is discussed in Section 8.3 of Reference 1.

7.3 *Standby Liquid Control System*

In the event that the control rod scram function becomes incapable of rendering the core in a shutdown state, the standby liquid control (SLC) system is required to be capable of bringing the reactor from full power to a cold shutdown condition at any time in the core life. The Susquehanna Unit 2 SLC system is required to be able to inject 660 ppm natural boron equivalent at 68°F into the reactor coolant (including a 25% allowance for imperfect mixing, leakage, and volume of other piping connected to the reactor). An analysis that demonstrates that the SLC system meets the required shutdown capability for Cycle 21 has been performed. The analysis was performed to support a coolant temperature of 349.65°F with a boron concentration equivalent to 660 ppm at 68°F. The temperature of 349.65°F corresponds to the RHR shutdown cooling system activation temperature, and represents the maximum reactivity condition with soluble boron in the coolant. The analysis shows the core to be subcritical throughout the cycle by at least 0.938% Δk. **[**

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7.4 *Fuel Criticality*

The new fuel storage vault criticality analysis for ATRIUM 11 fuel is presented in Reference 9. The spent fuel pool criticality analysis for ATRIUM 11 fuel is presented in Reference 10. The ATRIUM 11 fuel assemblies identified for loading in Cycle 21 meet both the new and spent fuel storage requirements (Reference 6).

7.5 *Strongest Rod Out Shutdown Margin*

The Susquehanna Unit 2 Cycle 21 core has a minimum strongest rod out shutdown margin of 1.23 %Δk. This value is produced at 18,790 MWd/MTU at the coolant temperature condition of 200°F. The BOC 21 shutdown margin is calculated to be 1.44% Δk and therefore the R value is 0.21 %Δk. These values assume that SUS2-20 ended operation at the lowest allowable exposure (short window exposure).

Table 7.1 Maximum System Pressures for ASME Overpressurization

Power/Flow (% rated)	Transient	Vessel Lower Plenum (psig)	Steam Dome (psig)
102 / 108	MSIV closure	1,342	1,308
102 / 99	MSIV closure	1,341	1,309

Table 7.2 Maximum System Pressures for ATWS Overpressurization

Power/Flow (% rated)	Transient	Vessel Lower Plenum (psig)	Steam Dome (psig)
100 / 99	MSIV closure	1,479	1,459

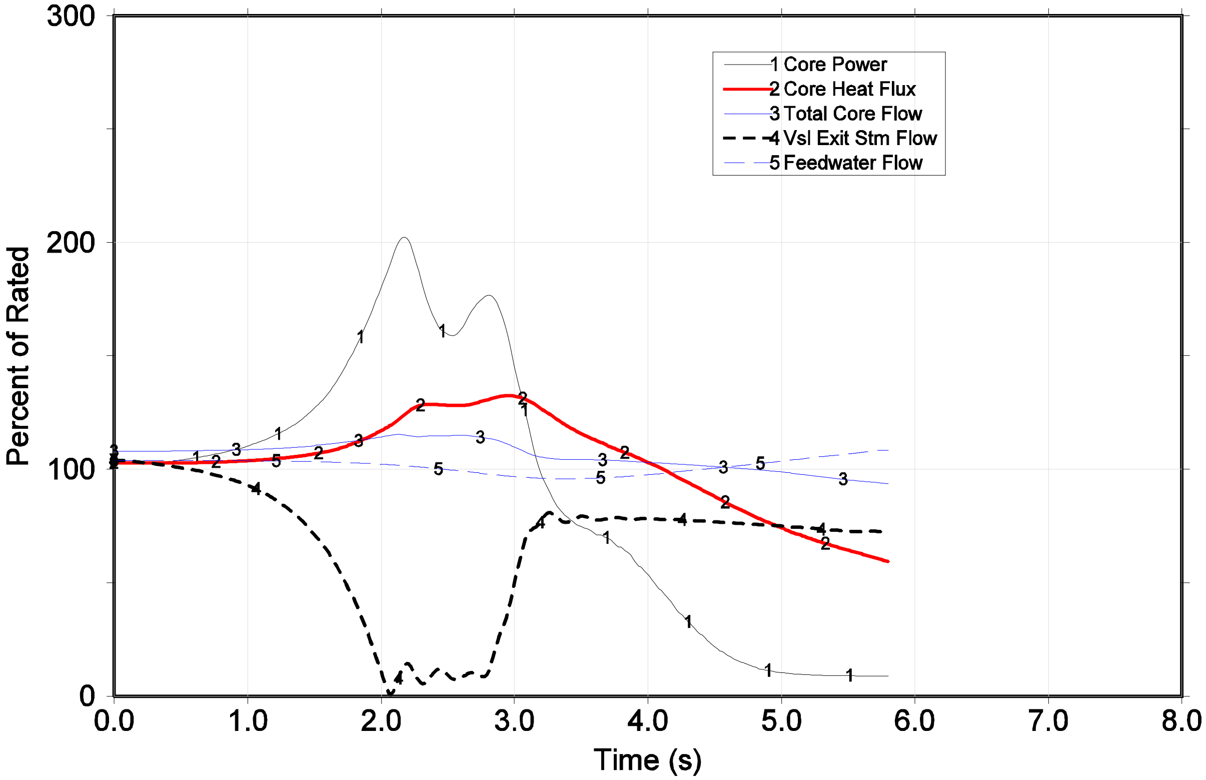

Figure 7.1 MSIV Closure ASME Overpressure Event at 102P/108F – Key Parameters

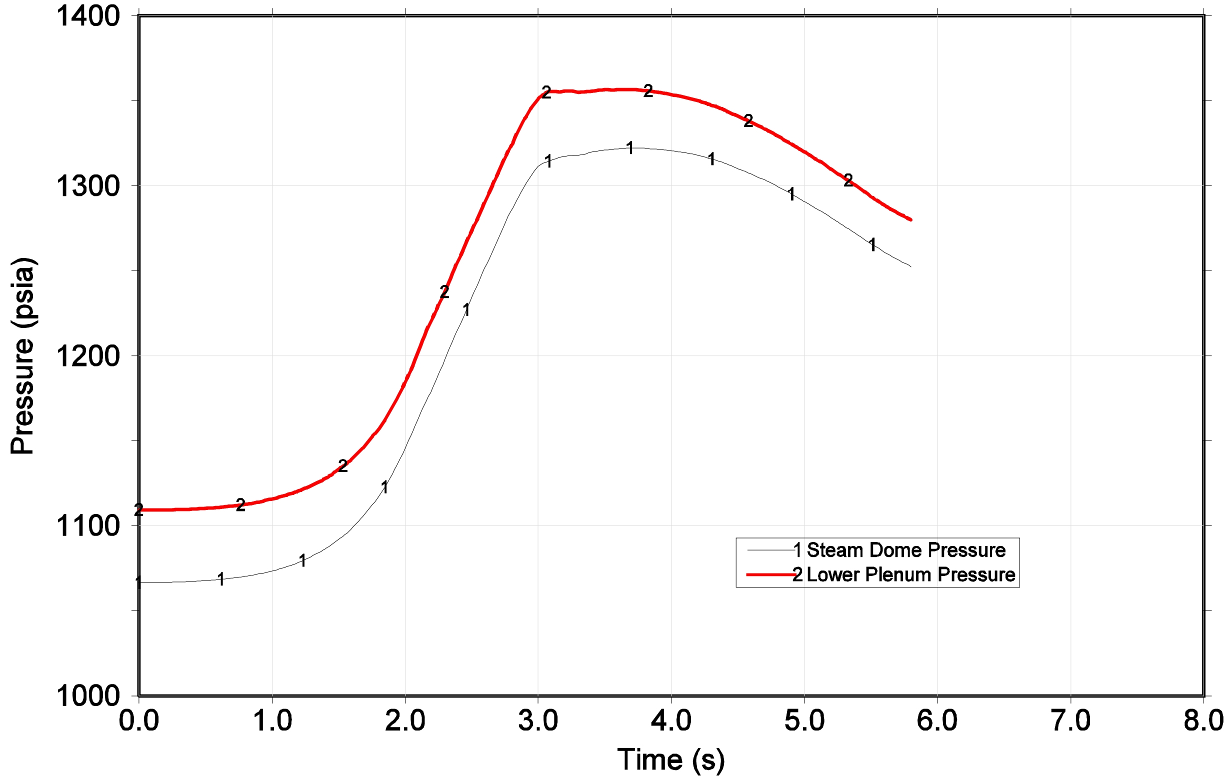

Figure 7.2 MSIV Closure ASME Overpressure Event at 102P/108F – Vessel Pressures

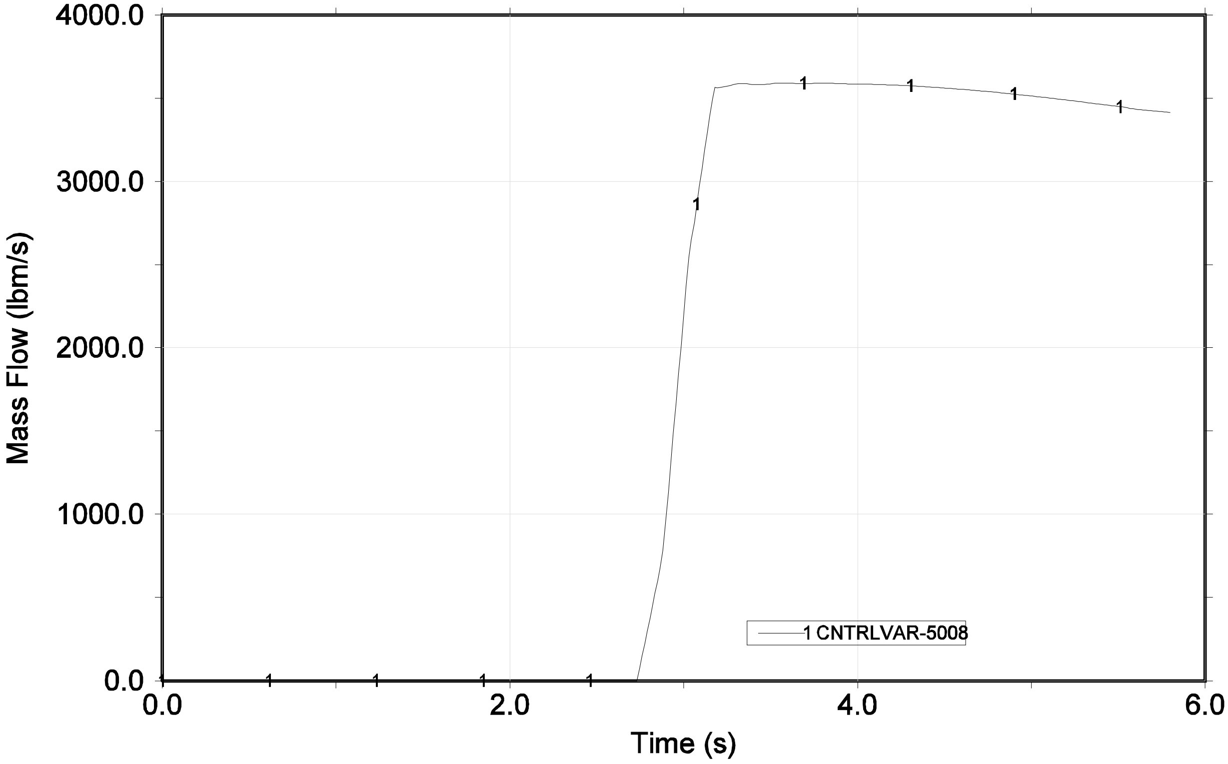

Figure 7.3 MSIV Closure ASME Overpressure Event at 102P/108F – Safety Valve Flow Rates

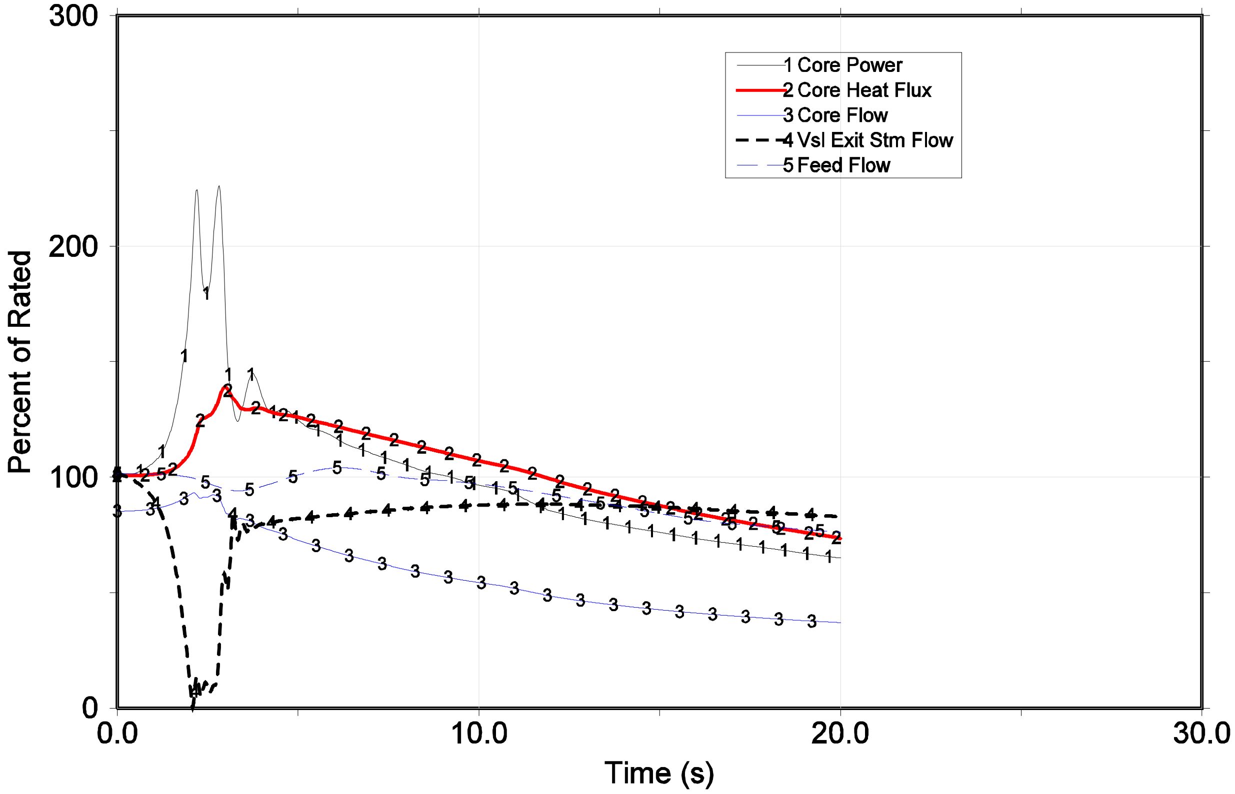

Figure 7.4 MSIV Closure ATWS Overpressurization Event at 100P/99F – Key Parameters

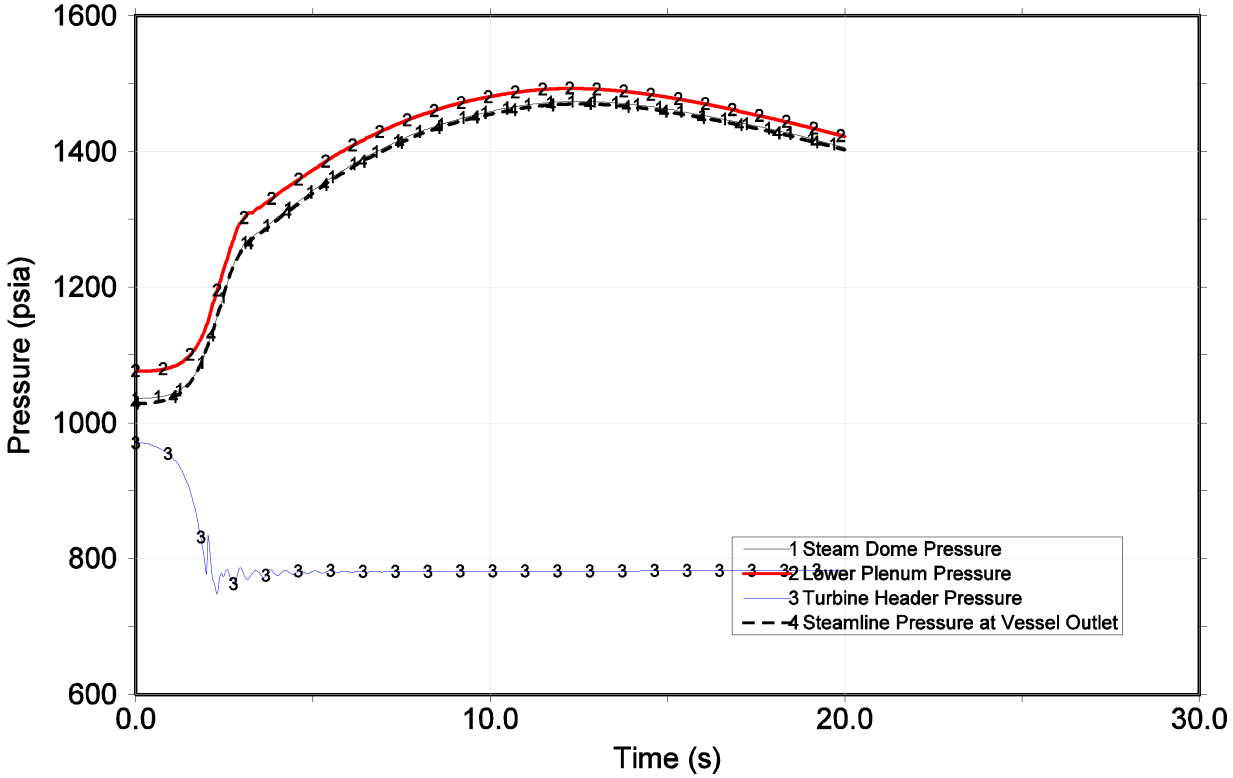

Figure 7.5 MSIV Closure ATWS Overpressurization Event at 100P/99F – Vessel Pressures

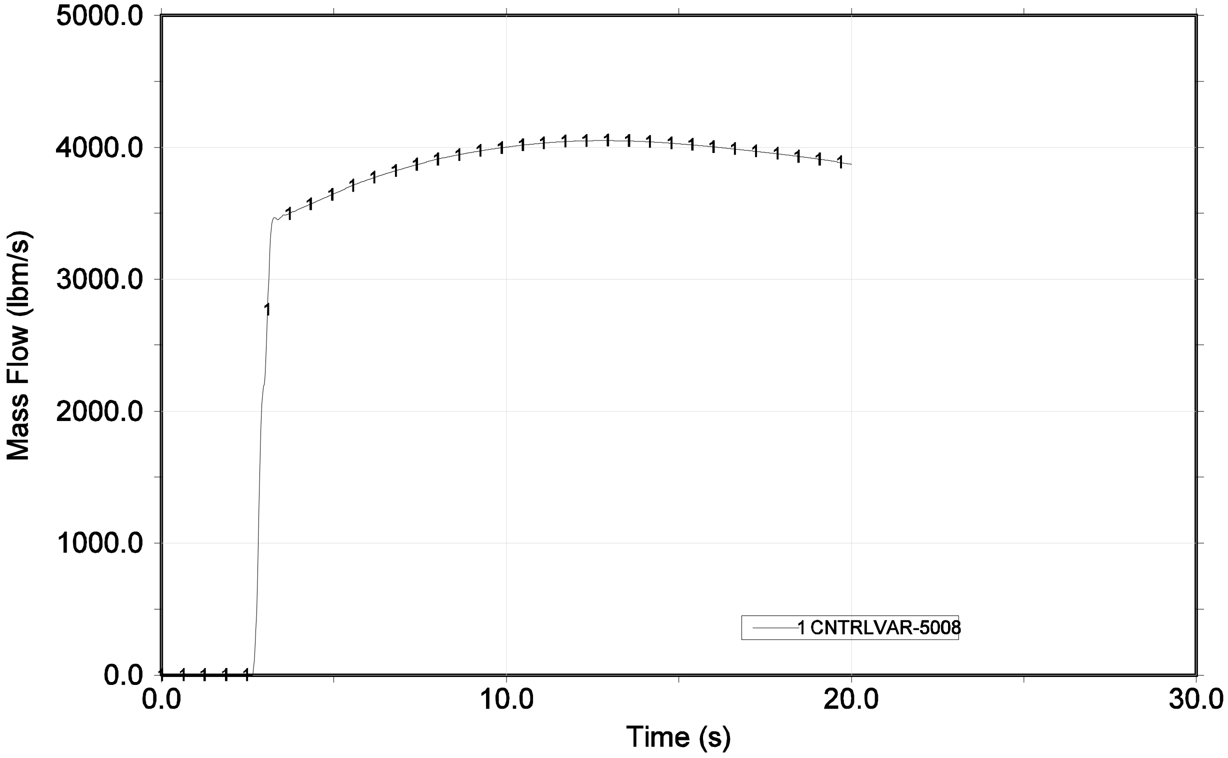

Figure 7.6 MSIV Closure ATWS Overpressurization Event at 100P/99F – Relief Valve Flow Rates

8.0 OPERATING LIMITS AND COLR INPUT

8.1 *MCPR Limits*

The determination of the MCPR limits is based on the analyses of the limiting anticipated operational occurrences (AOOs). For Susquehanna Unit 2 Cycle 21, **[**

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The MCPR operating limits are established so less than 0.1% of the fuel rods in the core are expected to experience boiling transition during an AOO initiated from rated or off-rated conditions and are based on a two-loop operation SLMCPR of 1.08 and a single-loop operation SLMCPR of 1.11. The MCPR limits were established to support operation from BOC to the maximum licensing core exposure (i.e., EOC), including coastdown, as defined by the core average exposures listed in Table 5.1. MCPR limits are established to support base case operation and the EOOS scenarios presented in Table 1.1.

Two-loop operation $MCPR_p$ limits for ATRIUM-10 and ATRIUM 11 fuel are presented in Table 8.1 and Table 8.2 for base case operation and EOOS conditions. Limits are presented for realistic average scram insertion times and maximum allowable average scram insertion times. Similarly, $MCPR_p$ limits for single-loop operation are provided in Table 8.3 and Table 8.4.

The results from the CRWE analysis (Table 5.14 through Table 5.17) are not used in establishing the $MCPR_p$ limits. Depending on the choice of RBM setpoints the CRWE analysis operating MCPR limit may be more limiting than the $MCPR_p$ limits. Therefore, Susquehanna may need to adjust these limits to account for CRWE results.

Since the pump seizure event for ATRIUM-10 was analyzed by another vendor, pump seizure results were not used in establishing the ATRIUM-10 $MCPR_p$ limits. Therefore, Susquehanna may need to adjust the ATRIUM-10 $MCPR_p$ limits to account for pump seizure results.

$MCPR_f$ limits are established to provide protection against fuel failures during a postulated slow flow excursion. The $MCPR_f$ limits for ATRIUM-10 and ATRIUM 11 fuel are presented in Table 8.5 for base case operation and EOOS conditions. These $MCPR_f$ limits are applicable for both realistic and maximum allowable average scram insertion times.

8.2 *LHGR Limits*

The LHGR limits for ATRIUM-10 fuel are presented in Table 8.6 (References 12 and 13). The LHGR limits for ATRIUM 11 fuel are presented in Table 8.7 (Reference 7). $LHGRFAC_p$ and $LHGRFAC_f$ multipliers are applied directly to the LHGR limits to protect against fuel melting and overstraining of the cladding during an AOO for both UO_2 and gadolinia bearing rods.

The ATRIUM 11 $LHGRFAC_p$ multipliers are determined using the RODEX4 thermal-mechanical methodology (Reference 36) using the AURORA-B transient simulations. For the $LHGRFAC_p$ evaluations [

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The ATRIUM-10 LHGRFAC$_p$ multipliers are used to ensure that Protection Against Power Transient (PAPT) LHGR limits, References 12 and 13, are not exceeded. For the LHGRFAC$_p$ evaluations **[**

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LHGRFAC$_p$ multipliers for ATRIUM-10 and ATRIUM 11 fuel are presented in Table 8.8 and Table 8.9 for base case operation and EOOS conditions. These LHGRFAC$_p$ multipliers are applicable for both realistic and maximum allowable average scram insertion times.

The results from the CRWE analysis (Table 5.18) are not used in establishing the LHGRFAC$_p$ multipliers. The CRWE analysis LHGRFAC$_p$ multipliers may be more limiting than the LHGRFAC$_p$ multipliers presented in Table 8.8 and Table 8.9. Therefore, Susquehanna may need to adjust these limits to account for CRWE results.

LHGRFAC$_f$ multipliers are established to provide protection against fuel centerline melt and overstraining of the cladding during a postulated slow flow excursion. The LHGRFAC$_f$ multipliers for ATRIUM-10 and ATRIUM 11 fuel are presented in Table 8.10 for base case operation and EOOS conditions. These LHGRFAC$_f$ multipliers are applicable for both realistic and maximum allowable average scram insertion times.

8.3 *MAPLHGR Limits*

The ATRIUM-10 TLO MAPLHGR limits are presented in Table 8.11. For operation in SLO, a multiplier of 0.80 must be applied to the TLO MAPLHGR limits.

The ATRIUM 11 TLO MAPLHGR limits are presented in Table 8.12. For operation in SLO, a multiplier of 0.80 must be applied to the TLO MAPLHGR limits.

Table 8.1 $MCPR_p$ Limits for Realistic Average Scram Insertion Time Two-Loop Operation (BOC to EOC)[*]

EOOS Condition	Core Power (% of Rated)	ATRIUM-10 $MCPR_p$[†]	ATRIUM 11 $MCPR_p$
Base case / 1 TCV/TSV closed[‡]	100	1.44	1.41
	80	1.53	1.53
	60	1.77	1.74
	50	1.84	1.84
	40	1.90	1.84
	26	2.27	2.02
	26 at > 50%F below P_{bypass}	2.92	2.57
	23 at > 50%F below P_{bypass}	3.03	2.65
	26 at ≤ 50%F below P_{bypass}	2.92	2.46
	23 at ≤ 50%F below P_{bypass}	3.03	2.57
TBVOOS	100	1.53	1.49
	80	1.53	1.55
	60	1.77	1.79
	50	1.85	1.89
	40	1.93	1.89
	26	2.30	2.08
	26 at > 50%F below P_{bypass}	3.23	3.07
	23 at > 50%F below P_{bypass}	3.38	3.31
	26 at ≤ 50%F below P_{bypass}	2.92	2.46
	23 at ≤ 50%F below P_{bypass}	3.09	2.67
RPTOOS	100	1.53	1.50
	80	1.53	1.54
	60	1.77	1.77
	50	1.84	1.87
	40	1.90	1.87
	26	2.27	2.04
	26 at > 50%F below P_{bypass}	2.92	2.57
	23 at > 50%F below P_{bypass}	3.03	2.65
	26 at ≤ 50%F below P_{bypass}	2.92	2.46
	23 at ≤ 50%F below P_{bypass}	3.03	2.57
PROOS	100	1.44	1.41
	80	1.54	1.53
	60	1.90	1.78
	50	1.99	1.92
	40	2.18	1.98
	26	2.47	2.25
	26 at > 50%F below P_{bypass}	2.92	2.57
	23 at > 50%F below P_{bypass}	3.03	2.65
	26 at ≤ 50%F below P_{bypass}	2.92	2.46
	23 at ≤ 50%F below P_{bypass}	3.03	2.57

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[*] These limits may need to be adjusted to account for the CRWE results provided in Table 5.14 - Table 5.17.

[†] The ATRIUM-10 limits may need to be adjusted to account for pump seizure results.

[‡] Operation with one TCV/TSV closed is supported for power levels ≤ 75% of rated.

Table 8.2 MCPR$_p$ Limits for Maximum Allowable Average Scram Insertion Time Two-Loop Operation (BOC to EOC)[*]

EOOS Condition	Core Power (% of Rated)	ATRIUM-10 MCPR$_p$[†]	ATRIUM 11 MCPR$_p$
Base case / 1 TCV/TSV closed[‡]	100	1.55	1.53
	80	1.61	1.69
	60	1.87	1.75
	50	2.10	1.91
	40	2.14	1.91
	26	2.49	2.12
	26 at > 50%F below P_{bypass}	2.92	2.57
	23 at > 50%F below P_{bypass}	3.03	2.65
	26 at ≤ 50%F below P_{bypass}	2.92	2.46
	23 at ≤ 50%F below P_{bypass}	3.03	2.57
TBVOOS	100	1.73	1.73
	80	1.73	1.79
	60	2.00	1.92
	50	2.22	2.03
	40	2.27	2.03
	26	2.58	2.26
	26 at > 50%F below P_{bypass}	3.32	3.17
	23 at > 50%F below P_{bypass}	3.47	3.41
	26 at ≤ 50%F below P_{bypass}	3.01	2.56
	23 at ≤ 50%F below P_{bypass}	3.18	2.77
RPTOOS	100	1.79	1.93
	80	1.82	2.11
	60	2.05	2.11
	50	2.22	2.12
	40	2.29	2.12
	26	2.61	2.35
	26 at > 50%F below P_{bypass}	3.04	2.77
	23 at > 50%F below P_{bypass}	3.15	2.85
	26 at ≤ 50%F below P_{bypass}	3.04	2.66
	23 at ≤ 50%F below P_{bypass}	3.15	2.77
PROOS	100	1.55	1.53
	80	1.68	1.69
	60	1.90	1.79
	50	2.10	1.92
	40	2.20	2.00
	26	2.49	2.25
	26 at > 50%F below P_{bypass}	2.92	2.57
	23 at > 50%F below P_{bypass}	3.03	2.65
	26 at ≤ 50%F below P_{bypass}	2.92	2.46
	23 at ≤ 50%F below P_{bypass}	3.03	2.57

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[*] These limits may need to be adjusted to account for the CRWE results provided in Table 5.14 - Table 5.17.

[†] The ATRIUM-10 limits may need to be adjusted to account for pump seizure results.

[‡] Operation with one TCV/TSV closed is supported for power levels ≤ 75% of rated.

Table 8.3 MCPR$_p$ Limits for Realistic Average Scram Insertion Time Single-Loop Operation (BOC to EOC)*

EOOS Condition	Core Power (% of Rated)	ATRIUM-10 MCPR$_p$†	ATRIUM 11 MCPR$_p$
Base case / 1 TCV/TSV closed‡	67.2	1.72	2.06
	60	1.80	2.06
	50	1.87	2.06
	40	1.93	2.06
	26	2.30	2.06
	26 at > 50%F below P_{bypass}	2.95	2.60
	23 at > 50%F below P_{bypass}	3.06	2.68
	26 at ≤ 50%F below P_{bypass}	2.95	2.49
	23 at ≤ 50%F below P_{bypass}	3.06	2.60
TBVOOS	67.2	1.72	2.06
	60	1.80	2.06
	50	1.88	2.06
	40	1.96	2.06
	26	2.33	2.11
	26 at > 50%F below P_{bypass}	3.26	3.10
	23 at > 50%F below P_{bypass}	3.41	3.34
	26 at ≤ 50%F below P_{bypass}	2.95	2.49
	23 at ≤ 50%F below P_{bypass}	3.12	2.70
RPTOOS	67.2	1.72	2.06
	60	1.80	2.06
	50	1.87	2.06
	40	1.93	2.06
	26	2.30	2.07
	26 at > 50%F below P_{bypass}	2.95	2.60
	23 at > 50%F below P_{bypass}	3.06	2.68
	26 at ≤ 50%F below P_{bypass}	2.95	2.49
	23 at ≤ 50%F below P_{bypass}	3.06	2.60
PROOS	67.2	1.80	2.06
	60	1.93	2.06
	50	2.02	2.06
	40	2.21	2.06
	26	2.50	2.28
	26 at > 50%F below P_{bypass}	2.95	2.60
	23 at > 50%F below P_{bypass}	3.06	2.68
	26 at ≤ 50%F below P_{bypass}	2.95	2.49
	23 at ≤ 50%F below P_{bypass}	3.06	2.60

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[*] These limits may need to be adjusted to account for the CRWE results provided in Table 5.14 - Table 5.17.

[†] The ATRIUM-10 limits may need to be adjusted to account for pump seizure results.

[‡] Operation with one TCV/TSV closed is supported for power levels ≤ 75% of rated.

Table 8.4 MCPR$_p$ Limits for Maximum Allowable Average Scram Insertion Time Single-Loop Operation (BOC to EOC)*

EOOS Condition	Core Power (% of Rated)	ATRIUM-10 MCPR$_p$[†]	ATRIUM 11 MCPR$_p$
Base case / 1 TCV/TSV closed[‡]	67.2	1.81	2.06
	60	1.90	2.06
	50	2.13	2.06
	40	2.17	2.06
	26	2.52	2.15
	26 at > 50%F below P_{bypass}	2.95	2.60
	23 at > 50%F below P_{bypass}	3.06	2.68
	26 at ≤ 50%F below P_{bypass}	2.95	2.49
	23 at ≤ 50%F below P_{bypass}	3.06	2.60
TBVOOS	67.2	1.94	2.06
	60	2.03	2.06
	50	2.25	2.06
	40	2.30	2.06
	26	2.61	2.29
	26 at > 50%F below P_{bypass}	3.35	3.20
	23 at > 50%F below P_{bypass}	3.50	3.44
	26 at ≤ 50%F below P_{bypass}	3.04	2.59
	23 at ≤ 50%F below P_{bypass}	3.21	2.80
RPTOOS	67.2	2.00	2.14
	60	2.08	2.14
	50	2.25	2.15
	40	2.32	2.15
	26	2.64	2.38
	26 at > 50%F below P_{bypass}	3.07	2.80
	23 at > 50%F below P_{bypass}	3.18	2.88
	26 at ≤ 50%F below P_{bypass}	3.07	2.69
	23 at ≤ 50%F below P_{bypass}	3.18	2.80
PROOS	67.2	1.86	2.06
	60	1.93	2.06
	50	2.13	2.06
	40	2.23	2.06
	26	2.52	2.28
	26 at > 50%F below P_{bypass}	2.95	2.60
	23 at > 50%F below P_{bypass}	3.06	2.68
	26 at ≤ 50%F below P_{bypass}	2.95	2.49
	23 at ≤ 50%F below P_{bypass}	3.06	2.60

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[*] These limits may need to be adjusted to account for the CRWE results provided in Table 5.14 - Table 5.17.

[†] The ATRIUM-10 limits may need to be adjusted to account for pump seizure results.

[‡] Operation with one TCV/TSV closed is supported for power levels ≤ 75% of rated.

Table 8.5 Flow-Dependent MCPR Limits for Realistic and Maximum Allowable Average Scram Insertion Times TLO and SLO (BOC to EOC)

EOOS Condition	Core Flow (Mlbm/hr)*	ATRIUM-10 $MCPR_f$	ATRIUM 11 $MCPR_f$
Base case / RPTOOS / PROOS	108	1.25	1.25
	40	1.51	1.51
	35	2.24	2.08
	30	2.39	2.23
TBVOOS	108	1.41	1.41
	45	1.73	1.73
	35	2.24	2.08
	30	2.39	2.23
1 TCV/TSV Closed[†]	108	1.43	1.43
	45	1.70	1.70
	40	2.09	1.93
	30	2.39	2.23

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[*] SLO is supported at core flows ≤ 52 Mlbm/hr.
[†] Operation with one TCV/TSV closed is supported at power levels ≤ 75% of rated.

Table 8.6 ATRIUM-10 Steady-State LHGR Limits

Peak Pellet Exposure (MWd/kgU)	LHGR (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Table 8.7 ATRIUM 11 Steady-State LHGR Limits

Peak Pellet Exposure (MWd/kgU)	LHGR (kW/ft)
0.0	13.6
21.0	13.6
53.0	10.2
80.0	3.5

Table 8.8 ATRIUM-10 LHGRFAC$_p$ Multipliers for Realistic and Maximum Allowable Average Scram Insertion Times TLO and SLO (BOC to EOC)*

EOOS Condition	Core Power (% Rated)	LHGRFAC$_p$
Base case / PROOS / 1 TCV/TSV closed†	100	1.00
	80	0.90
	26	0.64
	26 at >50%F below P_{bypass}	0.45
	23 at >50%F below P_{bypass}	0.35
	26 at ≤50%F below P_{bypass}	0.49
	23 at ≤50%F below P_{bypass}	0.42
TBVOOS / RPTOOS	100	1.00
	80	0.83
	26	0.59
	26 at >50%F below P_{bypass}	0.40
	23 at >50%F below P_{bypass}	0.35
	26 at ≤50%F below P_{bypass}	0.49
	23 at ≤50%F below P_{bypass}	0.42

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* These multipliers may need to be adjusted to account for the CRWE results provided in Table 5.18.

† Operation with one TCV/TSV closed is supported at power levels ≤ 75% of rated.

Table 8.9 ATRIUM 11 $LHGRFAC_p$ Multipliers for Realistic and Maximum Allowable Average Scram Insertion Times TLO and SLO (BOC to EOC)*

EOOS Condition	Core Power (% Rated)	$LHGRFAC_p$
Base case / PROOS / 1 TCV/TSV closed†	100	1.00
	80	0.91
	26	0.65
	26 at >50%F below P_{bypass}	0.40
	23 at >50%F below P_{bypass}	0.37
	26 at ≤50%F below P_{bypass}	0.43
	23 at ≤50%F below P_{bypass}	0.37
TBVOOS / RPTOOS	100	0.99
	80	0.91
	26	0.65
	26 at >50%F below P_{bypass}	0.37
	23 at >50%F below P_{bypass}	0.33
	26 at ≤50%F below P_{bypass}	0.43
	23 at ≤50%F below P_{bypass}	0.37

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[*] These multipliers may need to be adjusted to account for the CRWE results provided in Table 5.18.

[†] Operation with one TCV/TSV closed is supported at power levels ≤ 75% of rated.

Table 8.10 LHGRFAC$_f$ Multipliers for Realistic and Maximum Allowable Average Scram Insertion Times TLO and SLO (BOC to EOC)

EOOS Condition	Core Flow (Mlbm/hr)*	ATRIUM-10 LHGRFAC$_f$	ATRIUM 11 LHGRFAC$_f$
Base case / RPTOOS / PROOS	108	1.00	1.00
	80	1.00	1.00
	40	0.64	0.64
	30	0.42	0.43
TBVOOS	108	1.00	1.00
	100	0.98	0.97
	40	0.55	0.56
	30	0.37	0.39
1 TCV/TSV Closed†	108	0.95	0.94
	40	0.52	0.52
	30	0.35	0.37

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* SLO is supported at core flows ≤ 52 Mlbm/hr.

† Operation with one TCV/TSV closed is supported at power levels ≤ 75% of rated.

Table 8.11 ATRIUM-10 MAPLHGR Limits

Average Planar Exposure (GWd/MTU)	MAPLHGR (kW/ft)
0.0	12.5
15.0	12.5
67.0	6.9

Table 8.12 ATRIUM 11 MAPLHGR Limits

Average Planar Exposure (GWd/MTU)	MAPLHGR (kW/ft)
0.0	12.0
20.0	12.0
60.0	9.0
69.0	7.2

Figure 8.1 [

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Figure 8.2 [

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Figure 8.3 [

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Framatome Affidavit for ANP-3884P, Revision 1, "Susquehanna Unit 2 Cycle 21 Reload Safety Analysis"

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-3884P, Revision 1 "Susquehanna Unit 2 Cycle 21 Reload Safety Analysis," dated February 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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(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.

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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: February 25, 2021

Alan B Meg

Alan B. Meginnis