NUSCALE POWER™

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Changes to Part 2 Tier 2 Final Safety Analysis Report, Chapters 1, 4 and 15 and Part 4, Technical Specifications

REFERENCES: 1. Letter from NuScale Power LLC, to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application," dated December 31, 2016 (ML17013A229)

2. Letter from NuScale Power LLC, to Nuclear Regulatory Commission, "NuScale Power Submittal of Licensing Topical Report, NuScale Power Critical Heat Flux Correlations TR-0116-21012, Revision 1, dated November 2017 (ML17335A089)

In Reference 2, NuScale Power, LLC (NuScale) submitted Revision 1 to Licensing Topical Report, NuScale Power Critical Heat Flux Correlations TR-0116-21012, which adopts a new critical heat flux (CHF) correlation, NSP4. This letter provides conforming updates to Final Safety Analysis Report (FSAR) Chapters 1, 4 and 15 as well as Part 4, Technical Specifications submitted in Reference 1. The Enclosure to this letter provides a mark-up of the affected FSAR and Technical Specification pages in redline/strikeout format. NuScale will include this change as part of a future revision to the NuScale Design Certification Application.

In addition to the conforming changes to implement the revised topical report, the FSAR Chapter 15 changes also include a revised sequence of events table to resolve an internally identified corrective action and additional Limiting Analysis Results tables in Sections 15.1 and 15.4 that required renumbering the remaining tables. The renumbered tables are included with the attached changes, but do not show revision lines unless other changes were made to table contents. Note that some of these attached change pages include redline/strikeout changes associated with responses to previous responses to Requests for Additional Information (RAIs). These RAI responses are not affected by this change except for several Critical Heat Flux Ratio (CHFR) figures included in the responses to eRAI 8764 and eRAI 8765 (ML17144A450 and ML17144A452). The CHFR figures in the enclosure to this letter supersede those figures.

Note that the Technical Specification Safety Limits affected by the implementation of the NSP4 correlation were also modified to relocate the critical heat flux correlation values from the Safety Limit to the Core Operating Limits Report (COLR). The requirement for and contents of the COLR are described in Technical Specification 5.6.3. This relocation is consistent with similar approved Technical Specification changes implemented at the Farley nuclear plant (ML013400451).

This letter makes no regulatory commitments or revisions to any existing regulatory commitments.

Please feel free to contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com if you have any questions.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8G9A
Gregory Cranston, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A
Marieliz Vera, NRC,OWFN-8G9A

Enclosure: Changes to NuScale Part 2 Tier 2 Final Safety Analysis Report Chapters 1, 4 and 15 and Part 4, Technical Specifications

Enclosure:

Changes to NuScale Part 2 Tier 2 Final Safety Analysis Report Chapters 1, 4 and 15 and Part 4, Technical Specifications

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

technology. This test program was inspected by the NRC in accordance with Inspection Procedures IP 35034, 35017, and 36100.

1.5.1.2 Critical Heat Flux Testing - NuFuel HTP2 Fuel Design

The primary objective for this test program was to obtain CHF data for the NuScale fuel design that employs AREVA HMP/HTP spacer grid technology (designated as NuFuel HTP2) to augment the existing database that was previously obtained for NuScale's preliminary fuel design (described in Section 1.5.1.1). The new data was used to validate NuScale's NSP-2 CHF correlation developed using the preliminary fuel design tests for the NPM application. In addition, this test allowed NuScale to obtain bundle subchannel exit temperatures to determine mixing coefficients, and to collect single-phase and two-phase pressure drop characteristics of the assembly for a range of bundle powers and hydraulic conditions.

The CHF test employed an electrically-heated test section that consisted of a 5x5 simulated fuel bundle built to prototypic geometry and employing AREVA HTP/HMP grid technology. Like the preliminary fuel tests, fuel assembly simulators with uniform and cosine-axial power shapes were tested using a 5x5 fuel bundle with or without the center fuel rod replaced by a guide tube. The testing was conducted by flowing water through the test section at specified flow rates over a range of hydraulic conditions of the NPM. At each test point, the loop was configured for a specified flow rate, inlet temperature, and exit pressure conditions, and the bundle power was increased until CHF was detected over a range of operating conditions and axial power shapes for vertical 5X5 fuel assembly configurations. The occurrence of CHF was indicated by an excursion of the fuel simulator surface temperatures.

The prototypic fuel design tests were conducted at the AREVA Karlstein Thermal Hydraulics (KATHY) facility located in Karlstein, Germany. The test data was used to validate the applicability of NuScale's NSP-2 CHF correlation and to develop the NSP4 correlation for the NuFuel HTP2 fuel design.

1.5.1.3 Steam Generator Tube Inspection Evaluation and Demonstration

The NuScale helical coil SG is designed to provide access for required inservice inspection and repair. Access to the tube sheet in the NuScale design requires traversal of the steam access flanges through the containment vessel and the SG steam and feed plena.

Due to the unique inspection configuration of the helical SG, a design evaluation and proof-of-concept demonstration by an SG non-destructive examination inspection supplier was conducted to inform the SG design and provide confidence in the ability to inspect the helical SG. Information and experience gained in the proof-of-concept evaluation and demonstration test helped to inform the design development of the SG and associated plena.

Tests were conducted at the Oregon industrial facilities in Albany, OR and included representative, full-scale helical coils in a vertical orientation. Two coils were used that corresponded to a curvature similar to the innermost and outermost column geometries to envelop the largest and smallest transition radii within the SG. Bend radii

RAI 07.0.DSRS-1, RAI 07.0.DSRS-2, RAI 07.0.DSRS-3, RAI 07.0.DSRS-4, RAI 07.0.DSRS-5, RAI 07.0.DSRS-6

Table 1.6-1: NuScale Referenced Topical Reports

Topical Report Number	Topical Report Title	Submittal Date	FSAR Section
NP-TR-1010-859-NP-A, Rev 3	NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant	~~May 2015~~December 2016	17
TR-0515-13952-A, Rev 0	Risk Significance Determination	July 2015	17, 19
TR-0815-16497, Rev 0	Safety Classification of Passive Nuclear Power Plant Electrical Systems	October 2015	8
TR-1015-18653-A, Rev ~~1~~2	Design of the Highly Integrated Protection System Platform Topical Report	~~November 2016~~September 2017	7, 15
TR-0915-17565, Rev ~~1~~2	Accident Source Term Methodology	~~April 2016~~September 2017	15
TR-0116-20825, Rev ~~0~~1	Applicability of AREVA Fuel Methodology for the NuScale Design	July 2016	4
TR-0616-48793, Rev 0	Nuclear Analysis Codes and Methods Qualification	August 2016	4
TR-0516-49417, Rev 0	Evaluation Methodology for Stability Analysis of the NuScale Power Module	July 2016	4
TR-0516-49422, Rev 0	LOCA Evaluation Model	December 2016	15
TR-0915-17564, Rev ~~0~~1	Subchannel Analysis Methodology	~~October 2016~~February 2017	4
TR-0516-49416, Rev ~~0~~1	Non-LOCA Methodologies	August 2017	15
TR-0116-21012, Rev ~~0~~1	NuScale Power Critical Heat Flux Correlations~~-NSP2~~	~~October 2016~~November 2017	4
TR-0716-50350, Rev 0	Rod Ejection Analysis Methodology	December 2016	15
TR-0716-50351, Rev 0	NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces	~~December 2015~~September 2016	4
TR-0915-17772, Rev 0	Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Site	December 2016	15

4.4 Thermal and Hydraulic Design

The thermal-hydraulic design of the NuScale Power Module (NPM) provides cooling for fuel and core components and protects the fuel and cladding during off normal conditions. Adherence to a set of specified acceptable fuel design limits (SAFDLs) preserves the integrity of the fuel and cladding and prevents release of fission products from the fuel.

The NPM is a natural circulation pressurized water reactor (PWR) with integral, once-through, helical coil steam generators (SGs). The driving force for natural circulation flow is the pressure head caused by the lower density water in the core and the higher density water in the downcomer. This pressure head varies with power and as a result there is a unique steady-state flow at each power level.

The methodology and analysis tools (i.e., licensing methodology) used in the thermal-hydraulic design are summarized in this section and described in detail in the referenced NuScale topical reports. The VIPRE-01 code is used for steady-state and transient subchannel fuel and temperature calculations. The PIM code is used for thermal-hydraulic stability calculations. The subchannel steady state results provided in this section and the transient results in Chapter 15 are performed in accordance with the methodology, including all restrictions, defined in Reference 4.4-3. The methodology is used to establish the power peaking limits and protect SAFDLs without using thermal margin-specific trips. Fuel rod thermal evaluations are performed at rated power and during transients up to the design limit burnup to verify the fuel temperature and integrity design bases described in Section 4.2.1 are satisfied. These analyses also provide input for the initial fuel rod thermal conditions used in Chapter 15 transient analyses.

Hydraulic flow instabilities are precluded by a regional exclusion solution. Detection and suppression of hydraulic instabilities is not required.

4.4.1 Design Basis

The design bases for the thermal-hydraulic design of the reactor are discussed below. The design bases for the mechanical design of the fuel are discussed in Section 4.2.1. The instrumentation and controls system design features that address the monitoring requirements in General Design Criterion (GDC) 13 and the protection system requirements in GDC 20 are described in Section 7.1.

Consistent with GDC 10, the thermal-hydraulic design of the reactor core includes sufficient margin to critical heat flux (CHF) to ensure adequate heat transfer with a 95-percent probability at the 95-percent (95/95) confidence level so that SAFDLS are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs) and conditions that result in unstable power oscillations with the reactor trip system available.

Consistent with GDC 12, the thermal-hydraulic design of the core includes design and operational limits that preclude power instability such that fuel design limits are not exceeded.

The following are the design bases for the NuScale thermal-hydraulic design:

4.4.1.1 Critical Heat Flux

Adequate heat transfer from the fuel cladding to the reactor coolant is provided by assuring that critical heat flux limits are met during normal operation, AOOs, and infrequent events (IE). ~~A~~ NuScale specific CHF correlations ~~is~~are used to ensure that CHF does not occur with a 95 percent probability at a 95 percent confidence level during normal operation and abnormal operating occurrences. For some accidents, some rods may be predicted to exceed CHF criteria as long as the requirements of 10 CFR 100 are met.

4.4.1.2 Fuel Temperature

For normal operation and AOOs, the fuel melting temperature is not exceeded in any part of the core. These analyses are performed at rated power and during transients up to the design limit burnup as described in Section 4.2.

4.4.1.3 Core Flow

The core flow design basis is that 91.5 percent of the minimum design flow passes through the core and provides fuel cooling. This is based on a bypass flow of 8.5 percent which accounts for flow through the fuel assembly guide tubes, the reflector block, and the gap between the reflector block and the core barrel.

4.4.1.4 Hydrodynamic Stability

The hydrodynamic stability design basis is that normal operation and AOO events do not lead to hydrodynamic instability as discussed in Section 4.4.7.

4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

The NPM uses natural circulation to drive the flow in the reactor coolant system (RCS) to provide core heat removal during normal plant operation, AOOs, infrequent events (IEs), and accidents. A description of the thermal-hydraulic characteristics of the core is provided in the following sections.

4.4.2.1 Summary Comparison

Figure 5.1-3 shows the significant hydraulic features of the natural circulation flow path. Table 4.4-1 provides geometrical information on the key components of the RPV flow path.

Table 4.4-2 and the Subchannel Analysis Methodology topical report (Reference 4.4-3) provide the relevant parameters for the thermal-hydraulic evaluation of core performance of the NPM. Table 4.4-2 also provides similar information for other recent PWR designs for comparison purposes.

Table 4.4-3 summarizes the applicable ranges for some of the existing CHF/departure from nucleate boiling (DNB) correlations in the public domain. The Babcock and Wilcox B&W-2 and Westinghouse W-3 CHF correlations were developed for their specific fuel assembly applications. The Electric Power Research Institute EPRI-1 critical heat flux correlation and the Atomic Energy of Canada Limited (AECL) critical heat flux look-up table were developed for general applications. The NuScale natural circulation flow conditions include conditions that are outside of the CHF test data provided by the B&W, EPRI, and Westinghouse correlations. Therefore, several test programs were conducted to obtain CHF test data for the NuScale reactor fuel design. These programs culminated in the development of the NuScale NSP2 and NSP4 CHF correlations which ~~is~~are described in Section 4.4.2.7.

4.4.2.2 Critical Heat Flux

The overall margin for protecting the fuel cladding SAFDLs is established by an analysis limit that accounts for testing uncertainties, manufacturing tolerances, and consideration of non-testing variations, such as rod bow, measurement uncertainties and instrumentation delays. Reference 4.4-3 provides a depiction of the critical heat flux ratio (CHFR) limits and thermal margins.

For subchannel analysis, the key fuel failure mechanism is clad overheating in off-nominal conditions, such as AOOs, infrequent events, and accidents. Fuel rod failure occurs when the heat transfer coefficient between the fuel rod clad and coolant degrades significantly to due to the formation of a continuous vapor layer on the fuel rod. The degradation of the heat transfer coefficient in a two-phase flow condition is dependent on local conditions such as pressure, flow rate, coolant quality, and boiling regime. Various terms have been used to describe this phenomenon, including CHF, departure from nucleate boiling, critical power ratio, boiling crisis, boiling transition, burnout, and dryout.

The low flow steady-state nominal conditions and hypothetical transient and accident conditions in the NPM suggest that both the "DNB" and "Dryout" CHF mechanisms are relevant. "CHF" is a more general term, including both "DNB" and "Dryout," which are specific CHF mechanisms. For internal consistency in modeling the range of NuScale-specific phenomena, NuScale thermal margin analyses use the generic term CHF.

The parameter of interest for preventing the occurrence of CHF is the ratio of the critical heat flux to local heat flux, or CHFR:

$$CHFR = q''_{CHF} / q''_{local} \qquad \text{Eq. 4.4-1}$$

where

q''_{CHF} = critical heat flux, 10^6 btu/hr-ft^2, and

q''_{local} = local heat flux, 10^6 btu/hr-ft^2.

The NuScale CHFR limit for the NSP2 correlation that corresponds to a 95 percent probability of CHF at a 95 percent confidence level is 1.17. The NuScale CHFR limit for the NSP4 correlation that corresponds to a 95 percent probability of CHF at a 95 percent confidence level is 1.21.

The range of applicability of the NSP2 CHF correlation is:

	Pressure, psia	300 to 2300
RAI 04.04-6, RAI 04.04-9	Local Mass Flux, $10^6 lb_m/hr\text{-}ft^2$	0.110 to 0.700
RAI 04.04-6, RAI 04.04-9	Local equilibrium quality, %	≤ 9~~5~~0.0
	Inlet equilibrium quality, %	≤ 0

The range of applicability for the NSP4 correlation is:

Pressure, psia	500 to 2300
Mass flux, 10^6 lbm/hr-ft^2	0.116 to 0.635
Local equilibrium quality, %	≤ 95
Inlet equilibrium quality, %	≤ 0

The transient response of the reactor system is dependent on the initial power distribution. Limits provided by the core system and the protection system ensure that the design meets CHF design bases for AOOs. The core operating limits report (COLR) specifies the cycle-specific, power-peaking limits that maintain the core power distribution within prescribed limits during power operation. These power-peaking limits are expressed as limits on total heat flux (F_Q), enthalpy rise ($F_{\Delta H}$), and axial peak (F_Z). These power peaking factors are functions of burnup and power level. Section 4.3.1 provides additional discussion about the development and use of these limits.

- Enthalpy rise hot channel factor ($F_{\Delta H}$) is the ratio of the power in the hot rod divided by the power in the average rod. This all rods out (ARO) limit ensures that the design basis value for the CHFR is met for normal operation, operational transients, and IEs.
- The heat flux hot channel factor (or total peaking factor), F_Q, is the ratio of maximum local heat flux on the surface of a fuel rod to the average fuel rod heat flux. The maximum F_Q value is used to calculate the peak linear heat generation rate. The limit on F_Q is established to ensure none of the fuel design criteria are exceeded and the assumptions made in the accident analysis remain valid.
- Axial Peaking Factor (F_Z), is defined as the maximum relative power at any axial point in a fuel rod divided by the average power of the fuel rod. F_Z can be defined for a rod, an assembly, or the entire core.

The Module Protection System (MPS) automatically initiates and controls the protective actions necessary to mitigate the effects of the design basis events (DBEs) identified in Table 7.1-1. The MPS reactor trip functions are listed in Table 7.1-3, including the associated parameter and analytical limits.

The core design and thermal limits are developed such that the thermal margin criteria are not exceeded for normal operation and AOOs. Specifically, there is a 95-percent probability at the 95-percent confidence level (95/95) that the hot rod in the core does not experience a CHF condition. For the purpose of this analysis, the CHF is assumed to occur if the subchannel analysis-calculated CHFR is less than the allowable limit. For IEs and accidents, the total number of fuel rods that exceed the criteria are assumed to fail and are used in determining the radiological dose source term.

4.4.2.3 Linear Heat Generation Rate

Limits on axial peaking factor are not required because the limits on F_Q and $F_{\Delta H}$ maintain a sufficiently flat power distribution, and axial peaking is treated in a multi-layered approach involving both operational restrictions and analysis.

A limit on peak linear heat generation rate (PLHGR) is specified to help ensure that fuel performance limits are not exceeded. The design limit on PLHGR maintains the fuel temperature below the centerline melt criterion and limits the peak cladding temperature so cladding-coolant chemical interactions remain within the acceptable range.

The total heat flux peaking factor (F_Q) is used to calculate the PLHGR. Section 4.3.1 provides a discussion on the calculation of the PLHGR based on the linear heat generation (LHGR) and the design F_Q.

4.4.2.4 Subchannel Analytical Results

Figure 4.4-2 through Figure 4.4-8 provide maps of the typical distribution of thermal hydraulic parameters throughout the NuScale core. These steady state analyses are performed using VIPRE-01 and are based on a middle of the equilibrium cycle power distribution at 100 percent power. The equilibrium cycle is the reference cycle described in Section 4.3. Each rod and subchannel in the one-eighth core is modeled in this analysis. Figure 4.4-2 provides the enthalpy rise hot channel factor for each individual rod ($F_{\Delta H}$) in the one-eighth core. Figure 4.4-3 shows the MCHFR for each subchannel (defined as the area formed by four fuel rods or three fuel rods and a guide tube). In this analysis, the MCHFR for the entire core (using the NSP2 correlation) is 3.8 (compared to the 95/95 limit of 1.17). Figure 4.4-4 shows the maximum clad outer wall temperature for each rod in the core. Figure 4.4-5 shows the maximum rod heat flux, Figure 4.4-6 the average channel mass flux, and Figure 4.4-7 the exit equilibrium quality for each subchannel. Figure 4.4-8 provides the coolant temperature at the exit of each subchannel. These values are best estimate values in that they do not include the uncertainties that are discussed later in this section and that are applied in a subchannel analysis. A figure for void fraction is not provided because the channel exit void fractions are zero except for the hot channel which is 0.04. Table 4.4-7 shows the exit void fractions for the core average and hot channel for the equilibrium power distribution in Figure 4.4-2. In addition, Table 4.4-7 shows the same void fraction values using the conservative 24-channel subchannel model with all uncertainties applied. The 24-channel model is described in Section 4.4.4.5.2 and in Reference 4.4-3.

It is important to maintain subcooled margin in the riser (area above the control rod guide tubes) during normal operation to ensure that margin-to-thermal-hydraulic stability is maintained as discussed later in Section 4.4.7. A reactor trip actuates 5 degrees F before the core outlet average temperature reaches saturation (Section 7.1).

4.4.2.5 Core Coolant Flow Distribution

The NuScale design uses natural circulation, and there is no active control of the core flow. The core inlet flow distribution is dependent upon the geometry of the RCS loop, including the lower core plate and bypass flow paths. The core bypass flow paths are discussed in Section 4.4.3.1. There are flow inlets for each of the fuel assemblies in the core, similar to currently licensed PWR fuel designs. However, the design of the NPM is unique because the flow distribution is dependent upon the buoyancy-driven flow rate and vessel design. The core inlet flow distribution changes based on power level, axial and radial power distribution, and core average temperature. The inlet flow distribution is determined by computational fluid dynamics. The analysis indicates that at full power the peripheral assemblies receive from 3.5 to 4.5 percent less than average flow, the central assembly receives up to 3.5 percent less than average, and the assemblies located between the central assembly and the peripheral assemblies receive up to 3.6 percent more flow than the average assembly.

Several inlet flow distributions are evaluated in Reference 4.4-3 to understand the effect on CHF. For up to a 15 percent inlet flow reduction to the hot fuel assembly, the flows equalize after the flow reaches approximately one-third of the active fuel length, resulting in an insignificant decrease in MCFHR. Additionally, for a given radial power distribution, there was no sensitivity observed to the inlet flow distribution. A 5 percent reduction in the flow to the hot assembly is used in the subchannel analysis.

4.4.2.5.1 Core Coolant Temperature Distribution

As discussed in Reference 4.4-3, the core inlet temperature distribution is a boundary condition input for steady-state and transient subchannel analysis that is dependent on nuclear steam supply system design geometry. In the helical coil SG design, the primary RCS flow is on the shell side and the secondary feedwater flow is through the tubes. The concentric geometry of the SGs relative to the core removes any asymmetric helical coil SG influences on the coolant through the downcomer into the core inlet. A computational fluid dynamics calculation for the RCS loop determines the core inlet temperature distribution for several power levels and power distributions in the NPM design. The largest deviation in core inlet temperature to a fuel assembly is less than 0.25 degrees F from the average inlet temperature. A uniform core inlet temperature that is 5 degrees F higher than design is assumed in the subchannel analysis for AOOs, infrequent events, and accidents.

4.4.2.5.2 Turbulent Mixing

The turbulent mixing model within VIPRE-01 accounts for the exchange of enthalpy and momentum between adjacent subchannels caused by turbulent flow. The coefficient for turbulent mixing and the turbulent momentum factor are the two inputs needed for this model. This mixing model is incorporated into the VIPRE-01 energy and momentum equations, which is dependent on the amount of turbulent crossflow per unit length.

The NuScale turbulent mixing coefficient ~~was~~is determined from thermal mixing tests~~. This~~and the value is fuel-design specific~~; however, the mixing coefficient used~~

~~for the NPM is a conservatively low value representative of non-mixing vane grids~~. This value is further justified based on parametric sensitivity analysis described in Reference 4.4-3.

The value for the turbulent momentum parameter is not measured and is justified based on parametric sensitivity analysis provided in Reference 4.4-3. The sensitivity study results demonstrate that the NPM base model is not sensitive to the turbulent momentum parameter.

4.4.2.6 Core Pressure Drops and Hydraulic Loads

4.4.2.6.1 Hydraulic Loads

The NuScale fuel assemblies do not experience liftoff from the lower core plate under normal operating conditions and AOOs as described in Section 4.2. A liftoff analysis is performed using the hydraulic flow loads from zero percent to 102 percent power. The analysis considers the weight of a fuel assembly, the displaced fuel assembly volume, the fuel assembly springs, the maximum design flow rate, and the core average temperature. The calculated maximum hydraulic lift force is a small fraction of the assembly weight.

4.4.2.6.2 Core Pressure Drop

Flow testing on a full-scale prototype fuel assembly was performed to establish flow component loss coefficients and other related flow characterization parameters for the NuScale fuel assembly. The form loss coefficients are used in the fuel assembly liftoff analysis and the subchannel analysis. Fuel assembly pressure drop tests were performed for a range of steady-state conditions as part of the CHF testing program. The pressure drop across the core at full power conditions is provided in Reference 4.4-3.

4.4.2.7 Correlations and Physical Data

As discussed in the Critical Heat Flux topical report (Reference 4.4-1), CHF tests were performed at Stern Laboratories and at the AREVA KATHY test facility. These tests obtained steady-state CHF data used in the creation and validation of the NSP2 and NSP4 critical heat flux correlations. The Stern tests were performed on an assembly comparable to the NuScale fuel design, but with simple non-mixing spacer grids rather than the HMP™ and HTP™ spacer grids. As described in Section 4.2, the NuScale fuel assembly contains four intermediate spacer grids (HTP™) which induce a swirling flow pattern in the coolant and a single HMP™ non-mixing grid at the bottom of the assembly. The three Stern tests provide data over a range that encompasses the NPM operating parameter values and are used to develop a base CHF correlation. A set of CHF tests from the AREVA KATHY facility tested an assembly design that included the HMP™ spacer grids. The test was conducted with two different axial power profiles, a uniform axial and cosine axial power profile. The tests included both unit cell (four fuel rods) and cells containing guide tubes (three fuel rods and a guide tube). The data provide the basis for the NSP2 and NSP4 correlations that conservatively predict~~s~~ NuFuel HTP2™ critical heat flux performance.

Comparisons between Stern Laboratories and KATHY data demonstrate that the NuScale fuel assembly design performs as well or better than an assembly with a simple non-mixing spacer grid. The trend is expected because the HTP™ spacer grid has a mixing feature that becomes more effective as flow increases.

The limiting non-loss-of-coolant accident (LOCA) analyses in Chapter 15 are performed using NRELAP5 code. Once the limiting cases for each transient are identified, the determination of the thermal margin is performed using the VIPRE-01 subchannel core model with the ~~NSP2~~ NSP4 critical heat flux correlation. This subchannel model is described in detail in Reference 4.4-3. It is used with a conservative radial and axial power distribution and with all uncertainties applied deterministically as described in the reference.

4.4.2.8 Thermal Effects of Operational Transients

The subchannel analysis approach described below demonstrates that thermal-margin specific trips are not necessary to mitigate AOOs. The CHF analyses demonstrate that safety limits are met with the minimal operational constraints described in Section 4.4.3.2.

This section also demonstrates that hydraulic flow instabilities are precluded by reactor trip signals that occur prior to the development of any flow instabilities so that detection and suppression of hydraulic instabilities is not required.

4.4.2.9 Uncertainties in Estimates

Uncertainties or biases are incorporated into the subchannel methodology to provide conservatism. These uncertainties establish the design limit for the CHF correlation as shown in Figure 4.4-1. The derivation of the penalties or conservative bias are discussed in Reference 4.4-3. Uncertainties in the CHF correlation, analytical methods, operating conditions, physical inputs, core inlet flow distribution, and core exit pressure are considered in the subchannel analysis.

4.4.2.9.1 Correlation Uncertainties

There are uncertainties that are accounted for in subchannel safety analysis calculations, including those from the analysis method, physical manufacturing design inputs to the model, and operating conditions. The application of uncertainties in the NuScale subchannel methodology is deterministic, which means that the uncertainty associated with a parameter is applied in the conservative direction without considering the statistical combination of the uncertainties.

4.4.2.9.1.1 Analysis Method Uncertainties

The analysis method uncertainties include the computer code uncertainty and CHF correlation uncertainty. The computer code uncertainty comes from axial and radial modeling and the approximations in the governing constitutive equations in the VIPRE-01 code. The adequacy of the axial and radial models were confirmed with sensitivity studies.

Code comparisons to data in applicable ranges are used to reduce code uncertainty. Most of this test validation is by benchmarking to COBRA-FLX (Reference 4.4-5), an approved subchannel analysis code with an approved Safety Evaluation Report (Reference 4.4-6). The benchmark results for VIPRE-01 compare well for conditions anticipated for NuScale Power Plant applications and establish that no penalty is needed for computer code calculation bias.

The CHF correlation uncertainty is included in the 95/95 minimum critical heat flux ratio (MCHFR) safety limit of the NuScale-specific NSP2 and NSP4 CHF correlations. The CHF correlations are developed from the local conditions derived from a simulated subchannel model of the CHF test, using the subchannel software, in this case VIPRE-01. Therefore, the uncertainty in the computer code is included in the CHF correlation itself.

The CHF correlation development inherently accounts for VIPRE-01 code uncertainty and the 95/95 CHF design limit accounts for this uncertainty. For this reason, no additional penalties for uncertainty in analysis method are added to the subchannel calculations.

4.4.2.9.1.2 Uncertainty in Operating Conditions

The operating boundary conditions are input into the subchannel analysis to account for measurement uncertainty. The values for these uncertainties are based on the instrumentation used for monitoring and are plant specific. The measurement uncertainties consist of those related to core power, system flow, core inlet temperature, and core exit pressure. The operating uncertainties are comparable to those used in the industry and are discussed in Reference 4.4-3.

The core bypass flow is important because bypass flow is not available for heat transfer from the cladding. The core inlet flow boundary condition accounts for the appropriate bypass flow. The bypass values used for safety analysis are determined as analytical maximum values rather than best-estimate values.

4.4.2.9.2 Uncertainties in Physical Data Inputs

The following uncertainties in physical data used in the VIPRE-01 subchannel analysis are accounted for in the VIPRE-01 subchannel analysis model:

- enthalpy rise engineering uncertainty ($F^E_{\Delta H}$)
- heat flux engineering uncertainty (F^E_Q)
- LHGR engineering uncertainty (F^E_{LHGR})
- radial power distribution uncertainty
- fuel rod bow and assembly bow uncertainty
- core inlet flow distribution uncertainty
- core exit pressure distribution uncertainty

remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. The subchannel core thermal-hydraulic analysis determines that the MCHFR is maintained above the 95/95 limit during normal operation and AOOs, ensuring the SAFDLs are satisfied and fuel cladding integrity is demonstrated. The thermal margin criteria are not exceeded for normal operation and AOOs. For IEs and accidents, the total number of fuel rods that exceed the criteria and are assumed to fail is used as input for radiological dose calculation purposes.

4.4.4.1 Critical Heat Flux Correlation

The functional form of the NSP2 and NSP4 critical heat flux correlations ~~is~~are expressed as a curve fit to a number of physical parameters including:

- pressure
- cold wall factor
- boiling length
- local mass flux
- local equilibrium quality

The coefficients of the ~~NSP2~~ critical heat flux correlations are determined with a ~~five-fold~~ cross-validation process and linear least-squares regression based on local condition parameters calculated with the VIPRE-01 subchannel thermal-hydraulics code. The form of the equation and correlation coefficients and the details of the development of the correlations are provided in Reference 4.4-1.

4.4.4.2 Core Hydraulics

As discussed in Section 4.4.2.5.1, a uniform inlet temperature distribution is assumed that is 5 degrees F above the expected inlet temperature.

Table 4.4-5 lists the principal flow elements in the RPV flow path and describes the flow path.

4.4.4.3 Influence of Power Distribution

The subchannel analysis basemodel is developed to conservatively represent a cycle-specific core as described in Reference 4.4-3. The model preserves limiting core conditions along with the operational envelope specified in the cycle-specific COLR. The envelope accounts for power distribution throughout the core using design peaking factors in combination with the limiting RCS parameters such as flow and pressure. An eighth-core symmetric subchannel analysis model is used to capture the limiting conditions of the cycle-specific core.

The radial power distribution for the core is characterized by the enthalpy rise hot channel factor $F_{\Delta H}$, which is the ratio of the maximum integrated rod power within the core to the average rod power. $F_{\Delta H}$ is variable depending on the cycle design, the exposure, fuel composition, burnable poison loading, operational history, and thermal-

- The flow in the RCS loop is modeled as non-equilibrium, two-phase flow in which a drift flux formulation accounts for mechanical (velocity) differences between the liquid phase and the vapor phase (if any vapor exists).
- The pressurizer is not modeled. Pressure is specified by code input and the dependence of thermodynamic properties on pressure is uniform.
- A simplified model for ambient heat losses along the downcomer to the containment vessel provides representative estimates for this small effect on natural circulation driving head, which has some effect on stability at low-power conditions.
- The solid structures within the RPV, with the exception of the fuel rods in the core and the SG tubes, are assumed to have no heat exchange with the circulating fluid.
- The total core thermal power, flow rate, pressure, and inlet temperature are specified initial conditions for the RCS and SG secondary side. The specified conditions are based on plant performance operational predictions associated with plant design activities, or as chosen for sensitivity studies.

The geometry representation of the NPM pressure vessel for the numerical simulation is given in Figure 4.4-12. The core is represented by a heated section at the bottom of the riser. The cold leg annulus is represented as a one-dimensional pipe with a generally-varying cross section area. The helical coils of the SGs fill part of the cold leg volume and heat is exchanged between the downward flow in the RCS loop and the secondary side (inside the helical coil SG tubes). The dashed line represents a pressure boundary condition that is imposed by the pressurizer.

4.4.7.3 Stability Protection Solution

Section 4.4.3.3.1 describes how the NPM meets GDC 12 requirements by using an operating domain that is protected by MPS reactor trips in the exclusion region where the reactor is not allowed to operate. The exclusion region, defined by the area in the operating map where stability criteria are not met, is enforced automatically by the MPS trip setpoints.

The reactor operating maps for the NuScale reactor are described in Section 4.4.3.3.

In summary, a detection and suppression solution is not used for the NuScale design. Flow stability is ensured by maintaining a suitable operating region using an exclusion region solution.

4.4.8 References

4.4-1 NuScale Power LLC, “NuScale Power Critical Heat Flux Correlations,” TR-0116-21012, Revision 1, ~~September 2016~~November 2017.

4.4-2 AREVA Incorporated, "Computational Procedure for Evaluating Fuel Rod Bowing," XN-75-32(P)(A), Supplement 1-4, February 1983.

4.4-3 NuScale Power LLC, "Subchannel Analysis Methodology," TR-0915-17564-P, ~~September 2016~~February 2017.

4.4-4 NuScale Power LLC, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417-P, July 2016.

4.4-5 AREVA NP Inc., "COBRA-FLX: A Core Thermal-Hydraulic Analysis Code Topical Report," ANP-10311P, Revision 0, March 2010.

4.4-6 Bahadur, Sher, U.S. Nuclear Regulatory Commission, letter to Pedro Salas, AREVA NP, Inc., January 29, 2013, Agencywide Document Access and Management System (ADAMS) Accession No. ML13135A053.

4.4-7 NuScale Power, LLC, "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825, Revision 1, dated June 2016.

Figure 4.4-1: Critical Heat Flux Ratio Limits and Thermal Margins

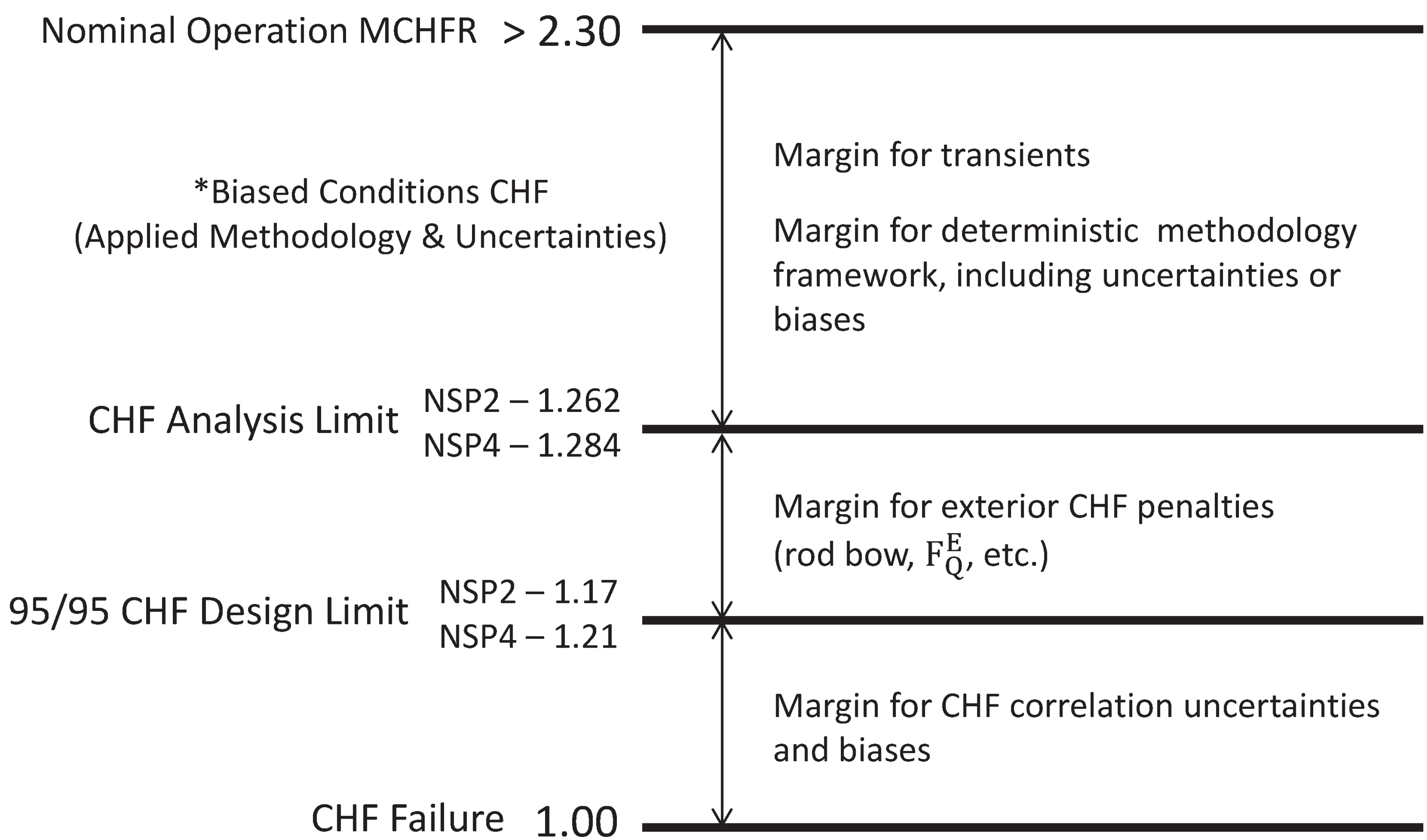


Figure 4.4-10: Minimum Critical Heat Flux Ratio versus Rated Core Power Axial Power Shapes

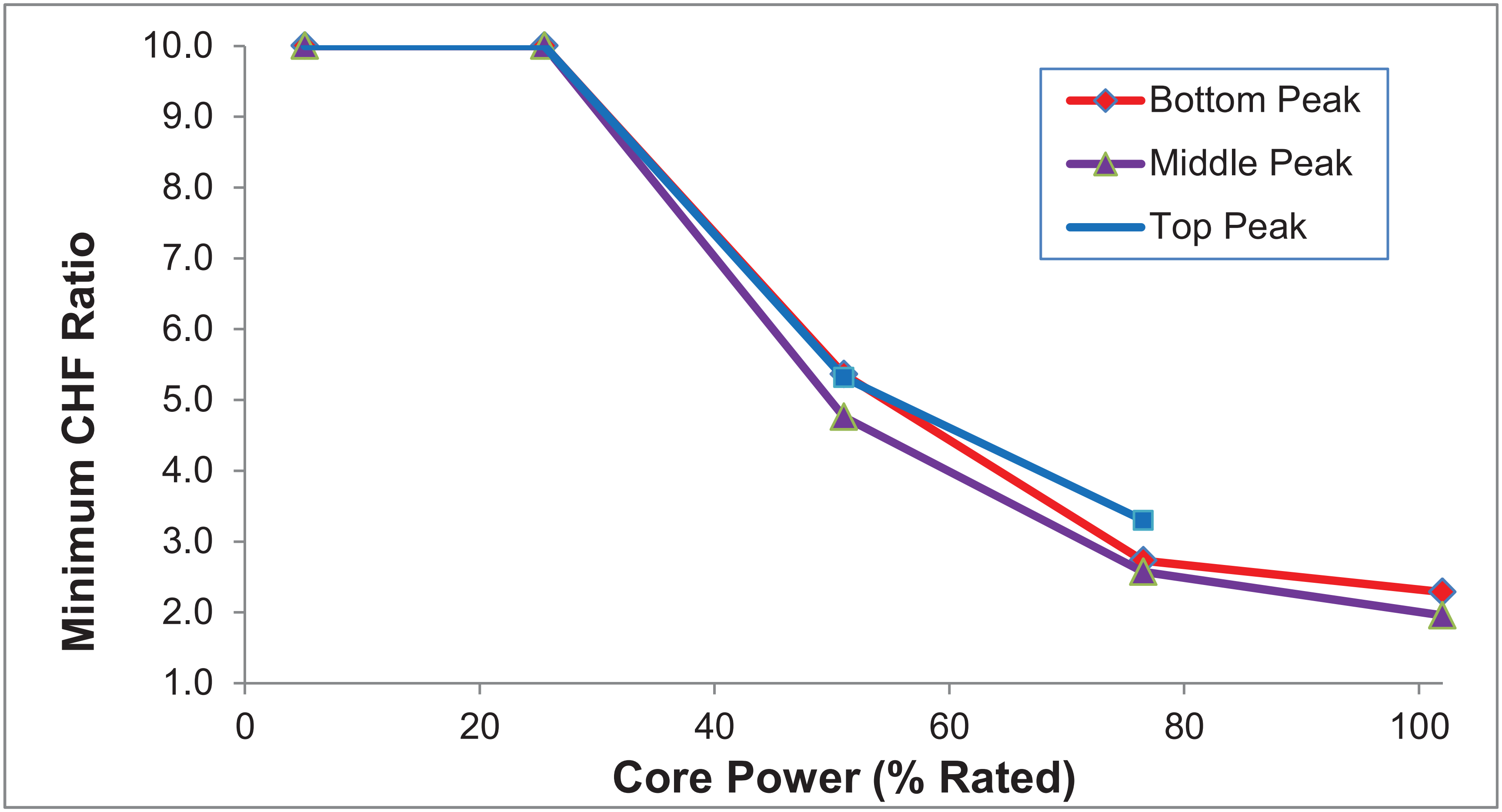
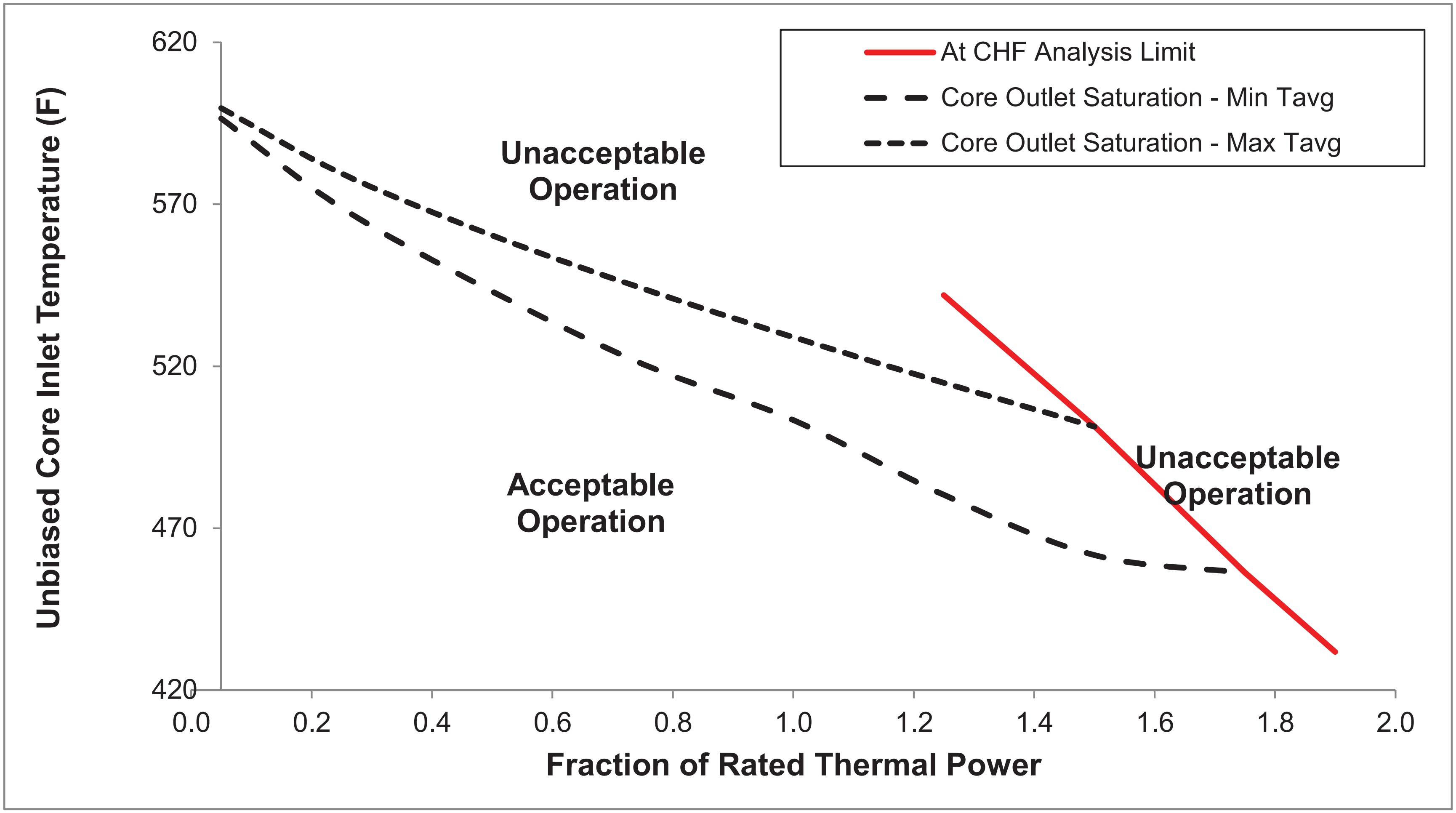


Figure 4.4-11: Thermal Margin Limit Map



- ~~The least negative Doppler coefficient is used because it results in the least strong negative reactivity feedback during the return to power, bounding the maximum peak power for the transient.~~
- ~~The DHRS heat transfer is increased by 30 percent to ensure the consequences of the cooldown are maximized after DHRS actuation.~~

~~Figure 15.0-8 shows the power response on a return to power. Minimum critical heat flux ratio is evaluated and the analysis confirms that design limits are not exceeded and the DHRS cooldown evaluation is non-bounding with respect to other events.~~

15.0.7 References

15.0-1 NuScale Power, LLC, Topical Report, "Subchannel Analysis Methodology," TR-0915-17564, Rev. 1~~0~~.

15.0-2 NuScale Power, LLC, Topical Report, "NuScale Power Critical Heat Flux Correlations ~~NSP2~~," TR-0116-21012, Rev. 1~~0~~.

15.0-3 NuScale Power, LLC, Topical Report, "LOCA Evaluation Model," TR-0516-49422, Rev. 0.

15.0-4 NuScale Power, LLC, Topical Report, "Accident Source Term Methodology," TR-0915-17565, Rev. 2~~1~~.

15.0-5 NuScale Power, LLC, Topical Report, "Non-LOCA Transient Analysis Methodology," TR-0516-49416, Rev. 1~~0~~.

15.0-6 Nuclear Energy Institute, Position Paper, "Small Modular Reactor Source Terms," December 27, 2012, Washington, DC.

15.0-7 NuScale Power, LLC, Technical Report, "Long-Term Cooling Methodology," TR-0916-51299, Rev. 0.

15.0-8 K.F. Eckerman et al., "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

15.0-9 K.F. Eckerman and J.C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

15.0-10 NuScale Power, LLC, Topical Report, "Evaluation Methodology for Stability Analysis of NuScale Power Module," TR-0516-49417, Rev.0.

15.0-11 NuScale Power, LLC, Topical Report, "NuScale Rod Ejection Accident Methodology," TR-0716-50350, Rev. 0.

Table 15.0-10: Referenced Topical and Technical Reports (Continued)

Topical or Technical Report	Report	Rev No	Description	NRC SER Reference
Accident Source Term Methodology (Topical)	TR-0915-17565	2~~1~~	Describes assumptions, codes, and methodologies used to calculate the radiological consequences of design basis accidents. Describes the methodology for establishing the NuScale design basis source term (DBST) release timing and magnitude that meets 10 CFR 52.47(a)(2)(iv). Describes the DBST associated aerosol transport and iodine re-evolution assessment methodologies. Describes the STARNAUA aerosol modeling to the range of post-accident containment conditions and justifies the assumption that no elemental iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment. Describes the use of ARCON96 for establishing offsite atmospheric dispersion.	Not issued
Subchannel Analysis Methodology (Topical)	TR-0915-17564	1~~0~~	Discusses how NuScale Power, LLC, meets the NRC requirements for use of VIPRE-01 Safety Evaluation Reports (SERs), the modelling methodology for performing steady state and transient subchannel analyses, and the qualification of the code for application to the NuScale Power Plant design. Explains why the methodology is independent of any one CHF correlation and may be used for NuScale applications if methodology requirements are satisfied. Describes methodology for treatment of uncertainties in the NuScale subchannel methodology.	Not issued
NuScale Power Critical Heat Flux Correlations~~-NSP2~~ (Topical)	TR-0116-21012	1~~0~~	Provides the bases for use of ~~the NSP2~~ critical heat flux (CHF) correlations in VIPRE-01within its range of applicability, along with its associated correlation limit, for the NuScale Power, LLC, Design Certification Application and for the safety analysis of the NPM with NuFuel-HTP2™ fuel. The report describes the tests, test facilities, statistical methods, base CHF correlation development, NSPX factor development, and final validation for the development of the CHF correlation.	Not issued

Table 15.0-10: Referenced Topical and Technical Reports (Continued)

Topical or Technical Report	Report	Rev No	Description	NRC SER Reference
NuScale Rod Ejection Accident Methodology	TR-0716-50350	0	Describes the codes and methodology used to analyze the rod ejection accident (REA). Describes the three-dimensional behavior using SIMULATE5 and SIMULATE-3K, the reactor system response using NRELAP5, and the subchannel thermal-hydraulic behavior and fuel response using VIPRE-01.	Not Issued
Non-LOCA Transient Analysis Methodology	TR-0516-49416	1~~0~~	Describes evaluation model that simulates the NPM transient response to non-LOCA events. Addresses the EMDAP process used to establish the adequacy of the non-LOCA methodology. Uses a graded approach to the EMDAP for development of the non-LOCA system transient evaluation model considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions of the LOCA EM in the non-LOCA plant transient calculations. Describes the non-LOCA PIRT assessment of the relative importance of phenomena and processes that may occur in the NuScale Power Module during non-LOCA events in relation to specified figures of merit. Describes the requirements for evaluation model capability developed from the non-LOCAPIRT. Explains how NRELAP5 assessments performed for LOCA EM development demonstrate NRELAP5 qualification for high rank/low knowledge-level non-LOCA PIRT phenomena: 1. Describes the separate effects testing of the full-length DHRS at the NIST facility to address DHRS heat transfer. 2. Presents the NRELAP5 assessments against the SIET TF-1 and TF-2 data to validate adequacy of SG heat transfer from the DHRS loop during non-LOCA transients. 3. Describes the integral effects test of the DHRS operation at the NIST facility. 4. Provides a code-to-code benchmark assessing the NRELAP5 prediction of the NPM response to reactivity insertion events using RETRAN-3D.	Not issued

prior to the reactor trip. The high hot leg temperature trip also actuates the DHRS valves to open. The feedwater isolation valves (FWIVs) and MSIVs close, isolating the SG from the rest of the secondary system.

Steam generator pressure does not change significantly during the initial phase of the transient. However, after DHRS actuation at 133 seconds, the closure of the FWIVs and MSIVs causes pressurization of the SG. Steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the decrease in feedwater temperature event itself. The maximum secondary pressure is reached just after main steam and feedwater isolation.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting decrease in feedwater temperature does not violate the CHF limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. At approximately 2000 seconds, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to a decrease in feedwater temperature, and a return to a stable condition with no operator actions.

15.1.1.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.1.1.5 Conclusions

The five Design Specific Review Standard (DSRS) acceptance criteria for this AOO are met for the limiting decrease in feedwater temperature case. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The pressure responses in the reactor pressure vessel (RPV) and in the main steam system (MSS) are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for the decrease in feedwater temperature event. The maximum pressure values for the cases analyzed are shown in Table 15.1-3

2) Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event ~~is 1.438, which~~ is above the 95/95 limit as shown in Table 15.1-3. Therefore this acceptance criterion is met.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. Eventually, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to increase in feedwater flow, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

15.1.2.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.4.

15.1.2.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting increase in feedwater flow case. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

    - The pressure responses in the RPV and in the MSS are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for increase in steam flow event. The maximum pressure values for the cases analyzed are shown in Table 15.1-6.

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event ~~is 1.474, which~~ is above the 95/95 limit as shown in Table 15.1-6. Therefore, this acceptance criterion is met.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of General Design Criteria 10, 13, 15, 20, and 26.

    - The instrument spans and setpoints discussed in Section 15.1.2.3.2 address the guidance in RG 1.105.

FWIVs and MSIVs causes pressurization of the SG. Steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the increase in steam flow. The maximum secondary pressure is reached after main steam and feedwater isolation.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting increase in steam flow case does not violate the design limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. After 2000 seconds, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to an increase in steam flow, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

15.1.3.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.1.3.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting increase in steam flow case. These acceptance criteria, followed by how the NuScale design meets them are listed below.

1) Pressure in the RCS and MSS should be maintained below 110 percent of the design values.

    - The pressure responses in the RPV and in the MSS are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for increase in steam flow event. The maximum pressure values for the cases analyzed are shown in Table 15.1-9.

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event ~~is 1.442, which~~ is above the 95/95 limit as shown in Table 15.1-9. Therefore, this acceptance criterion is met.

3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

    - The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting increase in steam flow case does not violate the design limit.

As in the SLB case with limiting RPV pressure, the limiting single failure assumed in this SLB case is the failure of the primary MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation. This single failure results in a more severe plant condition by releasing more mass through the break. It also causes a higher heat load on the intact DHRS train. However, this failure does not affect the CHFR because it occurs after the MCHFR.

The result of this SLB case is a stable plant condition, where the DHRS maintains core cooling. For a discussion of a possible return to power scenario, see Section 15.0.6.

SLB Cases Resulting in Limiting Radiological Consequences

The sequence of events for a representative limiting SLB case for radiological consequences is provided in Table 15.1-12. Figure 15.1-45 and Figure 15.1-46 demonstrate the break flow rate and integral break flow for this case. This SLB scenario maximizes the time between the reactor trip and secondary isolation to maximize iodine spiking time. The break is small enough to avoid the low SG pressure trip, but eventually causes a high power trip based on the moderator temperature feedback due to the cooldown. The break continues to cool the RCS until the temperature drops sufficiently that the low superheat DHRS actuation signal isolates the break. The calculated break flow characteristics and reactor trip timing for this case, as well as an undetected small break case are used in the downstream radiological consequences analysis. Table 15.1-14 provides the integrated mass flows from these cases with additional margin that is used in the downstream radiological analysis presented in Section 15.0.3.

15.1.5.4 Radiological Consequences

The radiological consequences of the SLB event are discussed in Section 15.0.3. The results are summarized in Table 15.0-12.

15.1.5.5 Conclusions

The four DSRS acceptance criteria for this accident are met for the limiting SLB cases. These acceptance criteria, followed by how the NuScale design meets them are listed below.

1) Pressure in the RCS and MSS should be maintained below acceptable design limits, considering potential brittle, as well as ductile failures.

    - The limiting RPV pressure for a SLB ~~is shown to be 2156 psia, which~~ is under the more conservative AOO acceptance criterion of 110 percent of design values. Representative MSS pressures demonstrate margin to the acceptance criterion.

Therefore, the acceptance criteria for pressures is met for this event. The maximum primary and SG pressure values for the cases analyzed are shown in Table 15.1-15.

2) The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for pressurized water reactors based on acceptable correlations (see DSRS Section 4.4).

   - The MCHFR for this event ~~is 1.428, which~~ is above the 95/95 limit as shown in Table 15.1-15. Therefore, this acceptance criterion is met.

3) The radiological criteria used in the evaluation of steam system pipe break accidents (pressurized water reactors only) appear in DSRS Section 15.0.3.

   - The radiological analysis of the SLB accident is presented in Section 15.0.3 and demonstrates that the acceptance criteria are met.

4) System(s) provided for decay heat removal must be highly reliable and, when required, automatically initiated. For the NuScale Power Plant design, the DHRS provides the safety-related means of decay heat removal.

   - The results of the analysis show that the DHRS initiates and provides heat removal during a SLB, ensuring that acceptance criteria are not challenged.

15.1.6 Loss of Containment Vacuum/Containment Flooding

15.1.6.1 Identification of Causes and Accident Description

A loss of containment vacuum and containment flooding events that result in an increase in RCS cooling are NuScale Power Plant design-specific events. The NuScale containment net volume is less than conventional designs and the module is partially immersed in a pool of borated water during normal operation. Since the containment operates at a vacuum during normal operation, air or water ingress into containment could increase heat transfer from the RPV to the reactor pool. This overcooling could lead to a higher reactor power, higher RCS pressure, and reduced MCHFR.

The containment evacuation system (CES) maintains the containment volume at a vacuum during normal operation. A failure in the CES could result in loss of vacuum since containment pressure would increase due to evaporation of any RCS fluid leaking into containment. If the failure of the CES or RCS fluid leakage is sufficiently severe, this could result in a loss of vacuum event. If the containment vacuum is lost, heat transfer from the reactor vessel will increase. The analysis of a loss of containment vacuum shows a negligible effect on reactor power, and is therefore bounded by a containment flooding event.

The reactor component cooling water system (RCCWS) provides heat removal to the control rod drive system. The RCCWS piping passes through containment to provide this function. If this piping were to leak or rupture inside the CNV, a containment flooding event would occur. Other potential containment flooding sources include: feedwater line break, main steam line break, CVCS line break, high point vent pipe break, and RCCWS line break. The feedwater line break event is evaluated in Section spikes from boiling, the increase in pressure inside of containment is on the order of 1 psia, which does not challenge the analytical limit for containment pressure.

As the heat transfer increases from the RPV to the CNV, the RCS is overcooled. The slightly cooler core inlet temperature introduces an increase in reactivity due to the moderator temperature reactivity feedback, resulting in an increase in reactor power. The magnitude of the power increase is smaller than the other overcooling events presented in Section 15.1. The smaller power increase ensures that the pressures of the RCS and MSS do not challenge the design pressures of the RPV and main steam piping, respectively. As with other overcooling events, if a containment flooding event trips the reactor, the subsequent pressure rises in the primary and secondary systems are a result of the isolation functions and not a direct consequence of the containment flooding.

Most containment flooding scenarios are terminated by the protection system before any significant change in CHFR. However, the containment flooding case presented in this section does not trip the reactor, which results in a slightly degraded CHFR. The MCHFR calculated for this event ~~is 1.888, which~~ does not challenge the design limit as shown in Table 15.1-17 and is bounded by the other overcooling events presented inSection 15.1.

This containment flooding event that is limiting for MCHFR approaches a new steady state at a slightly higher power, demonstrating a stable condition.

15.1.6.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.1.6.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting containment flooding case. These acceptance criteria, followed by how the NuScale design meets them are listed below.

1) Pressure in the RCS and MSS should be maintained below 110 percent of the design values.

    - The pressure responses in the RPV and in the MSS are less severe than those of the AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met. The maximum primary and SG pressure values for the cases analyzed are shown in Table 15.1-17.

2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

    - The MCHFR for this event ~~is 1.888, which~~ is above the 95/95 limit as shown in Table 15.1-17, and bounded by the other overcooling events presented in Section 15.1. Therefore, this acceptance criterion is met.

Table 15.1-1: Sequence of Events Limiting MCHFR Case (15.1.1 Decrease in Feedwater Temperature)

Event	Time [s]
Feedwater temperature begins to decrease	0
Regulating bank begins to withdraw in response to a decrease in average RCS temperature	28
High hot leg temperature limit is reached	125
High reactor power limit is reached.	131
Peak reactor power/Limiting MCHFR	133
Peak RPV pressure	133
Reactor trip actuation	133
DHRS actuation	133
Feedwater and main steam isolation	140
Feedwater temperature reaches 100°F	160
Peak MSS pressure	181
Low low PZR level is reached (Containment and CVCS isolation begins after 3-second delay)	1382

Table 15.1-2: Decrease in Feedwater Temperature - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1174 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	+35psia
FW temperature	300 °F	+10 °F
MTC	EOC	Most Negative
DTC	BOC	Least Negative

Table 15.1-3: Decrease in Feedwater Temperature (15.1.1) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	1959 psia
Maximum SG Pressure	2310 psia	1432 psia
MCHFR	1.284	1.921

Table 15.1-4: Sequence of Events (15.1.2 Increase in Feedwater Flow)

Event	Time [s]
Feedwater flow begins to increase	0
Regulating bank begins to withdraw in response to a decrease in average RCS temperature	~4
Low steam superheat limit is reached	24
High reactor power limit is reached	25
Reactor trips on high core power signal	27
Peak reactor power	28
DHRS actuation	32
Peak MSS pressure	84

Table 15.1-5: Increase in Feedwater Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1173 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	+35psia
MTC	EOC	Most Negative
DTC	BOC	Least Negative

Table 15.1-6: Increase in Feedwater Flow (15.1.2) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	1936 psia
Maximum SG Pressure	2310 psia	1424 psia
MCHFR	1.284	1.944

Table 15.1-7: Sequence of Events (15.1.3 Increase in Steam Flow)

Event	Time [s]
Steam flow begins to increase	0
Peak reactor power is reached	55
High hot leg temperature limit is reached	60
Reactor trips on high hot leg temperature signal	60
DHRS actuation	68
Peak RPV pressure is reached	69
FWIVs and MSIVs fully close	75
Peak MSS pressure is reached	123
CVCS isolation on low pressurizer pressure	1353

Table 15.1-8: Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1172 lbm/s
RCS average temperature	545 °F	+10 °F
SG temperature	500 psia	+35psia
MTC	EOC	Most Negative
DTC	BOC	Least Negative

Table 15.1-9: Increase in Steam Flow (15.1.3) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2018 psia
Maximum SG Pressure	2310 psia	1208 psia
MCHFR	1.284	1.957

Table 15.1-10: Sequence of Events (15.1.5 Steam Line Break, Limiting Reactor Pressure Vessel Pressure Case)

Event	Time [s]
SLB occurs	0
AC power is lost/turbine trip/feedwater pump trip	0
High pressure limit is reached	19
Reactor trips on high pressure signal	21
DHRS actuated	21
RSV lift point is reached	25
Peak RPV pressure reached	26
RSV reseats	35

Table 15.1-11: Sequence of Events (15.1.5 Steam Line Break, Limiting Minimum Critical Heat Flux Ratio Case)

Event	Time [s]
SLB occurs	0
High reactor power limit is reached	47
Peak reactor power reached	49
Control rods fully inserted	51
High pressurizer pressure limit is reached	52
DHRS actuated	54
Peak MSS pressure reached	128

Table 15.1-12: Sequence of Events (15.1.5 Steam Line Break, Radiological Input Case)

Event	Time [s]
SLB occurs	0
High power limit reached	38
Reactor trip	40
Low steam superheat limit reached	152
DHRS actuated	154
FWIV and MSIV fully closed	161
Dryout of affected SG line	477

Table 15.1-13: Steam Piping Failure - Inputs

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia (Limiting MCHFR) -70psia (Limiting RPV pressure)
Pressurizer Level	60%	+8% (Limiting MCHFR) -8% (Limiting RPV pressure)
RCS flow rate	See Table 15.0-6 for range	1179 lbm/s
RCS average temperature	545 °F	+10°F
SG pressure	500 psia	+35psia
DHRS heat transfer multiplier	1	+30%
Core Exposure (Limiting MCHFR)	EOC	Most Negative MTC, Least Negative DTC
Core Exposure (Limiting RPV pressure)	BOC	Most Positive MTC and DTC

Table 15.1-14: Steam Piping Failure - Inputs to Radiological Analysis

Parameter	Units	SLB with Maximum Mass Release[1]	SLB with Maximum Spiking Time[1]
Integrated mass through break - pre-trip	lbm	7500	1550
Integrated mass through break - from trip to SG line empty	lbm	8300	10500
Integrated secondary flow - intact steam line - pre-trip	lbm	14000	4000
Integrated secondary flow - intact steam line - from trip to isolation	lbm	2300	3300

[1] Values include margin added to mass release calculated by NREALP

Table 15.1-15: Steam Piping Failure (15.1.5) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2156 psia
Maximum SG Pressure	2310 psia	1346 psia
MCHFR	1.284	1.861

Table 15.1-16: Loss of Containment Vacuum/Containment Flooding - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
Pressurizer Level	50%	+5%
RCS flow rate	See Table 15.0-6 for range	1179 lbm/s
RCS average temperature	545 °F	+10 °F
Reactor pool temperature	N/A	40 °F[1]
MTC	EOC	Most Negative
DTC	BOC	Least Negative

[1]Reactor pool temperature is assumed to be at a conservatively low value of 40 °F

Table 15.1-17: Loss of Containment Vacuum/Containment Flooding (15.1.6) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	1992 psia
Maximum SG Pressure	2310 psia	1342 psia
MCHFR	1.284	2.761

Figure 15.1-9: Critical Heat Flux Ratio (15.1.1 Decrease in Feedwater Temperature)

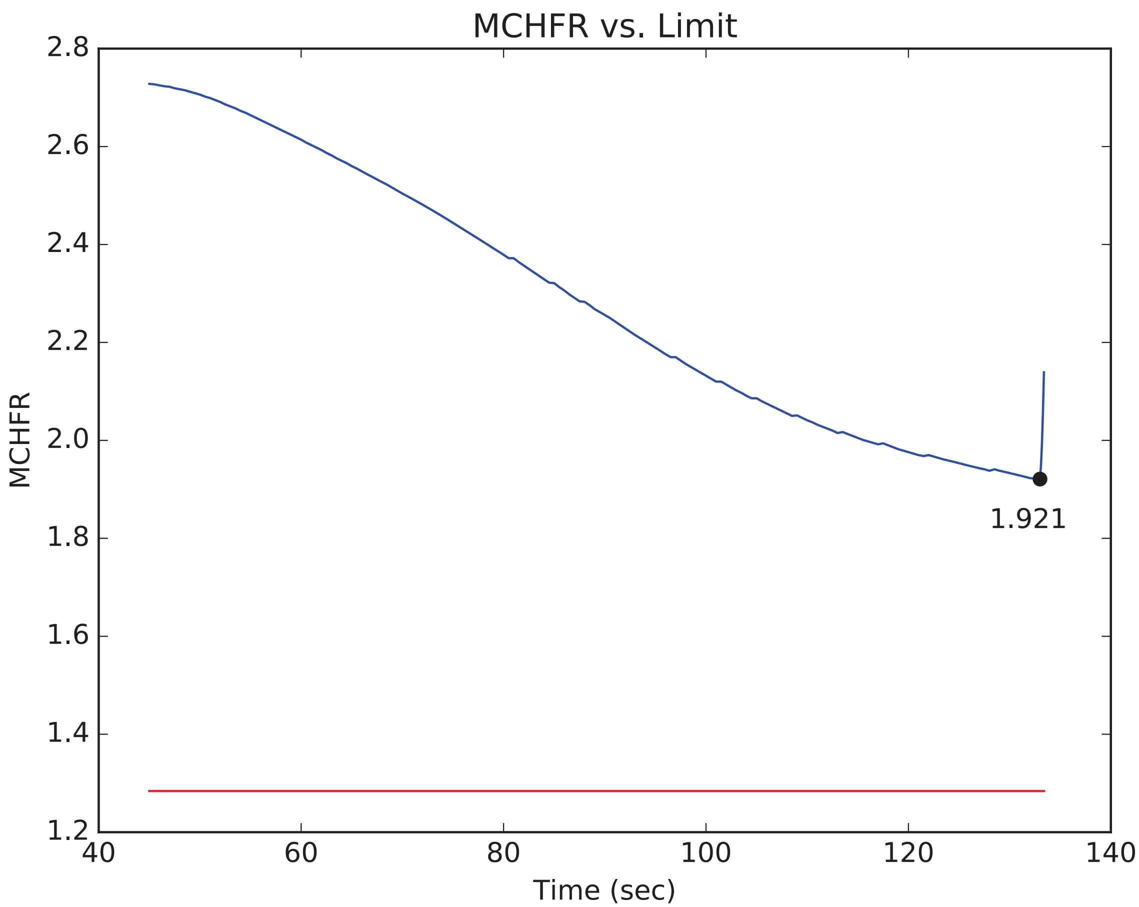


Figure 15.1-19: Critical Heat Flux Ratio (15.1.2 Increase in Feedwater Flow)

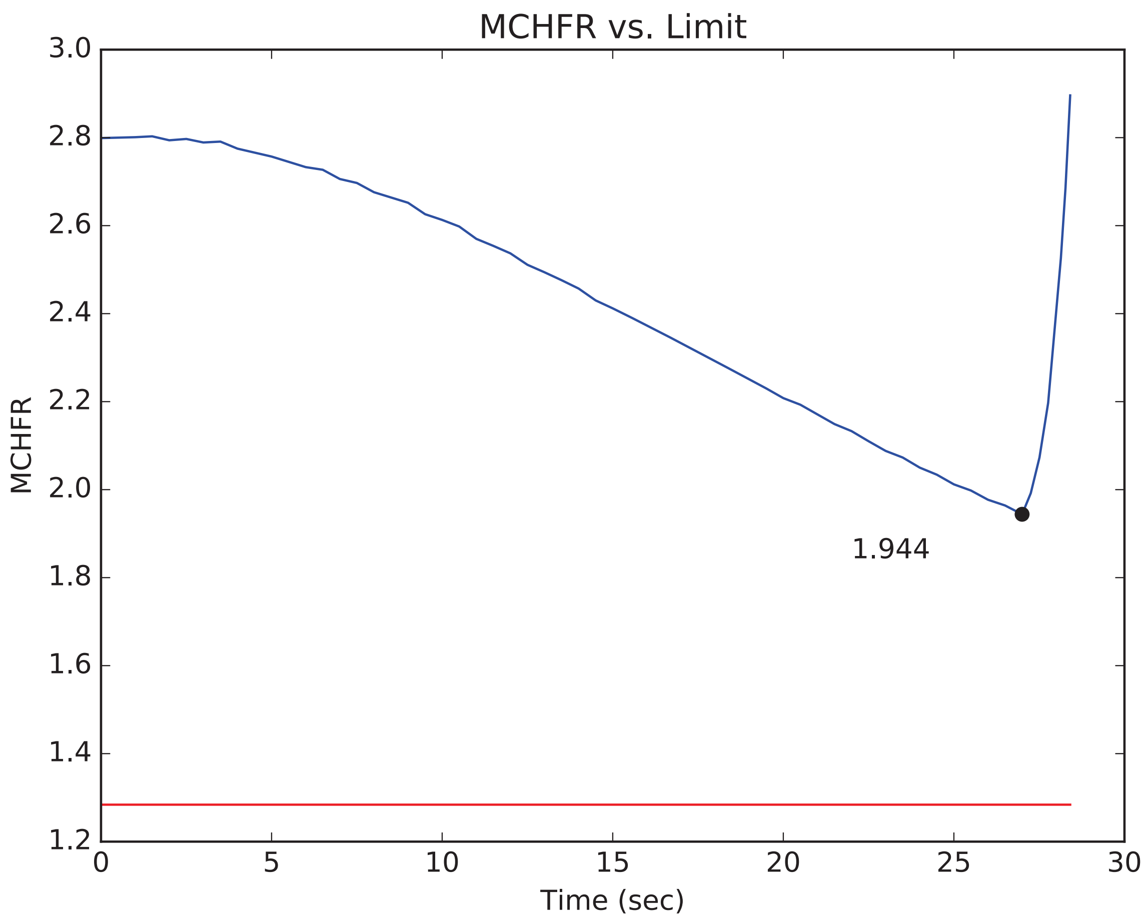


Figure 15.1-30: Critical Heat Flux Ratio (15.1.3 Increase in Steam Flow)

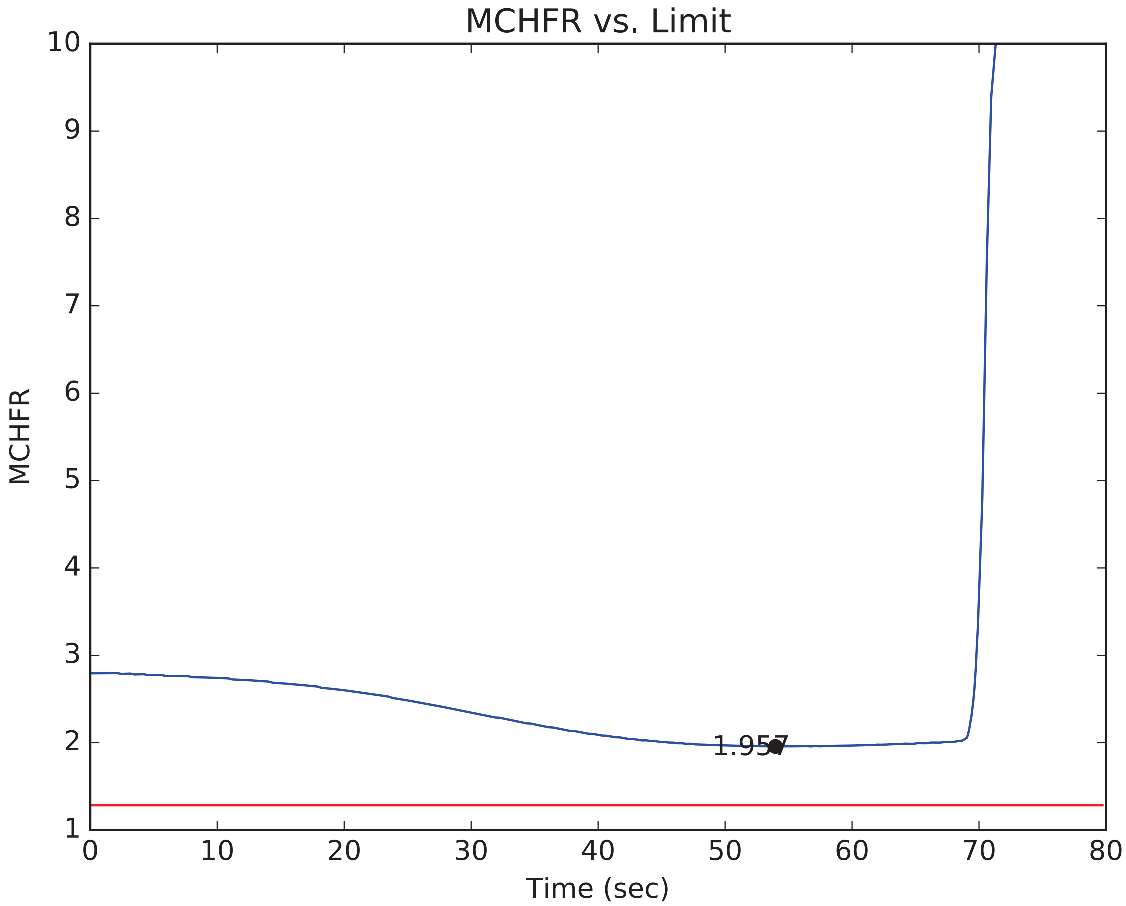


Figure 15.1-44: Critical Heat Flux Ratio (15.1.5 Steam Piping Failure, Limiting Minimum Critical Heat Flux Ratio)

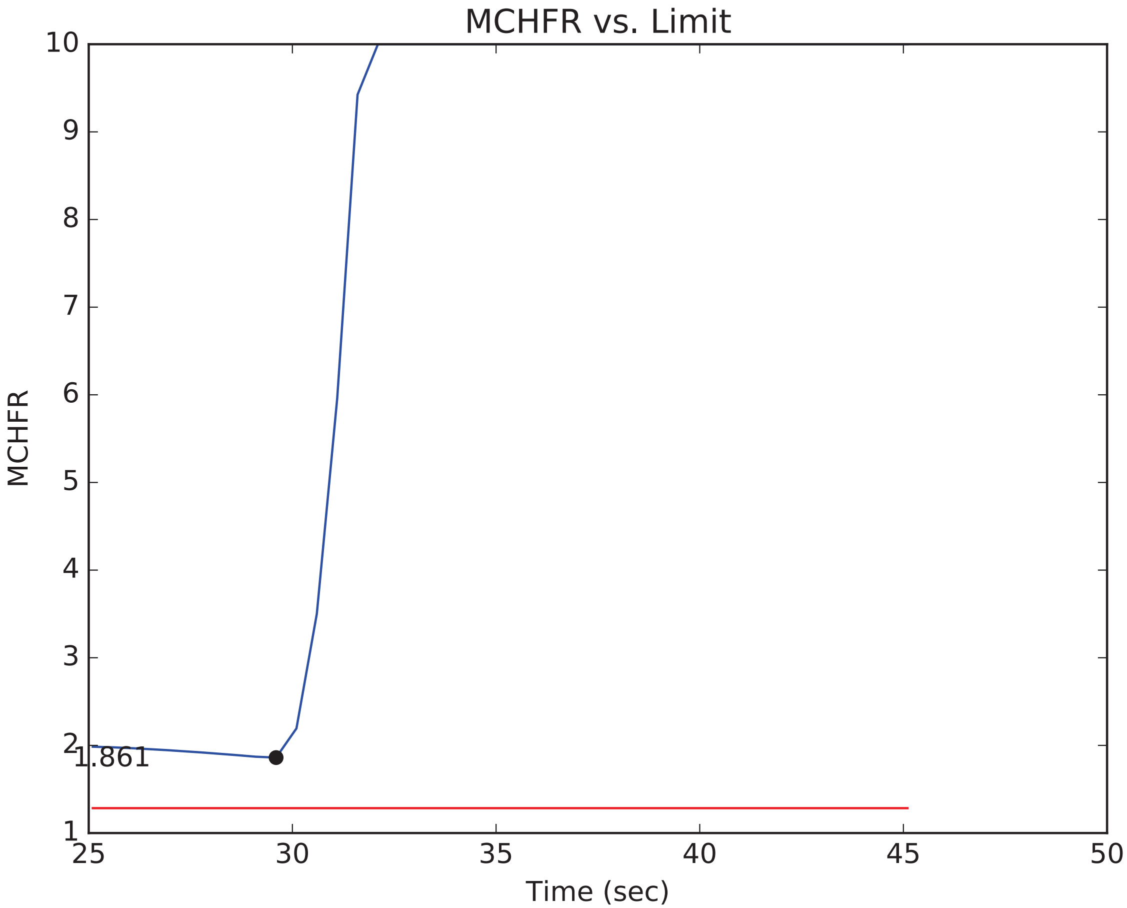


Figure 15.1-53: Critical Heat Flux Ratio (15.1.6 Loss of Containment Vacuum/Containment Flooding)

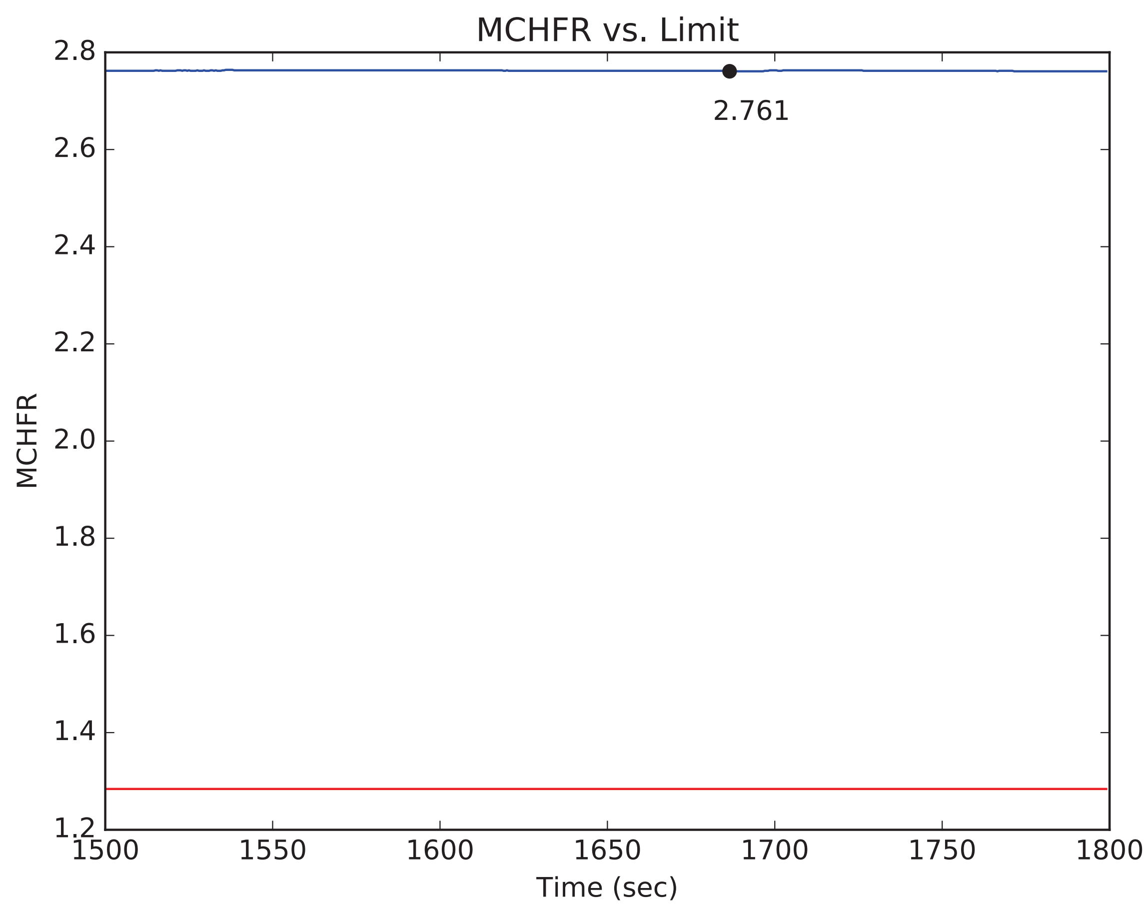


Table 15.2-7: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2158 psia
Maximum SG Pressure	2310 psia	1474 psia
MCHFR	~~1.262~~1.284	~~1.826~~2.579

Table 15.2-14: Main Steam Isolation Valve Closure - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2158
Maximum SG Pressure	2310 psia	1481
MCHFR	~~1.262~~1.284	~~1.894~~2.567

Table 15.2-19: Loss of Non-Emergency AC Power - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2162 psia
Maximum SG Pressure	2310 psia	1361 psia
MCHFR	~~1.262~~1.284	~~1.820~~2.569

Table 15.2-21: Loss of Feedwater Event - Maximum SG Pressure - Sequence of Events

Event	Time [s]
Loss of feedwater initiation Feedwater flow begins 0.1 second ramp down to 97.7% of initial value	0
RCS hot leg high temperature MPS signal	690
~~Minimum CHF ratio~~	~~696~~
Turbine Trip	697
Loss of Normal AC	697
MSIVs close signal	697
RTS actuation	698
DHRS actuation	698
Control rods fully inserted	700
Peak RCS pressure	708
DHRS valve fully open	728
Peak secondary pressure	769

Table 15.2-23: Loss of Feedwater - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2159 psia
Maximum SG Pressure	2310 psia	1422 psia
MCHFR	~~1.262~~1.284	~~1.831~~2.569

Table 15.2-29: Feedwater Line Break- Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2164 psia
Maximum SG Pressure	2310 psia	1328 psia
MCHFR	~~1.262~~1.284	~~1.798~~2.607

Table 15.2-33: Inadvertent Operation of Decay Heat Removal System - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2163 psia
Maximum SG Pressure	2310 psia	1582 psia
MCHFR	~~1.262~~1.284	~~1.856~~2.489

Figure 15.2-9: Hot Channel Node MCHFR - Limiting MCHFR Case (15.2.1-15.2.3 LOEL-TT-LOCV)

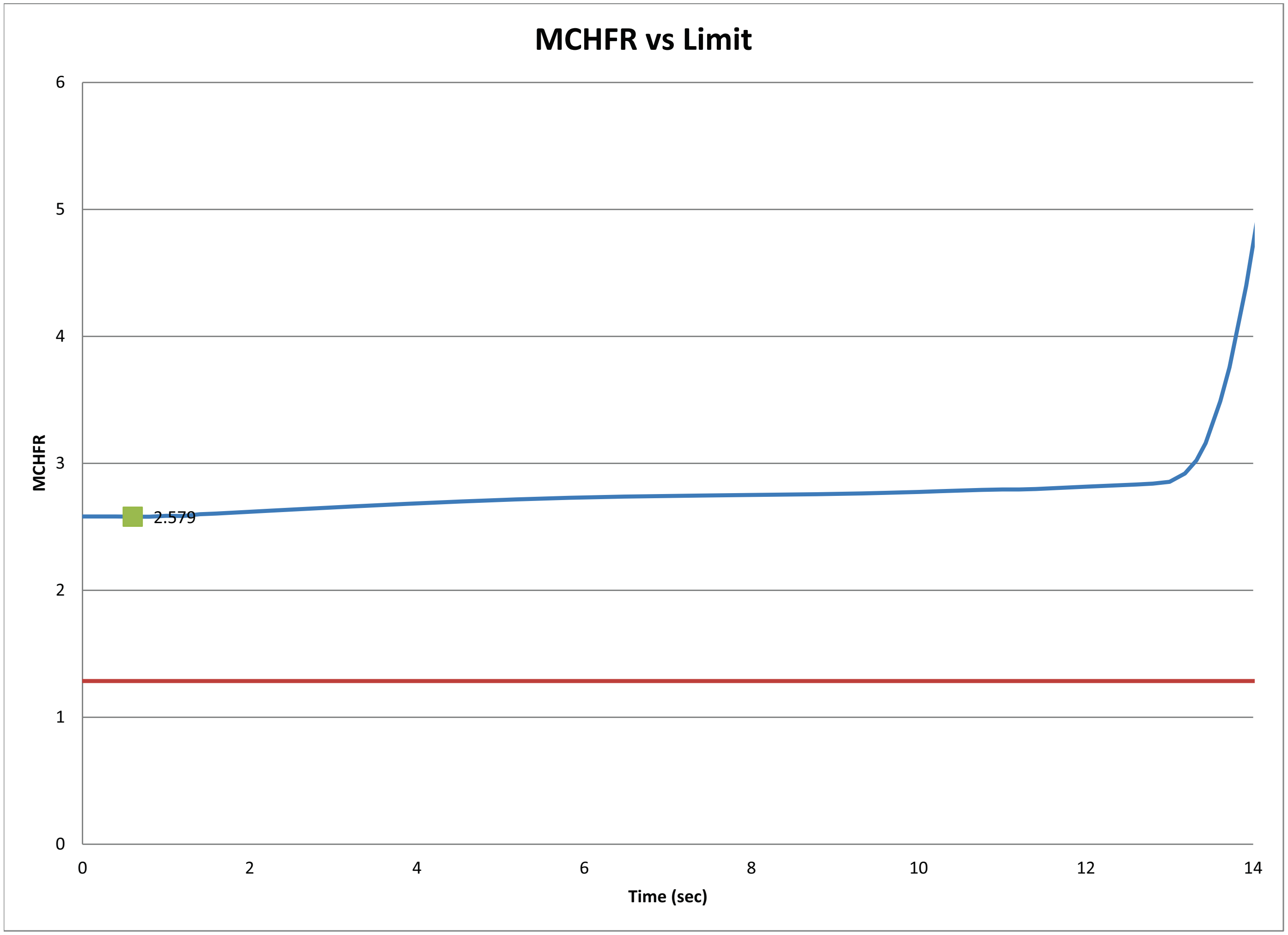


Figure 15.2-18: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.4 MSIV Closure)

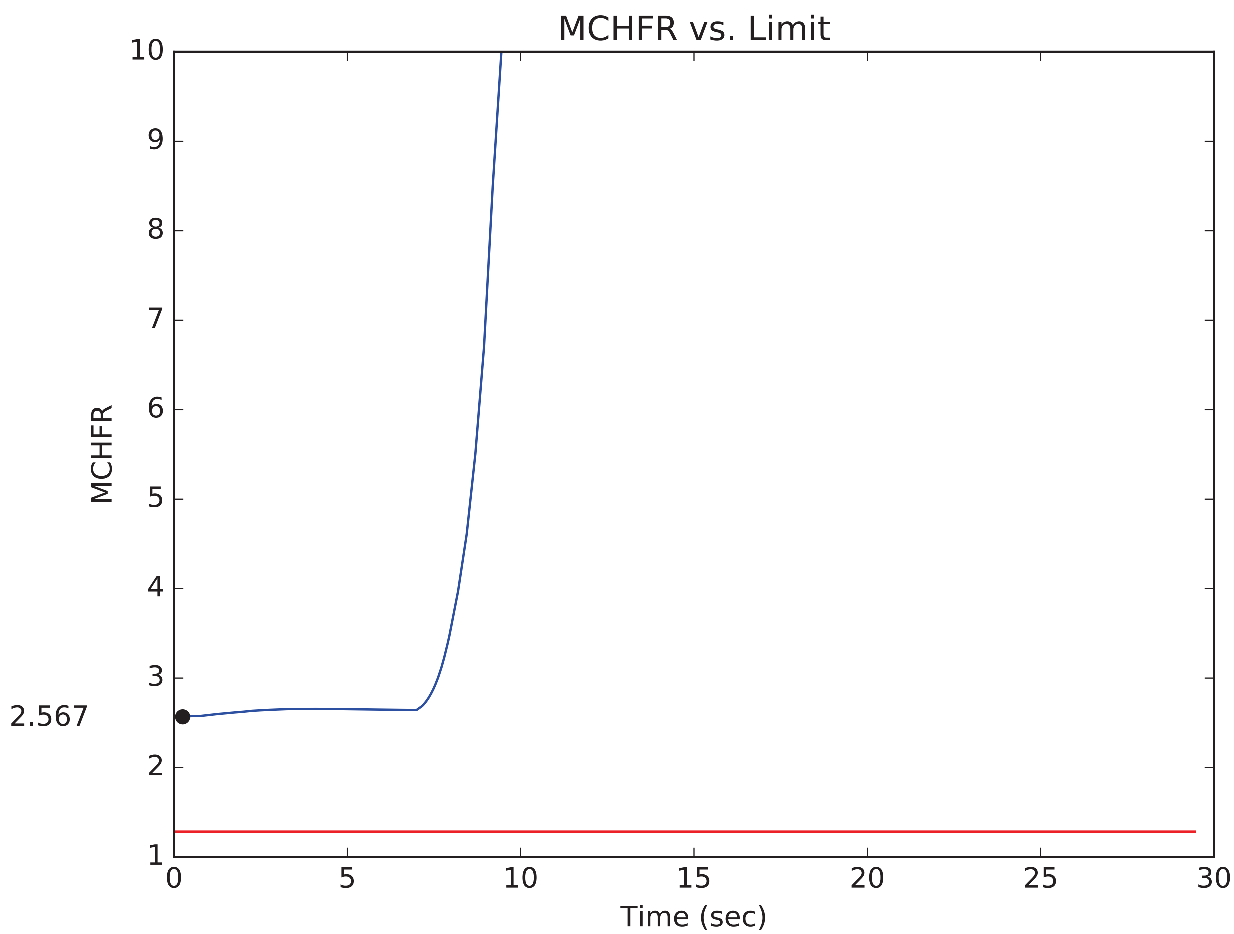


Figure 15.2-26: Hot Channel Node Critical Heat Flux Ratio - MCHFR Case (15.2.6 Loss of AC Power)

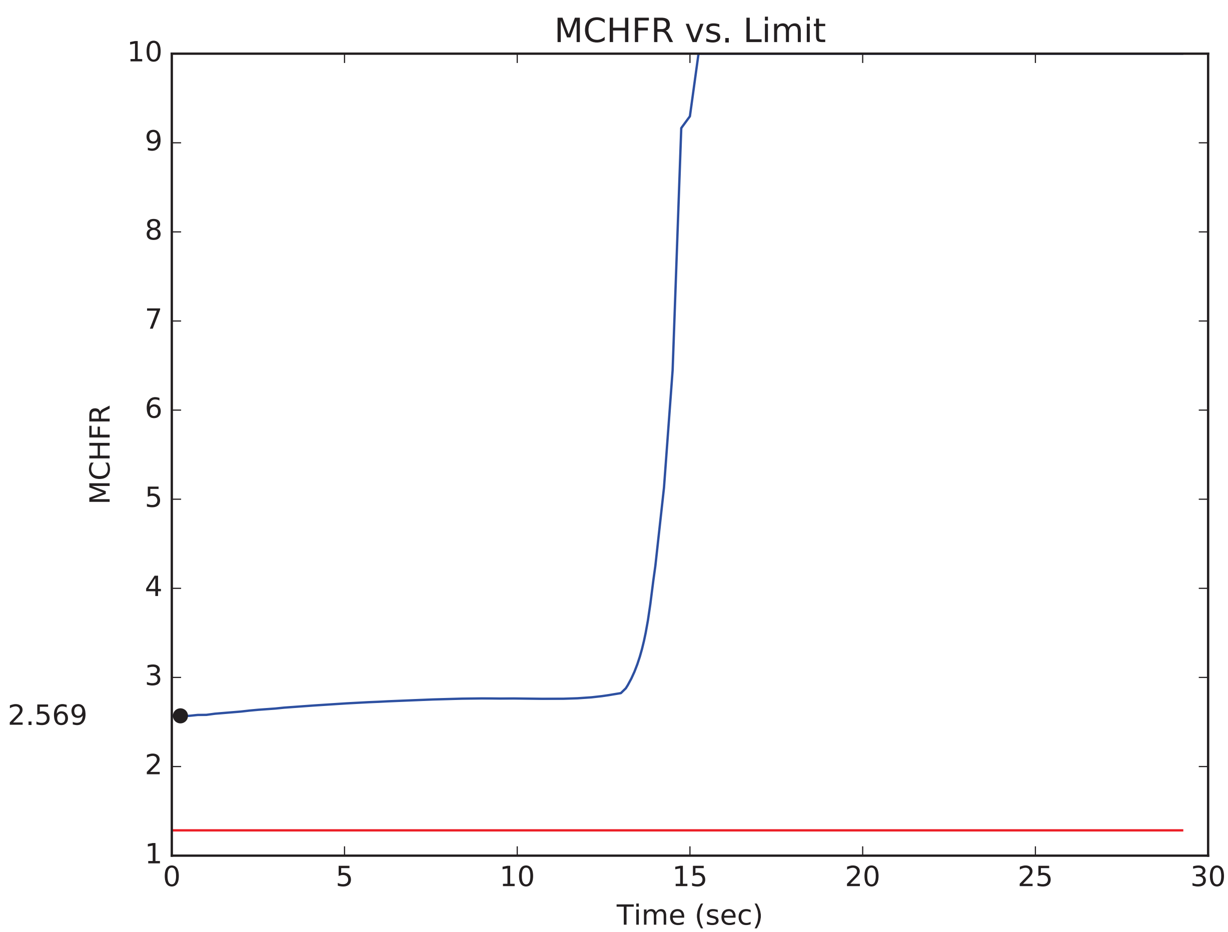
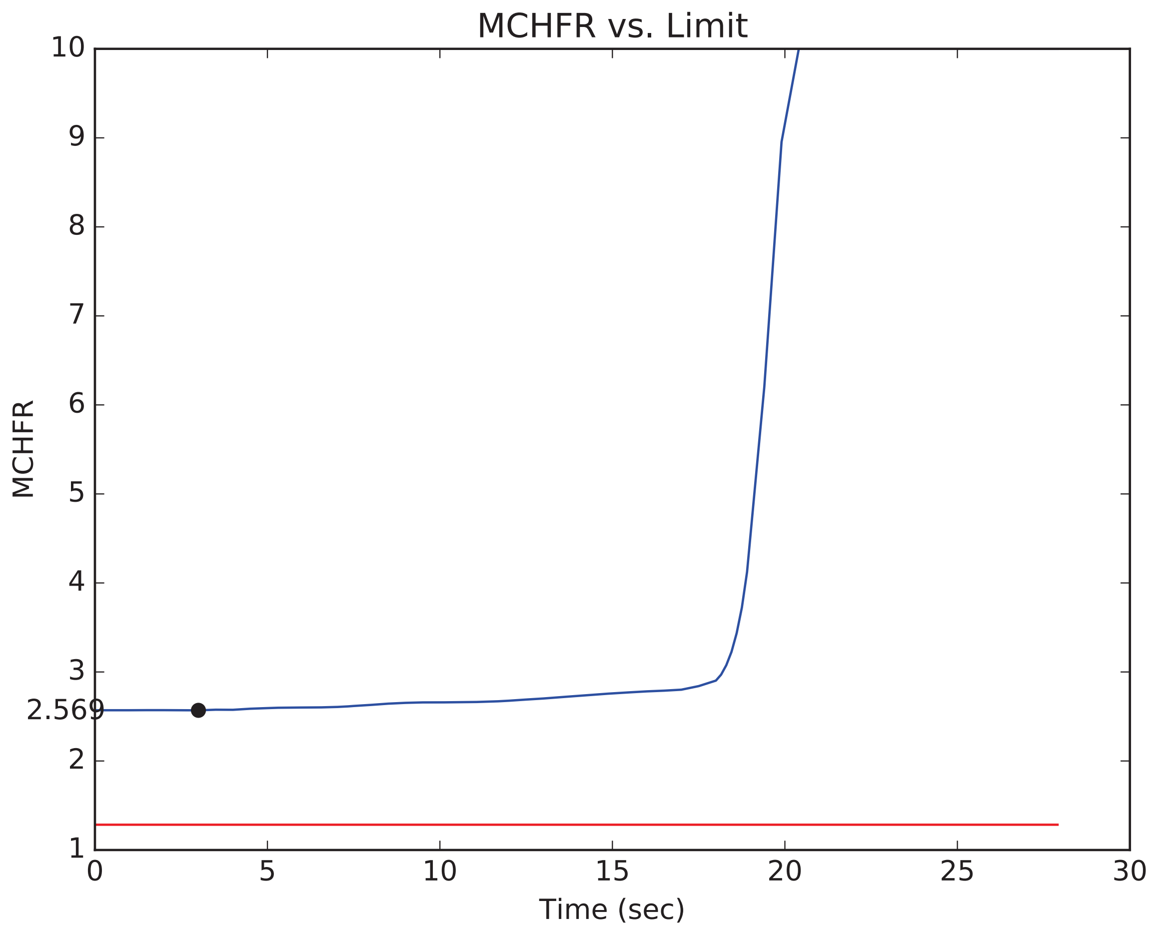


Figure 15.2-34: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.7 Loss of Feedwater)



RAI 15.02.08-1

Figure 15.2-44: Hot Channel Node MCHFR - ~~Peak RCS Pressure~~Limiting MCHFR Case (15.2.8 Feedwater Line Break)

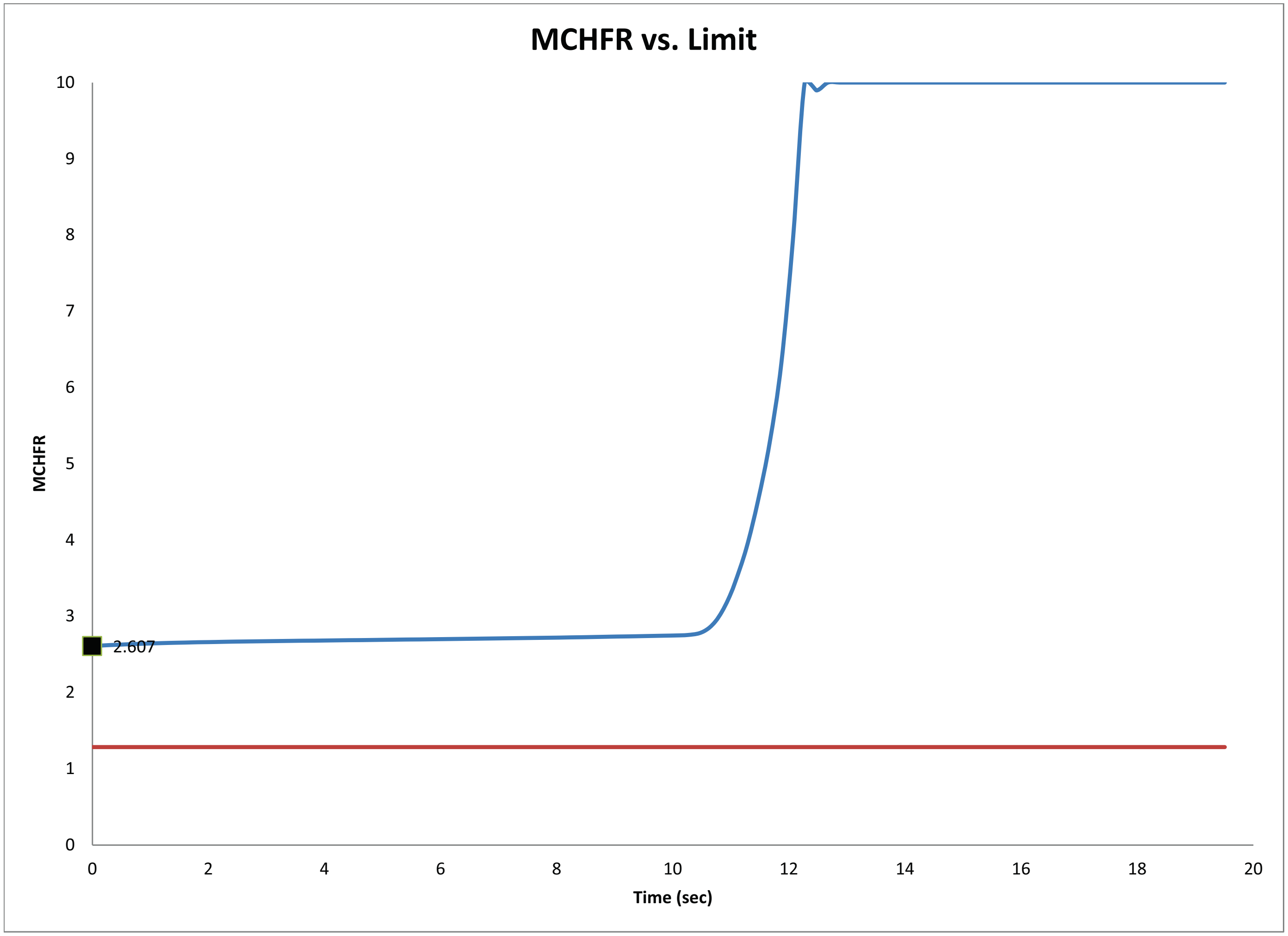
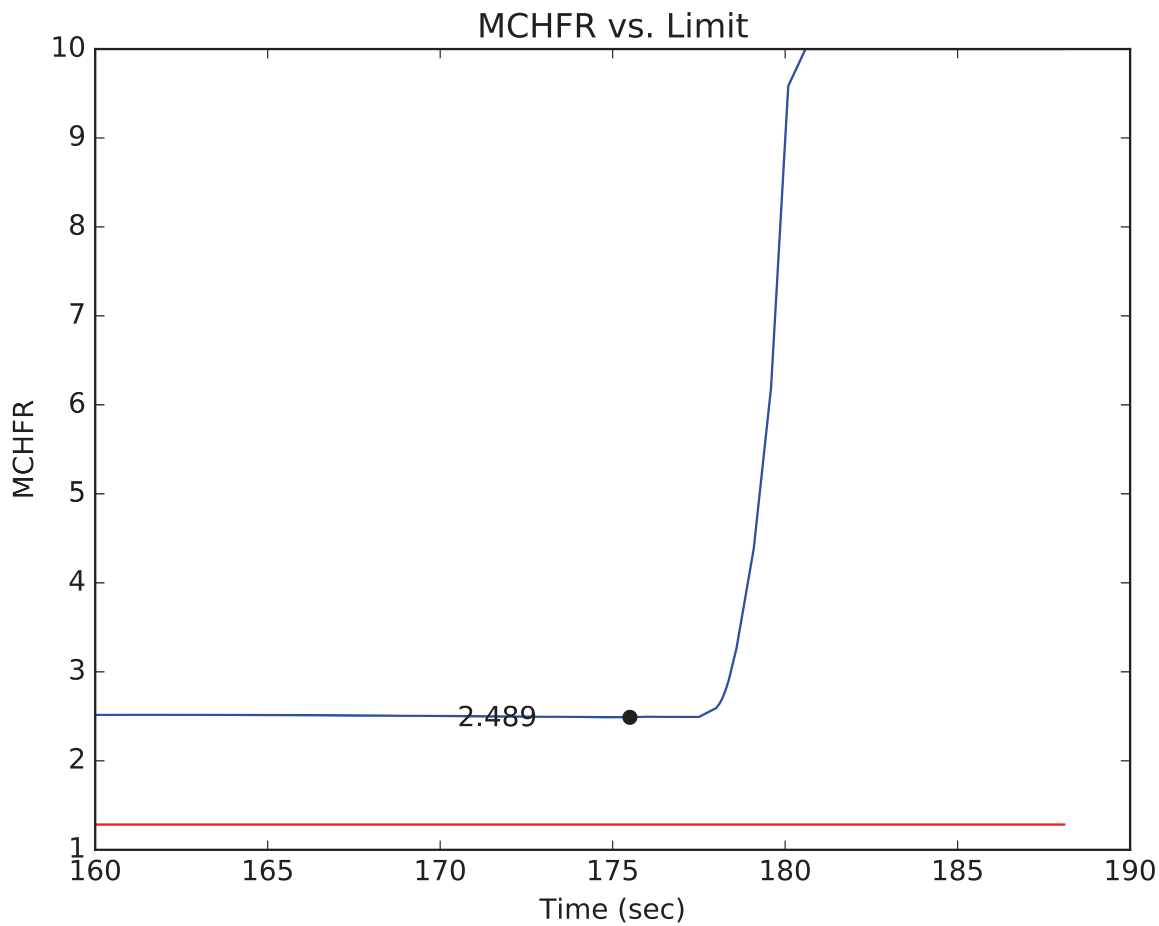


Figure 15.2-55: Minimum Critical Heat Flux Ratio - Limiting MCHFR Case (15.2.9 Inadvertent Operation of DHRS)



- The maximum worth is assumed in the bank withdrawal to provide the highest possible peak power.

RAI 15.04.01-1, RAI 15.04.01-2

- Reactivity insertion rate: The positive reactivity inserted by the CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. The maximum rod speed of 15 inches/min corresponds to a maximum reactivity insertion of 18.15 pcm/s. However to bound the reactivity insertion from possible boron dilution scenarios, a maximum reactivity insertion of 35 pcm/s is analyzed.
  - The reactivity insertion rate for the limiting MCHFR case is 13.37 pcm/s.
  - The reactivity insertion rate for the limiting RCS pressure case is 0.01337 pcm/s.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel model is discussed in Section 15.0.2.

15.4.1.3.3 Results

RAI 15.04.01-1, RAI 15.04.01-2

The sequence of events for a representative uncontrolled CRA withdrawal from a low power or startup condition is provided in Table 15.4-1. Figure 15.4-1 through Figure 15.4-5 show the transient behavior of key parameters for the case that is limiting with respect to MCHFR. Figure 15.4-34 shows the RCS pressure from the limiting pressure case.

The CRA bank begins to withdraw at the transient initiation, which begins to raise power, RCS temperature, and RCS pressure. The cases that are limiting for RCS pressure and MCHFR have an initial power of 1MW. This initial power, coupled with a slow withdrawal rate, demonstrates that initially avoiding the high power and power rate trips allows a heatup and pressurization prior to scram. The reactor power rises until the high power (25%) limit is reached. This initiates a scram 2 seconds later, when the peak power is achieved. The maximum pressure occurs after the scram has completed.

The limiting cases for an uncontrolled CRA withdrawal from a low power or startup condition demonstrate margin to the acceptance criteria. The peak RCS pressure for this CRA withdrawal is shown to be below the design pressure of the RPV. The limiting MCHFR for this event is above the design limit, and the fuel centerline temperature is below the fuel melting temperature. The limiting values for these acceptance criteria are ~~discussed in Section 15.4.1.5~~shown in Table 15.4-3.

15.4.1.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.1.5 Conclusions

The two applicable acceptance criteria for this AOO are met for the limiting cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

- The thermal margin limits departure from nucleate boiling ratio for PWRs as specified in SRP Section 4.4, subsection II.1, are met.
  - The MCHFR for the limiting case is ~~8.657, which is~~ above the design limit, as shown in Table 15.4-3.
- Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.
  - The fuel centerline temperature of the limiting case ~~is 890.8°F, which~~ is below the fuel melting temperature, as shown in Table 15.4-3.

RAI 15.04.01-1, RAI 15.04.01-2

The evaluation of an uncontrolled CRA withdrawal from a subcritical or low power startup condition demonstrates that the RCS pressure and SG pressure do~~does~~ not exceed the ~~RPV~~ design limits, as shown in Table 15.4-3. The limiting peak RCS pressure for this event is ~~2038 psia~~acceptable, as shown in Figure 15.4-34.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

A spurious CRA withdrawal that occurs when the reactor is at power leads to an unexpected addition of positive reactivity into the reactor. An uncontrolled CRA withdrawal at power results in an increase in core power with a corresponding increase in heat flux. Due to the time lag in the response of the secondary system, the heat removal from the steam generators follows the heat increase in the primary system. The result is an increase in RCS temperature and pressure. These conditions could challenge design pressures and the SAFDLs. The power range neutron excore detectors provide high power and high flux rate core protection. For cases where the reactivity insertion is sufficiently slow, the high pressurizer pressure and high hot leg temperature limits provide protection. These MPS limits are analyzed for a spectrum of uncontrolled CRA withdrawal conditions to ensure that protection functions are actuated to prevent the violation of the design safety limits.

An uncontrolled CRA withdrawal is expected to occur one or more times in the life of the reactor, and it is classified as an AOO. The categorization of the NuScale design basis events are provided in Table 15.0-1.

uncontrolled CRA withdrawal cases with higher reactivity insertion rates, the MPS trips the reactor on high pressurizer pressure or high power rate. These cases are non-limiting because the reactor is tripped before the maximum amount of reactivity can be inserted. The limiting combination of reactivity insertion and reactivity feedback produces the maximum possible power increase prior to trip. The power increase in the limiting MCHFR case is terminated by a reactor trip after a signal delay. The high hot leg temperature limit, the high pressurizer pressure limit, and high power limit are all reached during the reactor trip delay time. The MPS trips the reactor and actuates the DHRS during this event. The most limiting MCHFR occurs at the time of the power peak. The MCHFR remains above the design limit, and no fuel centerline melting is predicted for the uncontrolled CRA withdrawal.

The maximum RCS pressure case is an uncontrolled CRA withdrawal at power with a loss of normal AC power at transient initiation. The pressure for the maximum pressure case is demonstrated in Figure 15.4-12. The loss of AC power at the beginning of the transient trips the turbine and stops feedwater, reducing the heat removal by the secondary side. Simultaneously, the reactivity insertion causes a rapid rise in power. The reactor trips on high power rate, reaching the high pressurizer pressure setpoint almost simultaneously. The pressure continues to rise after the reactor trip, and peaks at the time a reactor safety valve (RSV) opens. Following the RSV opening and reactor trip, the RCS temperature and pressure steadily decrease. The maximum RCS pressure stays below the RPV design limit.

The uncontrolled CRA withdrawal at power cases that result in a reactor trip, actuate DHRS, and maintain stable core cooling.

15.4.2.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.2.5 Conclusions

The two applicable acceptance criteria for this AOO are met for the limiting uncontrolled CRA withdrawal cases. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below.

1) The thermal margin limits departure from nucleate boiling ratio for pressurized water reactors as specified in SRP Section 4.4, subsection II.1, are met.

   - The MCHFR for the limiting uncontrolled CRA withdrawal ~~is 1.316, which~~ is above the design limit, as shown in Table 15.4-6. Therefore, this criterion is met.

2) Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.

RAI 15.04.06-1

   - As discussed in Reference 15.4-1, a steady-state linear heat generation rate (LHGR) protection limit can be applied to an uncontrolled CRA withdrawal at power event to ensure that the fuel centerline temperatures do not exceed the melting point. The LHGR for the limiting CRA withdrawal ~~is 8.44, which~~ is below the limit, as shown in Table 15.4-6.

RAI 15.04.02-1, RAI 15.04.02-2

The evaluation of an uncontrolled CRA withdrawal ~~from a subcritical or low power startup condition~~at power demonstrates that the RCS pressure does not exceed the RPV design limit, as shown in Table 15.4-6. ~~The limiting peak RCS pressure for this event is 2160 psia~~.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

A control rod misoperation (CRM) could introduce an unexpected change of reactivity in the core. The resulting change in power distribution could lead to a decrease in the CHF ratio. CRM events analyzed in this section are:

- Control rod assembly misalignments. A single CRA or multiple CRAs are displaced in a position relative to the bank while the other rods in its bank are in another allowed position.
- Single CRA withdrawals. A single CRA is inadvertently withdrawn from the bank insertion limit. The withdrawal of an entire bank of CRAs is analyzed in Section 15.4.1 and Section 15.4.2.
- Control rod assembly drop. A single CRA or multiple CRAs are dropped inadvertently into the reactor core.

The CRA misalignment is an event where a single or multiple CRAs are out of alignment with the remaining rods in the bank. The alignment may be higher or lower than the expected rod position. A misalignment can occur based on the uncertainty in the rod position from its indicated or expected position. A limiting misalignment occurs when the core is operating at steady-state full power with the rods inserted to the PDILs except one rod is left withdrawn. This is a postulated condition as the rod position indicators will alarm when the rods are out of alignment beyond the position uncertainty. The effects of this misalignment are bounded by the analysis of the withdrawal of a single CRA. Another limiting misalignment that is postulated is where all CRAs are withdrawn, except one that is misaligned in to the 25% rated power PDIL position. The analysis of this misalignment is presented in this section. The postulated misalignment of the rods inserted to the PDIL but with one CRA fully inserted is not a credible condition for the NuScale core. Reactor hold points will prohibit the movement of rods for that severe of a peaking distortion and therefore is not analyzed for NuScale. The CRA misalignments are classified as AOOs.

The single rod withdrawal transient occurs when a control rod is set at the bank position PDIL and is postulated to withdraw. This event may occur due to wiring failures or operator error in which one rod is pulled with disregard for rod position information. The single rod withdrawal adds reactivity and initiates a power increase transient. The power distribution in the core becomes asymmetric and peaking can challenge the MCHFR safety limit. A CRM that results in a withdrawal is classified as an AOO.

power rate limit is reached just after 1 second into the transient. The MPS sends a reactor trip signal, terminating the event. At lower powers, the power decrease is less pronounced, and the reactor does not trip. In the lower power cases, the regulating CRA bank brings the reactor back to the initial power after an initial power overshoot. However, these cases are non-limiting with respect to MCHFR.

RAI 15.04.03-1

Exceeding the RPV design pressure is not a concern for the limiting rod drop case, which is demonstrated in the RCS pressure plot. The MCHFR for the limiting case, Figure 15.4-36, remains above the design limit. The LHGR calculated for the limiting rod drop case is below the limits for fuel melting and cladding strain.

15.4.3.6 Radiological Consequences

The normal leakage related radiological consequences of these events are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.3.7 Conclusions

The CRM events meet the SRP 15.4.3 acceptance criteria as follows:

1) The thermal margin limits departure from nucleate boiling ratio, as specified in SRP Section 4.4, subsection II.1, are met.

   - The MCHFR for the limiting CRA misalignment is ~~1.827, which is~~ above the CHF design limit, as shown in Table 15.4-11.
   - The MCHFR for the limiting single CRA withdrawal is ~~1.431, which is~~ above the CHF design limit, as shown in Table 15.4-11.
   - The MCHFR for the limiting CRA drop ~~is 1.316, which~~ is above the CHF design limit, as shown in Table 15.4-11.

2) Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.

   - As discussed in Reference 15.4-1, a steady-state linear heat generation rate protection limit can be applied to the CRM events to ensure that the fuel centerline temperatures do not exceed the melting point. The linear heat generation rate for the limiting CRA misalignment is ~~7.10 kW/ft, which is~~ below the limit for fuel melt, as shown in Table 15.4-11.
   - The linear heat generation rate for the limiting single CRA withdrawal is ~~7.84 kW/ft, which is~~ below the limit for fuel melt, as shown in Table 15.4-11.
   - The linear heat generation rate for the limiting CRA drop is ~~8.42 kW/ft, which is~~ below the limit for fuel melt, as shown in Table 15.4-11.

3) Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1 percent.

Rotational Misloads

The fuel assembly top nozzle has two holes that mate with pins in the upper core plate, and a third alignment hole that mates with the fuel handling equipment (Section 4.2). These features collectively, prevent fuel assembly rotational misloads. Nevertheless, 180 degree rotational misloads are conservatively examined.

15.4.7.3 Core and System Performance

15.4.7.3.1 Evaluation Model

The design and analysis of the NuScale Power Module reactor core is performed with the Studsvik Scandpower Core Management Software suite of reactor simulation tools. A discussion of the analysis tools and analytical methods is provided in Section 4.3.3.

SIMULATE is an advanced three-dimensional (3D), steady-state, multi-group nodal reactor analysis code capable of multi-dimensional nuclear analyses of reactors. SIMULATE is used to determine the limiting undetectable fuel misload, and to provide peaking factors to the subchannel analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 and Section 4.4 for a discussion of the VIPRE-01 code and evaluation model.

15.4.7.3.2 Input Parameters and Initial Conditions

The fuel misload event changes the power distribution of the core, but the thermal hydraulic boundary conditions remain the same. Therefore, there is no need for an NRELAP5 analysis to ensure that the RCS pressure remains below the design limit of the RPV. The power distribution of the equilibrium core analysis is discussed in Section 4.3.

The power peaking augmentation factors for the limiting undetectable misload are calculated using SIMULATE, and provided as input to the steady-state subchannel analysis to determine the MCHFR for this event. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1.

15.4.7.3.3 Results

The limiting undetectable fuel misloading event results in an MCHFR, which is above the 95/95 CHFR limit. Fuel temperature margin to centerline melt is calculated for the worst case fuel assembly misloading event. The calculated value of Linear Heat Generation Rate (LHGR) for the worst misload is below the limiting LHGR. These results are provided in ~~Table 15.4-15~~Table 15.4-20. Because MCHFR is above the limit and fuel centerline melting is not expected to occur, no fuel damage is expected. These events change the power distribution within the core,

- Average RCS temperature biased high - The higher temperature corresponds to a higher coefficient of expansion. This exacerbates the REA-induced core pressure pulse and inlet flow slow-down, minimizing MCHFR.
- RCS flow biased low - The lower core flow minimizes MCHFR.
- Fuel and gap conductivities are maximized - Maximizing the conductivities increases the energy flow into the coolant, which maximizes the inlet flow slow-down.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine if the MCHFR design limit is met for this event. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1. The results of the subchannel analysis and adiabatic fuel energy calculation determine if there is any potential fuel damage resulting from an REA. The REA event-specific methodology is provided in Reference 15.4-2.

15.4.8.3.3 Regulatory Criteria for NuScale

Reference 15.4-2 discusses the various REA regulatory acceptance criteria and how they apply to the NuScale design. A summary of these acceptance criteria are provided in this section.

Fuel Cladding Failure

- For zero power conditions, the high temperature cladding failure threshold is expressed in cladding differential pressure. The peak radial average fuel enthalpy must be below the 100 cal/g associated with the maximum peak rod differential pressure of $\Delta P \geq 4.5$ MPa.
- For intermediate and full power conditions, fuel cladding failure is presumed if local heat flux exceeds the critical heat flux (CHF) thermal design limit.
- The PCMI failure threshold is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 of NUREG-0800 SRP 4.2.
- If fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions, then fuel cladding failure is presumed.

Core Coolability

- Peak radial average fuel enthalpy will remain below 100 cal/g.
- No fuel melt will occur.

RCS Pressure

The maximum RCS pressure must remain below 120% of design pressure. Therefore, the peak pressure during an REA is limited ~~to~~below 2520 psia.

15.4.8.3.4 Fuel and Cladding Integrity Results

S3K provides the power and reactivity response to an REA for each statepoint discussed in Section 15.4.8.3.2. Each initial power level is evaluated for BOC, MOC (4.0 GWD/T), and EOC conditions. The S3K analysis assumes the maximum reactivity insertion from ejecting the highest worth CRA for each of these statepoints. ~~Table 15.4-15~~Table 15.4-22 provides the maximum power in percent HFP as well as the inserted reactivity of the ejected rod for a spectrum of initial power levels and times in cycle. Figure 15.4-29, Figure 15.4-30, and Figure 15.4-31 provide the maximum power pulse for ~~B~~EOC, MOC, and ~~E~~BOC conditions, respectively. The plots show a rapid reactivity excursion when the rod is ejected, but the power pulse is mitigated, due to fuel feedback effects. The reactor is tripped when the power reaches 15% above the initial power level, and the rods are inserted after a 2.0 second delay. At BOC and 4.0 GWD/T, the peak power pulse occurred at 70% power with the ejection of a rod from the inner bank of control rods. At EOC, the initial 50% power case provided the largest power pulse from an ejection from the inner bank of control rods. Sensitivities of power cases around the EOC, 50% case reveal that the 55% power case at EOC produced the largest power pulse at 661% of HFP.

An adiabatic fuel response calculation evaluates these power responses to determine if any fuel failures occur due to the fuel temperature or enthalpy increase. ~~Table 15.4-16~~Table 15.4-23 provides the peak fuel temperature, change in fuel enthalpy, and net fuel enthalpy for a spectrum of initial power levels and times in cycle. A summary of the limiting ~~results shown in the table~~conditions below is ~~provided below~~in Table 15.4-24:

- The limiting peak radial average fuel enthalpy at zero power conditions ~~is 34.6 cal/g.~~
- The limiting peak radial average fuel enthalpy for intermediate and full power conditions ~~is 84 cal/g.~~
- The limiting change in radial average fuel enthalpy ~~is 28.7 cal/g.~~
- The limiting peak fuel temperature ~~is 2162°F.~~

The subchannel analysis evaluates a spectrum of REA conditions that are provided by the S3K and NRELAP5 analyses. The REA case that results in the limiting MCHFR is an REA that occurs at an initial power of 70% at EOC. The limiting MCHFR ~~is 1.47, which~~ is above the design limit as demonstrated in ~~Figure 15.4-32~~ Figure 15.4-37.

The maximum possible mass and energy release to containment due to a postulated control rod housing failure is bounded by an inadvertent opening of an RVV. A postulated control rod housing failure would represent a maximum break size of 2.375 inch (inner diameter). This break size is smaller than the opening of an RVV. The additional energy from the power excursion of an REA is not sufficient to exceed the energy release of an inadvertent opening of an RVV. The inadvertent opening of an RVV is discussed in Section 15.6.6. The limiting containment peak pressures and temperatures for design basis events are discussed in Section 6.2.

The REA acceptance criteria for fuel cladding failure and core coolability are met by the NuScale design, which indicates that no fuel failures are predicted in the event of an REA.

15.4.8.3.5 Maximum Reactor Coolant System Pressure Results

The sequence of events provided in Table 15.4-21 is for the REA that results in the limiting RCS pressure. Figure 15.4-33 shows the pressure for this REA case. The pressure rises until it peaks a few seconds after the reactor trip. The maximum pressure ~~of 2076 psia~~shown in Table 15.4-24 is below the safety valve opening limit and well below 120 percent of the RPV design pressure.

15.4.8.4 Radiological Consequences

No fuel damage is predicted for the limiting REA. Therefore, the radiological consequences of a REA are bounded by the consequences of other accidents presented in Section 15.0.3.

15.4.8.5 Conclusions

The applicable acceptance criteria for this accident are met for the limiting REA cases. These acceptance criteria are provided in Section 15.4.8.3.3. The NuScale Power Plant design meets these criteria as discussed in the summary below.

Fuel Cladding Failure

- The limiting peak radial average fuel enthalpy at zero power conditions is ~~34.6 cal/g, which is~~ below the fuel cladding failure limit ~~of 100 cal/g~~, as shown in Table 15.4-24.
- For intermediate and full power conditions, the limiting MCHFR is ~~1.47, which is~~ above the design limit, as shown in Table 15.4-24.
- The limiting change in radial average fuel enthalpy is ~~28.7 cal/g, which is~~ below 75 cal/g (A conservative value on Figure B-1 of NUREG-0800 SRP 4.2).
- The limiting peak fuel temperature is ~~2162°F, which is~~ below the fuel melting temperature, as shown in Table 15.4-24.

These limiting fuel cladding results indicate no fuel failures.

Core Coolability

- The limiting peak radial average fuel enthalpy is ~~84 cal/g, which is below 100 cal/g~~below the limit, as shown in Table 15.4-24.
- The limiting peak fuel temperature ~~is 2162°F, which~~ is below the fuel melting temperature, as shown in Table 15.4-24.

These limiting results indicate that core coolability will be maintained for the limiting REA.

RCS Pressure

- The limiting RCS peak pressure is ~~2076 psia, which is~~ below the RPV design limit ~~of 2520 psia~~, as shown in Table 15.4-24.

This limiting result indicates that RPV integrity is maintained during the limiting REA.

15.4.9 Spectrum of Rod Drop Accidents

15.4.9.1 Identification of Causes and Accident Description

This event is specific to boiling water reactors and not applicable to the NuScale design. The pressurized water reactor equivalent of a rod drop, the rod ejection, is addressed in Section 15.4.8. Control rod mioperations, including a dropped control rod assembly, are addressed in Section 15.4.3.

15.4.10 References

15.4-1 NuScale Power LLC, "Subchannel Analysis Methodology," TR-0915-17564-P, Revision 1~~0~~.

15.4-2 NuScale Power LLC, "Rod Ejection Methodology," TR-0716-50350, Revision 0.

Table 15.4-1: Sequence of Events for Limiting MCHFR Case (15.4.1 Uncontrolled CRA Withdrawal from Subcritical or Low Power Condition)

Event	Time [s]
Rod withdrawal initiates	0
High power (25%) limit reached	~~102~~2204
Reactor trip actuation	~~104~~2206
Maximum power/MCHFR	~~105~~2207
Scram complete	~~106~~2209

Table 15.4-2: Key Inputs for Limiting MCHFR Case (15.4.1 Uncontrolled CRA Withdrawal from Subcritical or Low Power Condition

Parameter	Nominal	Bias
Initial power	1 MW	N/A[1]
Pressurizer pressure	1848 psia	Nominal
Pressurizer level	60%	+8%
Core inlet temperature	425 °F	+80 °F
MTC	+6 pcm/°F	Most positive
FTC	-1.40 pcm/°F	Least negative

[1]A spectrum of initial powers is analyzed, and this value provided the limiting MCHFR results.

Table 15.4-3: Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition (15.4.1) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2038 psia
Maximum SG Pressure	2310 psia	685 psia
MCHFR	1.284	10.0
Maximum Fuel Centerline Temperature	4791°F	890.8°F

RAI 15.04.02-1, RAI 15.04.02-2

Table 15.4-4: Sequence of Events MCHFR Case - 75% Power (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

Event	Time [s]
CRA bank begins to withdraw	0
High hot leg temperature limit reached	~~72~~178
High ~~reactor power~~pressurizer pressure limit reached	~~78~~184
High ~~pressurizer pressure~~reactor power limit reached	~~78~~186
Reactor trip actuated	~~80~~186
MCHFR occurs	187
Maximum RCS pressure occurs	~~89~~191
DHRS valves fully open	~~110~~217

RAI 15.04.02-1, RAI 15.04.02-2

Table 15.4-5: Key Inputs for Limiting MCHFR Case (15.4.2 Uncontrolled CRA Withdrawal at Power)

Parameter	Nominal	Bias
Initial power	160 MW	~~+2~~Analyzed 75%
RCS Flowrate	See Table 15.0-6 for range	~~1178.2~~1056.7 lbm/s (low[1])
RCS Pressure	1850 psia	~~Nominal~~ -70 psia
Pressurizer Level	60%	~~Nominal~~ -8%
MTC	0.0 pcm/°F	Most positive
FTC	-1.40 pcm/°F	Least negative

[1] RCS flow rate is near the minimum for 75% power.

Table 15.4-6: Uncontrolled Control Rod Assembly Withdrawal at Power (15.4.2) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure (100% Power)	2310 psia	2160 psia
Maximum SG Pressure (100% Power)	2310 psia	1326 psia
MCHFR (75% Power)	1.284	1.624
Peak LHGR	21.22 kW/ft	8.97 kW/ft

Table 15.4-7: Sequence of Events (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

Event	Time [s]
Single CRA begins to withdraw	0
High hot leg temperature limit reached	155
High RCS pressure limit reached	162
Reactor trip actuation	163
Maximum RCS pressure occurs	170
DHRS valves fully open	193

Table 15.4-8: Key Inputs for Single CRA Withdrawal with Limiting MCHFR

Parameter	Normal	Bias
Initial power	75% of full power	Nominal
Pressurizer level	60%	-8%
RCS pressure	1850 psia	-70 psia
RCS average temperature	545 °F	-10 °F
MTC[1]	-6 pcm/°F	Least Negative
DTC	-1.4 pcm/°F	Least Negative

[1] Power dependent MTCs are used in the single CRA withdrawal analyses. The -6pcm/°F value corresponds to the initial power of 75%.

Table 15.4-9: Sequence of Events (15.4.3 Control Rod Misoperation, Control Rod Assembly Drop)

Event	Time [s]
CRA begins to drop	0
High power rate change limit reached	1
Reactor trip actuation	3
Low RCS pressure limit reached	71
DHRS actuation valves open	103

Table 15.4-10: Key Inputs for CRA Drop with Limiting MCHFR

Parameter	Nominal	Bias
Initial power	160 MW	+2%
RCS flow rate	See Table 15.0-6 for range	1176 lbm/s (low)
RCS pressure	1850 psia	+70 psia
RCS average temperature	545 °F	+10 °F
MTC~~[1]~~	-43.0~~6~~ pcm/°F	Most Negative
DTC	-2.25~~43~~ pcm/° F	Most Negative

Table 15.4-11: Control Rod Misoperation (15.4.3) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR CRA misalignment	1.284	2.638
MCHFR Single CRA withdrawal	1.284	1.614
MCHFR CRA drop	1.284	1.641
Peak LHGR CRA misalignment	21.22 kW/ft	7.10 kW/ft
Peak LHGR Single CRA withdrawal	21.22 kW/ft	7.84 kW/ft
Peak LHGR CRA drop	21.22 kW/ft	8.42 kW/ft

Table 15.4-12: Internal Flooding Sources (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Break Source	Volume [gal]
Fire suppression header	100,000
Automatic fire suppression water	54,000
Site cooling water header piping	200,000
Site cooling water heating ventilation and air conditioning support piping	40,000
Demin/utility water pipe	12,000
Main steam pipe	77,000
Feedwater pipe	24,000

Table 15.4-13: Bounding Critical Boron Concentrations and Boron Reactivity Coefficients (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Operation Mode	Critical Boron Concentration (ppm)	Boron Reactivity Coefficient (pcm/ppm)
Mode 1, 100% power	1400	-10
Mode 1, hot zero power	1800	-10
Mode 2	600	-11
Mode 3	650	-12.5
Mode 4 and 5	1800	-11.5

RAI 15.04.06-1

Table 15.4-14: Mode 1, Hot Full Power Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Dilution rate (gpm)	5	25 (1 pump)	50 (2 pumps)
Reactivity insertion rate (~~complete mixing~~perfect mixing model) pcm/second	0.0997	0.4985	0.997
Initial reactivity insertion rate (dilution front model) pcm/second	2.97	14.83	29.66
Range of reactivity insertion rates analyzed in uncontrolled control rod assembly withdrawal (~~Section 15.4.1~~Section 15.4.2) pcm/sec	~~0.2~~0.05 to ~~21~~35		
Reactor trip/DWS isolation time (seconds)	736.26	160.5	77.6
Shutdown margin remaining (complete mixing) at the time of RX trip (pcm)	1957	1944	1912

RAI 15.04.06-1

Table 15.4-15: Mode 1, Hot Zero Power Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Dilution rate (gpm)	5	25 (1 pump)	50 (2 pumps)
Initial reactivity step (pcm)	141.29	684.95	N/A[1]
Initial reactivity insertion rate (Dilution Front model) pcm/second	3.47	17.33	
Reactivity insertion rate (Perfect Mixing Model) pcm/second	0.12	0.59	
Duration of the reactivity insertion rate for each wave (seconds)	40.8	39.51	
Range of reactivity insertion rates assumed in Section 15.4.1 (pcm/sec)	0.00067 to 35		
Reactor Trip/DWS Isolation Time From Initiation of Dilution (minutes)	34.8	33.8	
Shutdown Margin Remaining (Dilution Front Model) at the time of RX Trip (pcm)	1,759.5	697.2	

[1]Two pump operation not allowed for power levels less than 50% power.

Table 15.4-16: Mode 2 Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Dilution rate[1] (gpm)	5	25 (1 pump)	50 (2 pumps)
Initial wave reactivity step (pcm)	67.83	328.81	N/A
Time to loss of shutdown margin, minutes	652.0	124.0	
Time of DWS isolation, (minutes)	427.0	88.0	
Shutdown margin remaining at DWS isolation trip (pcm)	608.5	517.4	

[1]Note that two pump operation (50 gpm) is prohibited in this mode of operation

Table 15.4-17: Not Used~~Results for Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position~~

~~Parameter~~	~~Limiting Value~~	~~Design Limit~~
~~Minimum Critical Heat Flux Ratio~~	~~1.727~~	~~1.262~~
~~Linear Heat Generation Rate~~	~~7.34 kw/ft~~	~~21.22 kw/ft~~

Table 15.4-18: Mode 3 Results (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Dilution rate[1] (gpm)	5	25 (1 pump)	50 (2 pumps)[1]
Initial wave reactivity step (pcm)	77.83	377.61	N/A
Time to loss of shutdown margin, minutes	558.0	106.0	
Time of DWS isolation, (minutes)	371.0	70.0	
Shutdown margin remaining at DWS isolation trip (pcm)	592.5	612.7	

[1]Note that two pump operation (50 gpm) is prohibited in this mode of operation

Table 15.4-19: Mode 5 Results
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Assumed initial mixing volume (ft^3)	143,652
Time to loss of shutdown margin (minutes)[1]	234
Time to loss of shutdown margin (hours)[1]	3.91
Total dilution volume to reduce shutdown margin to zero (gallons)	234,430

Note 1: Assuming a 1000 gpm dilution.

Table 15.4-20: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR	1.284	2.410
Peak LHGR	21.22 kW/ft	7.34 kW/ft

Table 15.4-21: Sequence of Events - Maximum RCS Pressure Case (15.4.8 Rod Ejection Accident)

Event	Time [s]
Rod ejection begins	0
High RCS pressure limit reached	60
Reactor trip actuates on high RCS pressure signal	62
Scram complete	64
Maximum RCS pressure reached	67

Table 15.4-22: REA Maximum Power Pulses and Reactivity Insertions

Time in Cycle	Initial Power (%)	Maximum Power (%)	Maximum Reactivity Insertion ($)
BOC	100	113	0.119
BOC	80	137	0.427
BOC	70	178	0.614
BOC	50	133	0.629
BOC	25	76	0.658
BOC	0	7	0.570
MOC	100	117	0.150
MOC	80	157	0.507
MOC	70	240	0.739
MOC	50	186	0.739
MOC	25	110	0.771
MOC	0	18	0.696
EOC	100	127	0.222
EOC	80	262	0.717
EOC	70	614	0.938
EOC	60	649	0.965
EOC	55	661	0.977
EOC	50	649	0.984
EOC	45	642	0.992
EOC	25	521	1.008
EOC	10	195	0.967
EOC	0	75	1.048

Table 15.4-23: REA Fuel Temperatures and Enthalpies for Limiting S3K Cases

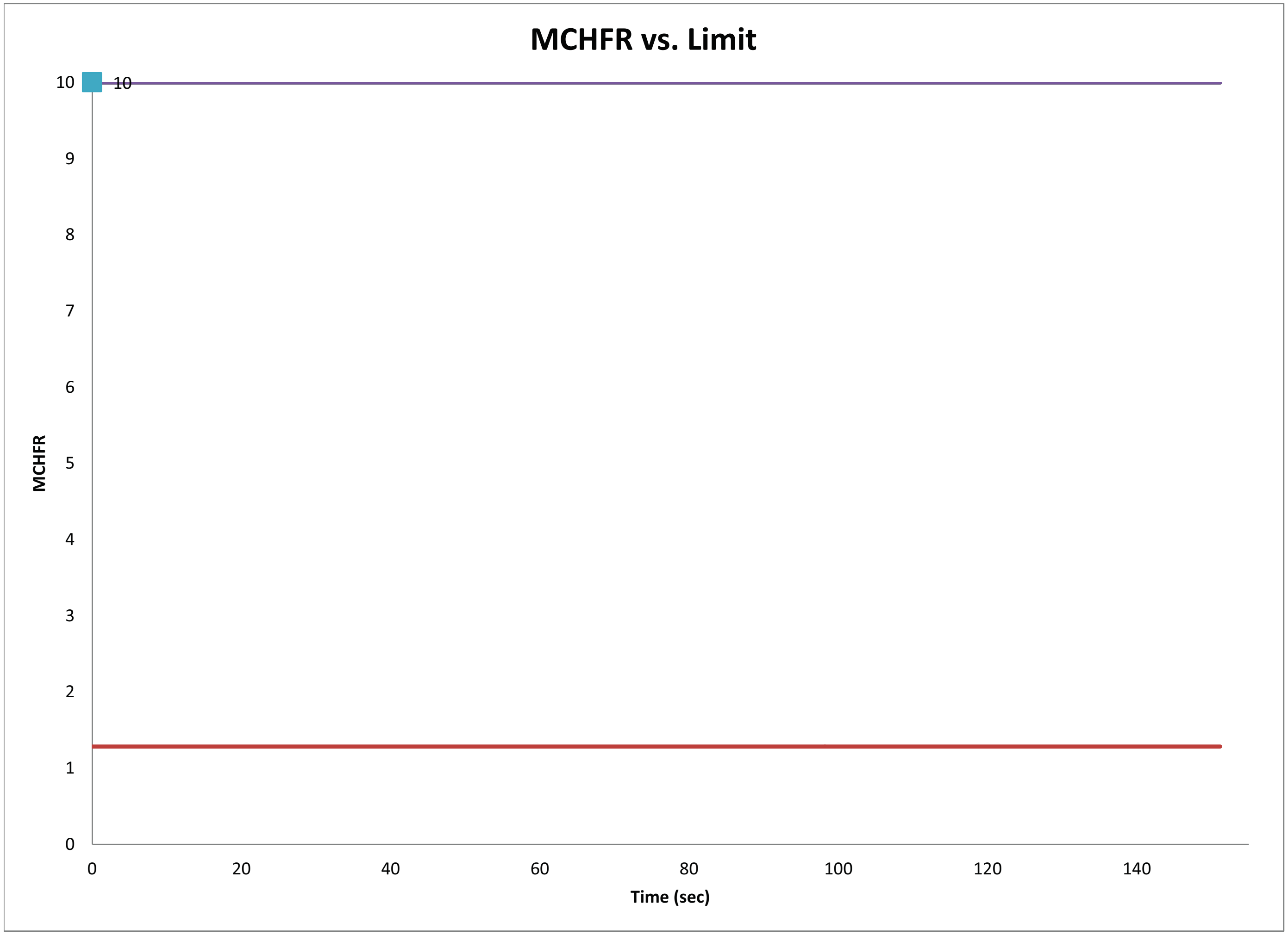
Time in Cycle[1]	Initial Power (%)	Peak Fuel Temperature (°F)	Delta Radially Averaged Fuel Enthalpy (cal/g)	Net Radially Averaged Fuel Enthalpy
BOC	25	1328	17.3	51.6
BOC	50	1813	24.6	70.5
BOC	70	2141	28.7	83.2
BOC	80	2162	26.0	84.0
MOC	25	1350	18.3	52.5
MOC	50	1798	24.3	69.9
MOC	70	2097	27.5	81.5
MOC	80	2118	24.9	82.3
EOC	0	890	18.1	34.6
EOC	10	1106	18.1	43.0
EOC	25	1350	19.4	52.5
EOC	45	1676	23.5	65.1
EOC	50	1730	23.7	67.2
EOC	55	1802	24.6	70.1
EOC	60	1862	25.0	72.4
EOC	70	1978	25.9	76.9
EOC	80	2053	25.4	79.8
EOC	100	1920	18.5	74.6

[1] The results for the HZP and full power cases for BOC and MOC conditions are covered by the analysis, but the magnitudes of these power pulses are not high enough to trip the reactor and are non-limiting, relative to the other cases.

Table 15.4-24: Spectrum of Rod Ejection Accidents (15.4.8) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Peak radial average fuel enthalpy at zero power	100 cal/gm	34.6 cal/gm
Change in radial average fuel enthalpy	75 cal/gm	28.7 cal/gm
Peak radial average fuel enthalpy	100 cal/gm	84 cal/gm
Maximum RCS Pressure	2310 psia	2076 psia
Peak Fuel Temperature	4791°F	2162°F
MCHFR	1.284	2.477

Figure 15.4-5: Critical Heat Flux Ratio
(15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low~~er~~ Power Condition)



RAI 15.04.02-1, RAI 15.04.02-2

Figure 15.4-11: ~~CHFR~~Critical Heat Flux Ratio
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

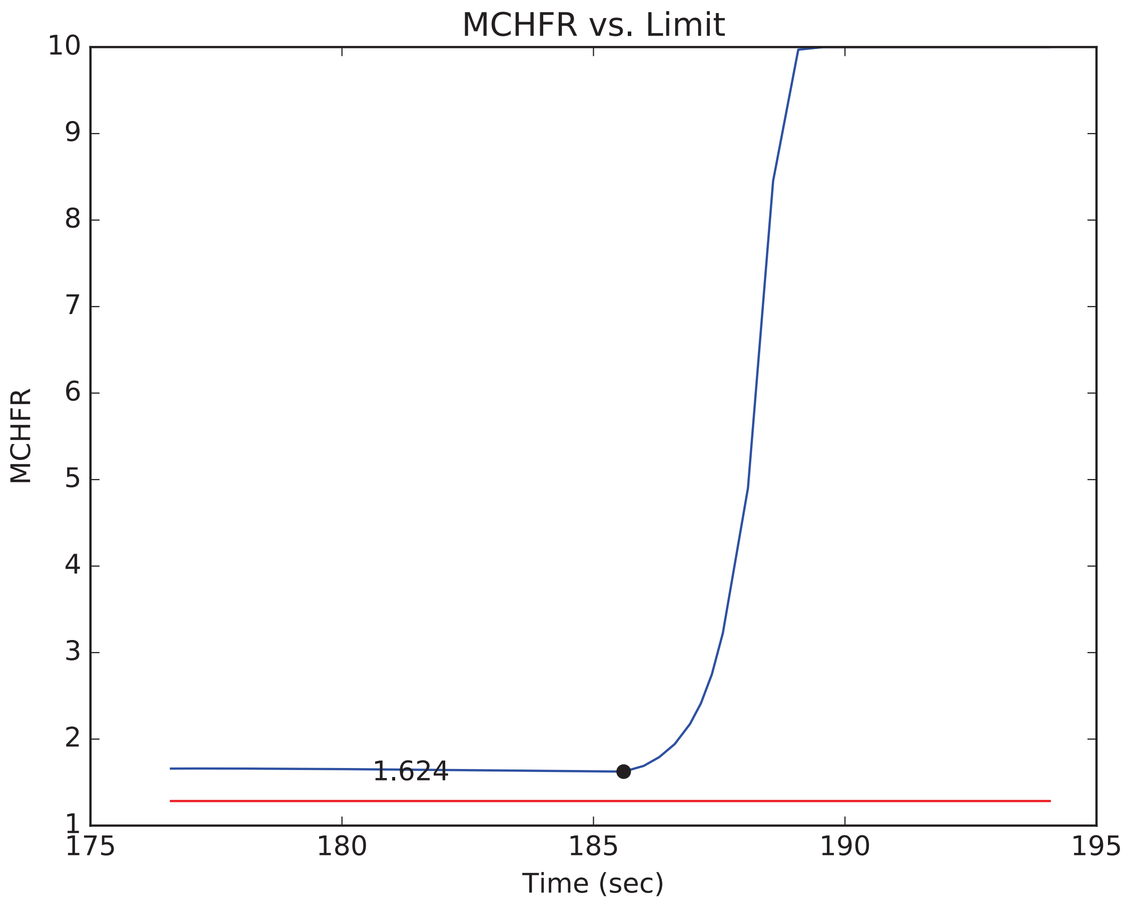
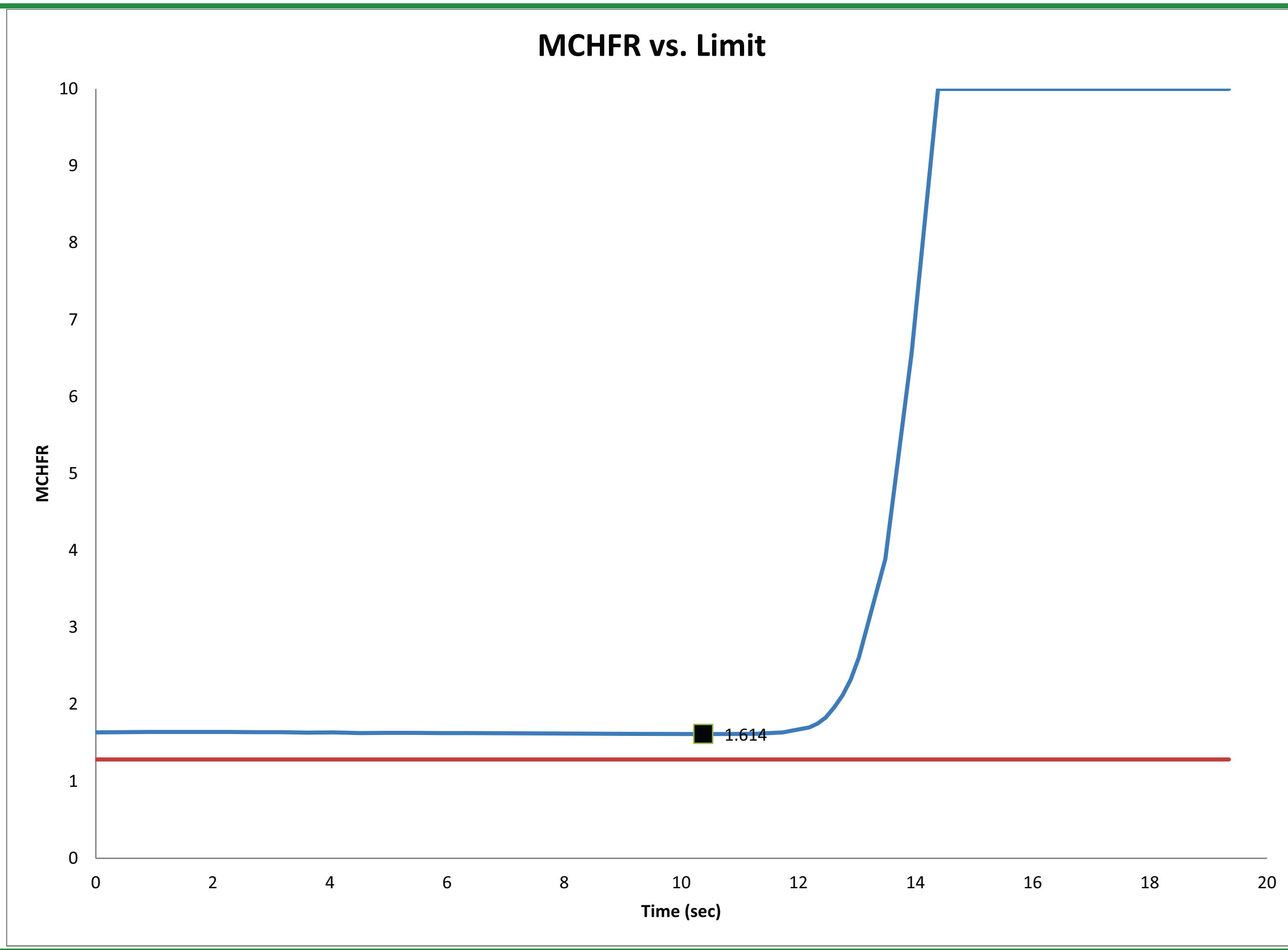


Figure 15.4-32: ~~CHFR (15.4.8 Rod Ejection Accident)~~Not Used

RAI 15.04.03-1

Figure 15.4-35: Critical Heat Flux Ratio (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)



RAI 15.04.03-1

Figure 15.4-36: Critical Heat Flux Ratio (15.4.3 Control Rod Misoperation, Control Rod Assembly drop)

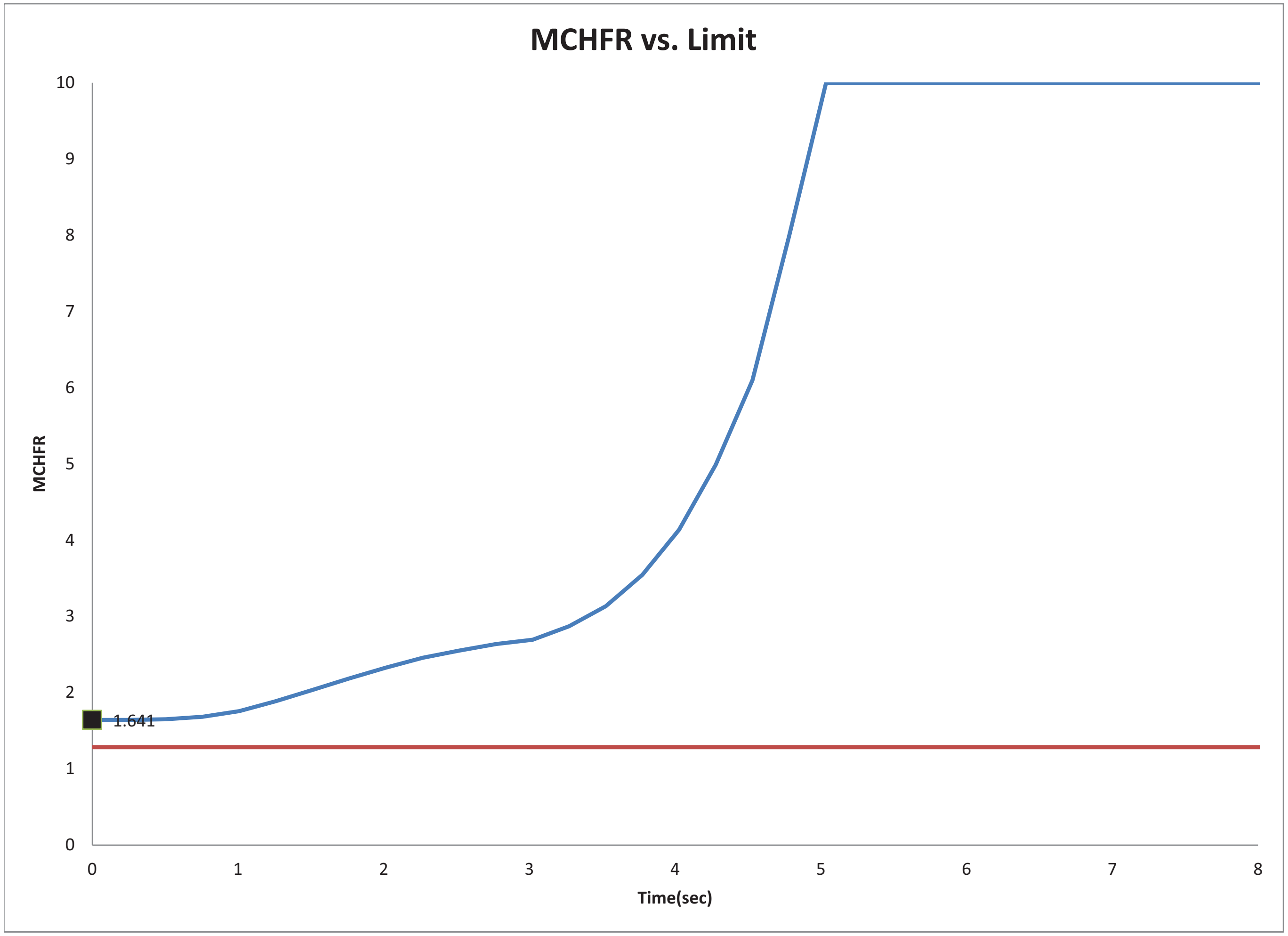


Figure 15.4-37: CHFR (15.4.8 Rod Ejection Accident)

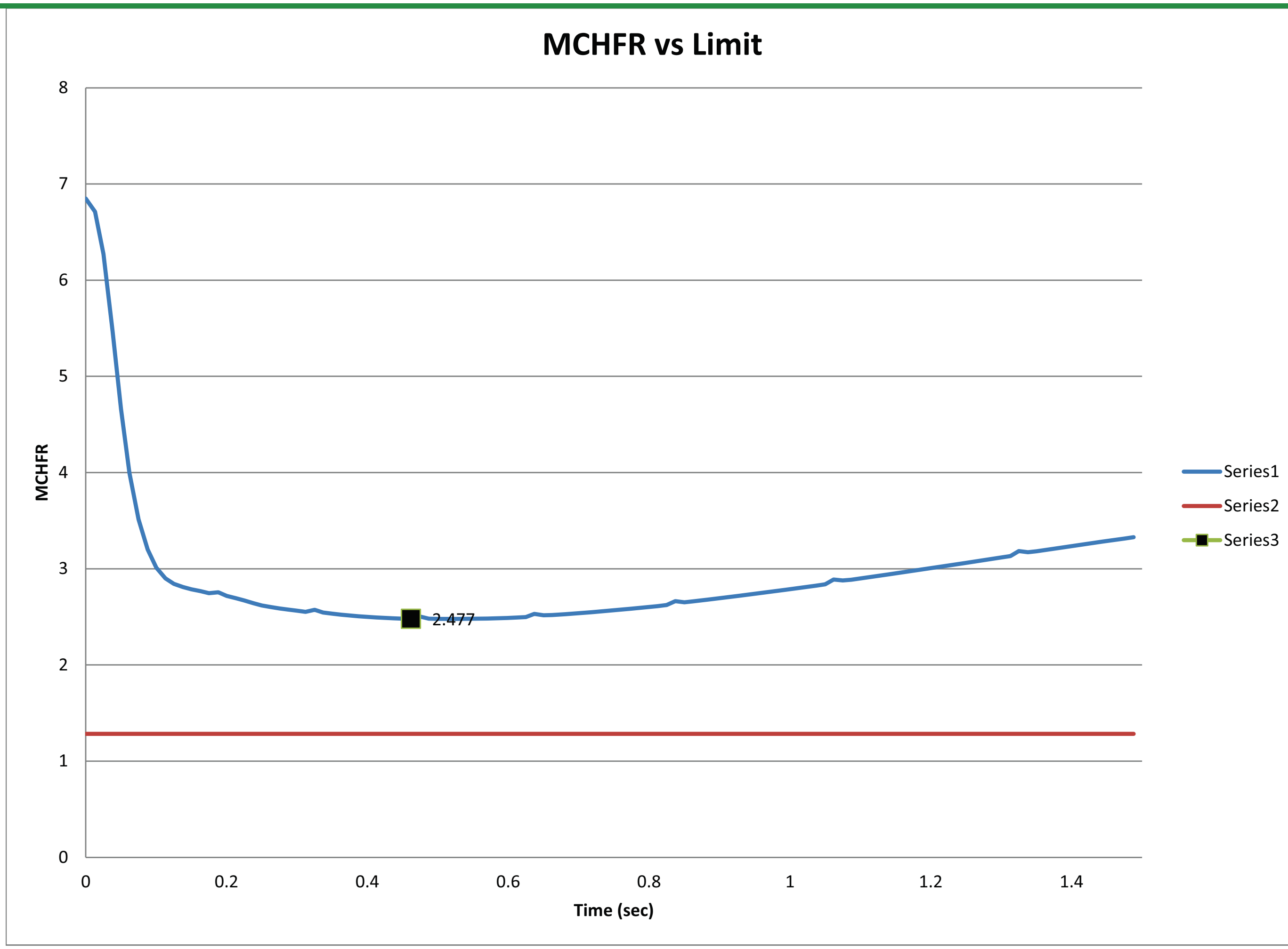
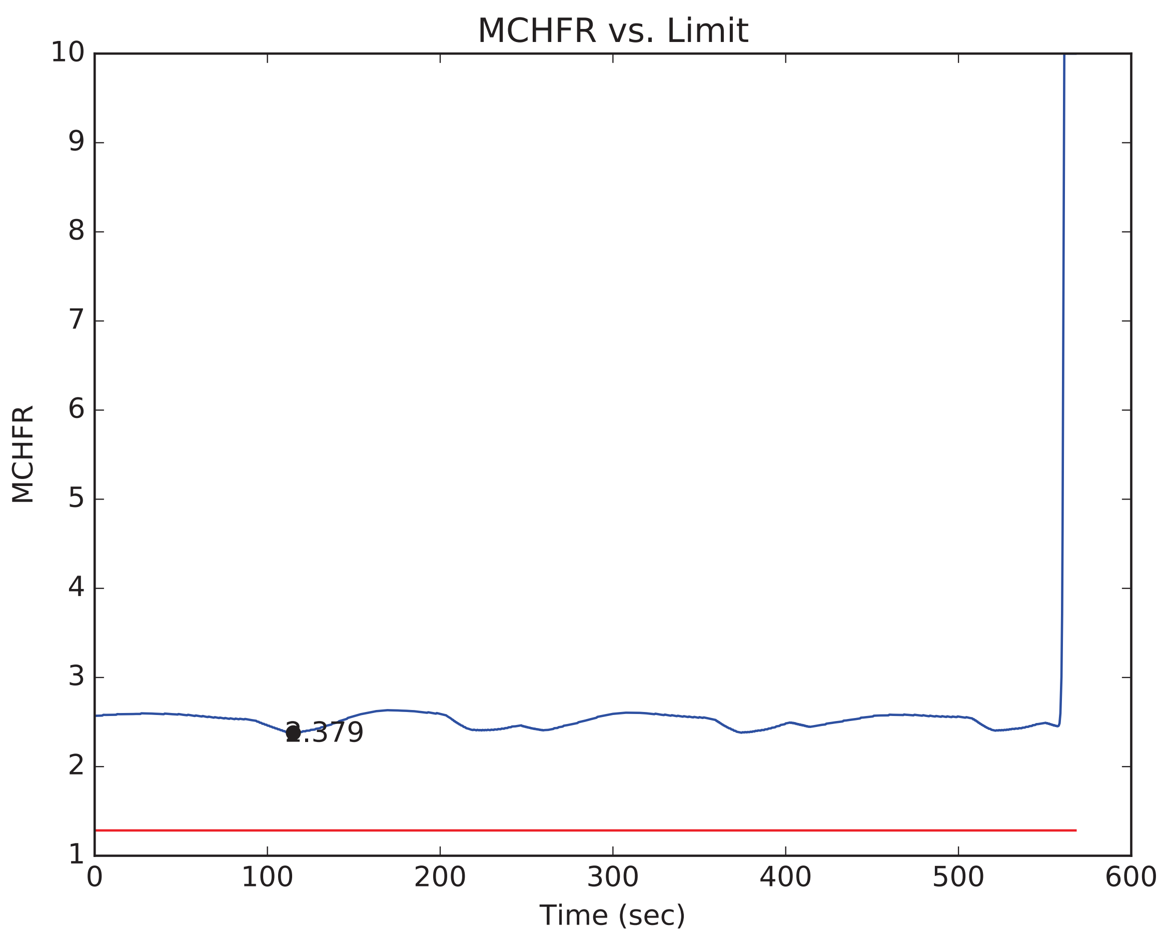


Table 15.5-4: Summary of Results CVCS Malfunction

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure (No PZR Spray)	2310 psia	2130 psia
Maximum SG Pressure (PZR Spray Available)	2310 psia	1418 psia
MCHFR (PZR Spray Available)	~~1.262~~1.284	~~1.811~~2.379

Figure 15.5-10: Minimum Critical Heat Flux Ratio - Increase in RCS Inventory (PZR Spray Available)



15.6-2 NuScale Power, LLC, Topical Report, "NuScale Power Critical Heat Flux Correlations ~~NSP2~~," TR-0116-21012, Rev. 1~~0~~.

15.6-3 NuScale Power, LLC, Topical Report, "Subchannel Analysis Methodology," TR-0915-17564, Rev. 1~~0~~.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODE 1 the critical heat flux ratio shall be maintained ≥ the 95/95 critical heat flux ratio criterion for the critical heat flux correlation(s) specified in Section 5.6.3.~~at or above the following CHF correlation safety limits:~~

~~Correlation~~	~~Safety Limit~~
~~NSP2~~	~~1.262~~
~~Extended Hench-Levy~~	~~1.122~~
~~Griffith-Zuber~~	~~1.37~~

2.1.1.2 ~~In MODE 1 the peak Linear Heat Rate shall be maintained ≤ 21.22 kW/ft.~~ In MODE 1 the peak fuel centerline temperature shall be maintained ≤ 4901 - (1.37E-3 x Burnup, MWD/MTU) °F.

2.1.2 RCS Pressure SL

In MODES 1, 2, and 3 pressurizer pressure shall be maintained ≤ 2285 psia.

2.2 Safety Limit Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 2 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1, restore compliance and be in MODE 2 within 1 hour.

2.2.2.2 In MODE 2 or 3, restore compliance within 5 minutes.

5.6 Reporting Requirements

5.6.3 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1, "Reactor Core Safety Limits";

3.1.1, "SHUTDOWN MARGIN (SDM)";

3.1.3, "Moderator Temperature Coefficient (MTC)";

3.1.4, "Rod Group Alignment Limits";

3.1.5, "Shutdown Group Insertion Limits";

3.1.6, "Regulating Group Insertion Limits";

3.1.8, "PHYSICS TESTS Exceptions";

3.1.9, "Boron Dilution Control";

3.2.1, "Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)~~ ~~";

3.2.2, "AXIAL OFFSET (AO)";

3.4.1, "RCS Pressure~~ and Temperature~~, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits"; and

3.5.3, "Ultimate Heat Sink".

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ~~Reload Safety Evaluation Methodology (later).~~[List of NRC-approved Topical Reports that are used to determine the core operating limits listed in 5.6.3.a above.]

~~2. TR0616-48793-NP, Rev 0, Nuclear Analysis Codes and Methods Qualification~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Passive Core Cooling Systems limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 CHF criterion) that the hot fuel rod in the core does not experience CHF; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Module Protection System (MPS) setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, ~~RCS Flow, ΔI,~~ and THERMAL POWER level that would result in a critical heat flux ratio (CHFR) of less than the CHFR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the MPS and the decay heat removal system.

The SLs represent a design requirement for establishing the MPS Trip System setpoints (Ref. 2). LCO 3.4.1, "RCS ~~Pressure Temperature~~ Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits," or the assumed initial conditions of the safety analyses (as indicated in FSAR ~~Section 7.2~~Chapter 15, Ref. 3~~2~~) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 CHF criterion) that the hot fuel rod in the core does not experience CHF; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various MPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). ~~The NSP2 correlation limit is used to evaluate non-LOCA transients as described in the FSAR (Ref. 3). The Extended Hench-Levy and Griffith-Zuber correlation limits are utilized to evaluate other transients that occur with high and low RCS flow rates respectively as also described in the~~

BASES

SAFETY LIMITS (continued)

~~FSAR Chapter 15 (Ref. 3).~~ To ensure that the MPS precludes violation of the above criteria, additional criteria are applied to the low pressurizer pressure reactor trip functions. That is, it must be demonstrated that the core exit quality is within the limits defined by the CHF~~R~~ correlation and that the low pressurizer pressure reactor trip protection functions continues to provide protection if core exit streams approach saturation temperature. Appropriate functioning of the MPS ensures that for variations in the THERMAL POWER, RCS Pressure and~~,~~ RCS ~~cold~~ temperature~~, and RCS flow rate that~~ the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODE 1 because this is the only MODE in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODE 1 to ensure operation within the reactor core SLs. The decay heat removal system and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function which forces the unit into MODE 2. Setpoints for the reactor trip functions are described~~specified~~ in LCO 3.3.1, "Module Protection System (MPS) Instrumentation~~.~~" and specified in the [Technical Requirements Manual]. In MODES 2, 3, 4, and 5, applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 2 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. FSAR ~~Section 7.2, "Reactor Trip."~~Chapter 7, "Instrumentation and Controls."

3. FSAR Chapter 15, "Transient and Accident Analyses."