RAIO-0518-60110

May 21, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 394 (eRAI No. 9407) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 394 (eRAI No. 9407)," dated March 21, 2018

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 9407:

- 15.02.01-1
- 15.02.01-2
- 15.02.01-3
- 15.02.01-4
- 15.02.01-5
- 15.02.01-6
- 15.02.01-7
- 15.02.01-8
- 15.02.01-9
- 15.02.01-10
- 15.02.01-11
- 15.02.01-12

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9407

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9407

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-1

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the general design criteria (GDC) in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A. Design-Specific Review Standard (DSRS) Section 15.2.1-15.2.5, "Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve; and Steam Pressure Regulator Failure (Closed)," provides guidance for meeting the requirements of GDC 10, "Reactor design"; 13, "Instrumentation and control"; 15, "Reactor coolant system design"; 17, "Electric power systems"; and 26, "Reactivity control system redundancy and capability." To meet these requirements, DSRS Section 15.2.1-15.2.5 states that one of the basic objectives of the review of the initiating events covered under the DSRS section is to identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting. Furthermore, DSRS Section 15.2.1-15.2.5 guides the reviewer to evaluate the applicant's justification if the applicant's technical submittal states that one of these transients is not as limiting as other similar transients.

FSAR Tier 2, Sections 15.2.1 through 15.2.3 describe the loss of external load (LOEL), turbine trip (TT), and loss of condenser vacuum (LOCV) events and include a single evaluation for all three events, presumably because FSAR Section 15.2 states that these events produced essentially identical responses for the acceptance criteria (reactor coolant system [RCS] pressure, steam generator [SG] pressure, and minimum critical heat flux ratio [MCHFR]). FSAR Sections 15.2.1 through 15.2.3 do not indicate which event (LOEL, TT, or LOCV) is most limiting for each acceptance criterion. The staff notes that phenomenological differences between the LOEL, TT, and LOCV events lead to some differences in the event timing and severity. For example, the staff would expect that the immediate loss of feedwater that occurs with a LOCV would cause a higher and faster peak RCS pressure than the LOEL and TT events.

For consistency with DSRS Section 15.2.1-15.2.5, and to enable the staff to conclude based upon docketed material that the most limiting cases have been accurately identified, please update the FSAR to indicate the events (LOEL, TT, or LOCV) that led to the limiting analysis sequences of events and results in FSAR Tier 2, Tables 15.2-4 through 15.2-7.

NuScale Response:

The most limiting conditions for the loss of external load (LOEL), turbine trip (TT), and loss of condenser vacuum (LOCV) transients are summarized in Table 1. The LOCV is most limiting for reactor coolant system (RCS) peak pressure and minimum critical heat flux ratio (MCHFR) while the TT is limiting for the steam generator system (SGS) peak pressure. The FSAR is modified to clarify the limiting conditions for the LOEV, TT and LOCV transients.

Table 1 Summary of Limiting Conditions for the LOCV, TT and LOCV Transients

	Transient		
	LOCV	**TT**	**LOCV**
Condition	Peak RCS pressure	Peak SGS pressure	MCHFR
Failure that initiates event	TSV closure & loss of FW flow, coincident loss of AC power, failure of one FWIV to close	TSV closure, failure of one FWIV to close	TSV closure & loss of FW flow
High analytical limit and when reached	High PZR pressure 105 sec	High steam line pressure 107 sec	High PZR pressure 110 sec
RSV Lift Time	110 sec	none	118 sec
Limiting Condition and When Reached	2158 psia 110 sec	1474 psia 182 sec	2.579 113 sec

Impact on DCA:

FSAR Sections 15.2.1.3.3, 15.2.2.3.3, 15.2.3.3.3 and Table 15.2-7 have been revised as described in the response above and as shown in the markup provided in this response.

15.2 Decrease in Heat Removal by the Secondary Side

This section addresses design basis events associated with a potential unplanned decrease in primary system heat removal through the steam generators (SGs). The decrease in heat removal causes the primary side temperature and pressure to rise and the pressurizer level to increase. SG pressure also increases. There are eight events that are defined for this category by the NuScale DSRS. One unique NuScale Power Module (NPM) event has been identified for this event type involving the decay heat removal system (DHRS) (Section 15.2.9).

The FSAR subsections are as follows:

- Section 15.2.1 - Loss of External Load
- Section 15.2.2 - Turbine Trip
- Section 15.2.3 - Loss of Condenser Vacuum
- Section 15.2.4 - Closure of Main Steam Isolation Valve (MSIV)
- Section 15.2.5 - Steam Pressure Regulator Failure
- Section 15.2.6 - Loss of Non-Emergency AC Power to the Station Auxiliaries
- Section 15.2.7 - Loss of Normal Feedwater Flow
- Section 15.2.8 - Feedwater System Pipe Breaks Inside and Outside of Containment
- Section 15.2.9 - Inadvertent Operation of the Decay Heat Removal System

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The evaluation of the first three events in Section 15.2, Loss of External Load, Turbine Trip and Loss of Condenser Vacuum (LOEL, TT and LOCV) produced essentially identical responses for the primary and secondary system and minimum critical heat flux ratio (MCHFR). Therefore, a single set of figures based on an enveloping analysis of these events is provided to show the bounding cases for these events. The Inadvertent Closure of a Main Steam Isolation Valve (MSIV) figures are presented separately. Section 15.2.5, Steam Pressure Regulator Failure, is not applicable to the NPM and is kept primarily to maintain Section organization and numbering. Figures for Section 15.2.6 through Section 15.2.9 are also presented individually for each section.

The NuScale DSRS states that for new applications the loss of offsite power (LOOP) must be considered in addition to a single active failure. For NuScale, a LOOP is determined by a loss of AC power at the ELVS level (480V). The highly reliable DC power system (EDSS) and the non-safety DC power (EDNS) are not safety-related and their loss of function to provide power is considered as a possible concurrent event with the loss of ELVS. If the EDSS system fails, a reactor trip and containment isolation will occur and the emergency core cooling system (ECCS) valves will open when RCS pressure drops below the ECCS inadvertent actuation block (IAB) threshold. The timing of ECCS operation is after the time period of concern for evaluation of maximum RCS pressure, maximum steam pressure or MCHFR for decrease in heat removal events presented in this section. Therefore, the potential for ECCS operation is addressed in Section 15.0.5, Long Term Decay and Residual Heat Removal.

While there are various integrated, automatic control systems that are expected to keep the reactor at power when a turbine trip or loss of external load occurs, sensors, signal processing

(Figure 15.2-5) drops initially due to the reactor trip and is reestablished as DHRS flow is established (Figure 15.2-6). Steam generator pressure for the peak SG pressure case is presented in Figure 15.2-8.

LOEL results in increased temperature in the RCS which could potentially challenge fuel parameters. Although RCS fluid and fuel temperatures increase, the core remains covered throughout the event, such that the MCHFR limits are not challenged. The limiting MCHFR is demonstrated in Figure 5.2-9.

15.2.1.4 Radiological Consequences

The radiological consequences of an LOEL event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.1.5 Conclusions

RAI 15.02.01-1

The six DSRS acceptance criteria for this AOO are met for the enveloping analysis which includes: Loss of External Load, Turbine Trip and Loss of Condenser Vacuum cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

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

   - The limiting RCS pressure for this event, shown in Table 15.2-7, is below 110% of the design value for the reactor coolant system.
   - The limiting steam generator pressure, shown in Table 15.2-7, is below 110% of the design value for the main steam system up to the MSIVs.

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

   - The MCHFR for this event, shown in Table 15.2-7, is above the 95/95 limit.

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

   - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met. The LOEL event does not lead to a more serious event.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the 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 GDCs 10 and 15.

Table 15.2-1 for the maximum RCS pressure event, Table 15.2-2 for the maximum SG pressure event and Table 15.2-3 for the limiting MCHFR event.

15.2.2.3.3 Results

RAI 15.02.01-1

The results for the turbine trip event are the same as those presented in Section 15.2.1.3.3 for the LOEL event. However, as shown in Table 15.2-7, the TT event is the limiting event for maximum SGS pressure, but the TT event does not challenge the limits for RCS pressure, SGS pressure or MCHFR.

15.2.2.4 Radiological Consequences

The radiological consequences of a turbine trip event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.2.5 Conclusions

The regulatory acceptance criteria for an AOO are met for the limiting turbine trip event. The results show significant margin between the system response and the design limits. The MCHFR for the limiting turbine trip event meets the acceptance criterion discussed in Section 4.4.4, demonstrating that this event does not result in any fuel damage. The high steam line pressure or high pressurizer pressure reactor trips and subsequent actuation of the DHRS terminate this transient by inserting the control rods and removing the decay heat, resulting in a safe stable condition following the event. No operator actions were credited for mitigation of the turbine trip event. A turbine trip does not lead to a more serious event.

The analysis for the turbine trip shows that the NPM design is acceptable with respect to transients resulting in unplanned decreases in heat removal by the secondary system, transients expected with moderate frequency, and transients where the predicted response meets the acceptance criteria to maintain RCS and secondary piping pressure below 110% of the design value and MCHFR is maintained above the design limit. Details of the DSRS acceptance criteria are discussed in Section 15.2.1 for the LOEL Event but are also applicable for the turbine trip event. The numerical values for the acceptance criteria are listed in Table 15.2-7.

15.2.3 Loss of Condenser Vacuum

15.2.3.1 Identification of Causes and Event Description

A loss of condenser vacuum (LOCV) event involves a disturbance that results in an increase in condenser pressure due to air inleakage or a reduction in cooling to the condenser. For the NPM design, the effect of a loss of condenser vacuum results in a trip of the turbine generator and a loss of feedwater flow, which causes the primary side temperature and pressure to increase because energy is not being removed through the steam generators to the condenser. The reactor trips on high pressurizer pressure or high steam line pressure, reducing power to decay heat levels. The Decay Heat Removal System (DHRS) actuates to transfer decay heat to the reactor pool.

15.2.3.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator density and fuel temperature is taken from the Beginning of Cycle (BOC). The reactivity coefficients for moderator density are least negative at BOC. Thus, they are conservative for undercooling events such as turbine trip, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102%. The most reactive rod is assumed to remain out of the core,along with a delay of 2 seconds between the reactor trip signal and scram initiation. The most limiting combination of reactivity coefficients for moderator density and fuel temperature is applied. Instrument inaccuracy is accounted for by examining the sensitivity to the setpoints over the given margin of error. The key parameters are listed in Table 15.2-1 for the maximum RCS pressure event, Table 15.2-2 for the maximum SG pressure event and Table 15.2-3 for the limiting MCHFR event.

15.2.3.3.3 Results

RAI 15.02.01-1

The results for the LOCV event are the same as those presented in Section 15.2.1.3.3 for the LOEL event. However, as shown in Table 15.2-7, the LOCV event is the limiting event for the maximum RSC pressure and MCHFR, but the LOCV event does not challenge the limits on RCS pressure, SGS pressure or MCHFR.

15.2.3.4 Radiological Consequences

The radiological consequences of a LOCV event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.3.5 Conclusions

The regulatory acceptance criteria for an AOO are met for the limiting LOCV event. The results show significant margin between the system response and the design limits. The MCHFR for the limiting LOCV event meets the acceptance criterion discussed in Section 4.4.4, demonstrating that this event does not result in any fuel damage. The high steam line pressure or high pressurizer pressure reactor trips and subsequent actuation of the DHRS terminate this transient by inserting the control rods and removing the decay heat, resulting in a safe stable condition following the event. No operator actions were credited for mitigation of the LOCV event. An LOCV event does not lead to a more serious event.

The analysis for LOCV shows that the NPM design is acceptable with respect to transients resulting in unplanned decreases in heat removal by the secondary system, transients expected with moderate frequency, and transients where the predicted response meets the acceptance criteria to maintain RCS and secondary piping pressure below 110% of the design value and MCHFR is maintained above the design limit.

RAI 15.02.01-1

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

Acceptance Criteria	Transient	Limit	Analysis Value
Maximum RCS Pressure	LOCV	2310 psia	2158 psia
Maximum SG Pressure	TT	2310 psia	1474 psia
MCHFR	LOCV	1.284	2.579

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-2

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26, and in doing so, guides the staff to review the extent to which credit is taken for the functioning of normally operating plant systems. Performance of non-safety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY 77-439, SECY 94-084, and RG 1.206.

FSAR Tier 2, Section 15.2.1, “Loss of External Load,” states that plant control systems perform as designed with allowances for instrument inaccuracy unless stated otherwise. FSAR Tier 2, Section 15.2.1 describes that the turbine bypass system is assumed not to perform as designed; however, there is no other discussion of credit for plant control systems (PCSs) for the LOEL, TT, and LOCV events. The staff would expect similar assumptions to be made for these events as are described in FSAR Section 15.2.4, “Closure of Main Steam Isolation Valve(s),” which discusses the functionality of the chemical and volume control system and pressurizer spray system. Please confirm whether any other non-safety-related PCSs, such as the pressurizer pressure or level control functions of the module control system, are assumed to operate during the LOEL, TT, and LOCV transients. Update the FSAR as appropriate.

NuScale Response:

For the loss of external load (LOEL), turbine trip (TT) and loss of condenser vacuum (LOCV) transients, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most plant control systems (PCSs) are disabled because their operation is beneficial with respect to the consequences for either the LOEL, TT or LOCV events. The disabled PCSs are: reactor coolant system (RCS) temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOEL, TT, or LOCV transients.

FSAR Sections 15.2.1.2, 15.2.2.2 and 15.2.3.2 are revised to explain the operation or lack thereof of the PCSs for the LOEL, TT and LOCV transients, respectively.

Impact on DCA:

FSAR Sections 15.2.1.2, 15.2.2.2 and 15.2.3.2 have been revised as described in the response above and as shown in the markup provided in this response.

and final control elements that support these automated controls are not safety-related. Therefore, mitigating control system responses are not credited for the events in this section, but their potential adverse impact to safety functions are considered.

15.2.1 Loss of External Load

15.2.1.1 Identification of Causes and Event Description

A loss of external load (LOEL) event is initiated by an electrical disturbance that results in the loss of a significant portion, or all, of the turbine generator load, leading to a turbine trip. The turbine trip causes the primary and secondary side temperatures and pressures to increase because energy is not being removed through the steam generators to the condenser. The reactor trip signal and decay heat removal system (DHRS) actuation signal are initiated on high pressurizer pressure or high steam line pressure. The reactor trip reduces power to decay heat levels. The DHRS actuates and transfers decay heat to the reactor pool. If offsite power is lost, with a coincident loss of DC power (EDSS and EDNS), the reactor trip, DHRS actuation and main steam isolation happen concurrently.

A LOEL event is expected to occur one or more times in the life of the plant. Therefore, a LOEL event is an AOO as indicated in Table 15.0-1.

15.2.1.2 Sequence of Events and Systems Operation

The severity of a LOEL event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine control valves. However, following the valve closures, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event. Key parameters (pressurizer level, reactor power, net reactivity, RCS average temperature, RCS flowrate, DHRS flow, and RCS reactor vessel pressure) are shown in Figure 15.2-1 through Figure 15.2-7 for the peak RCS pressure case. Note that these figures show 100 seconds of steady state operation before the initiation of the event.

RAI 15.02.01-2

~~Unless stated otherwise, the plant control systems (PCSs) and the engineered safety features (ESFs) perform as designed with allowances for instrument inaccuracy. For the LOEL event, the turbine bypass system would normally actuate, allowing steam flow to bypass the turbine to the condenser and the steam generators would continue to remove heat from the reactor coolant system (RCS). However, the turbine bypass system is assumed to fail resulting in heat up of the primary and secondary plant. No operator action is credited to mitigate the effects of an LOEL event.~~Unless stated otherwise, the plant control systems (PCSs) and the engineered safety features (ESFs) perform as designed with allowances for instrument inaccuracy. For the LOEL event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for an LOEL event. The disabled PCSs are for reactor coolant system (RCS) temperature control, pressurizer pressure control,

pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOEL event.

For the limiting RCS pressure case, the module protection system (MPS) initiates a reactor trip and actuates DHRS on a high pressurizer pressure signal. In addition to opening the DHRS valves, actuating DHRS also closes the FWIVs, FW regulating valves, MSIVs and secondary MSIVs and deenergizes the pressurizer heaters. The FW regulating valves and secondary MSIVs are nonsafety-related and close in 30 seconds. These valves are credited as having redundant isolation capability in the case of the failure of the FWIVs or MSIVs to close. The limiting assumption for RCS pressure during an LOEL event is a loss of AC power, with DC power available, primarily due to the heatup caused by an immediate loss of feedwater. No single failure resulted in a more limiting RCS pressure. The RCS pressure increase is mitigated by opening one of the two redundant RSVs.

For the limiting SG pressure case, the MPS initiates a reactor trip and actuates the DHRS on high steam pressure. The limiting single failure for an LOEL for peak SG pressure is the failure of a FWIV to close with AC and DC power available. Loss of AC and DC power would initiate MPS and ESF functions earlier in the event and therefore are not limiting. If a FWIV fails to close, FW flow will be provided to the SG until the FW pumps are secured or the FW regulating valves close. The feedwater regulating valves are nonsafety-related but are credited to close within 30 seconds in the event of a failure of the safety-related FWIV. The feedwater regulation valves get a close signal on DHRS actuation or containment isolation. The valves also close on a loss of DC power (EDSS). The FWIV failure results in the highest SG peak pressure and in the worst case would result in disabling one DHRS train due to overfilling. The remaining DHRS train is adequate for heat removal. The RCS pressure does not reach the RSV actuation setpoint in this event. The peak SG pressure is shown on Figure 15.2-8.

For the limiting MCHFR case, the MPS initiates a reactor trip and actuates the DHRS on high steam pressure. The limiting assumption for MCHFR during an LOEL event is a loss of AC power with DC power available. Loss of DC power would initiate MPS and ESF functions earlier in the event and, therefore, is not limiting. No single failure resulted in a more limiting MCHFR. The RCS pressure does not reach the RSV actuation setpoint in this event. The limiting MCHFR versus time is shown in Figure 15.2-9.

For the peak RCS pressure and MCHFR cases, a loss of AC power provides the limiting results. Loss of DC power would initiate MPS functions earlier in the event and therefore is not limiting.

For the peak SG pressure case, normal AC and DC power are assumed to be available. A loss of AC or DC power is not conservative for this event because the loss of power would terminate feedwater flow and actuate MPS functions earlier in the event sequence.

The sequence of events for the LOEL events are described in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure event, and Table 15.2-6 for the limiting MCHFR event.

RAI 15.02.01-2

Unless stated otherwise, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. For the turbine trip event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for a turbine trip event. The disabled PCSs are for RCS temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of a turbine trip event.

The description for the remaining sequence of the turbine trip event is the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios. The sequence of events for the turbine trip are described in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure events and Table 15.2-6 for the limiting MCHFR event.

15.2.2.3 Thermal Hydraulic and Subchannel Analyses

15.2.2.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a turbine trip event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) 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 for a discussion of the VIPRE-01 code and evaluation model.

15.2.2.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator density and fuel temperature is taken from the Beginning of Cycle (BOC). The reactivity coefficients for moderator density are least negative at BOC. Thus, they are conservative for undercooling events such as turbine trip, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102%. The most reactive rod is assumed to remain out of the core, along with a delay of 2 seconds between the reactor trip signal and the scram initiation. The most limiting combination of reactivity coefficients for moderator density and fuel temperature is applied. Instrument inaccuracy is accounted for by examining the sensitivity to the setpoints over the given margin of error. The key parameters are listed in Table 15.2-1 for the maximum RCS pressure event, Table 15.2-2 for the maximum SG pressure event and Table 15.2-3 for the limiting MCHFR event.

15.2.3.2 Sequence of Events and Systems Operation

The severity of a LOCV event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine stop valves (TSVs). Following the valve closures, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event.

RAI 15.02.01-2

Unless stated otherwise, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. For the LOCV event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for an LOCV event. The disabled PCSs are for RCS temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOCV event.

The description for the remaining sequence of the LOCV event are the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios. The sequence of events for the LOCV are presented in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure events and Table 15.2-6 for the limiting MCHFR event.

15.2.3.3 Thermal Hydraulic and Subchannel Analyses

15.2.3.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a LOCV event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) 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 for a discussion of the VIPRE-01 code and evaluation model.

15.2.3.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator density and fuel temperature is taken from the Beginning of Cycle (BOC). The reactivity coefficients for moderator density are least negative at BOC. Thus, they are conservative for undercooling events such as turbine trip, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-3

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26. To meet these requirements, DSRS Section 15.2.1-15.2.5 states that the most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

FSAR Tier 2, Section 15.2.1 states that "no single failure resulted in a more limiting RCS pressure." Based on its audit of EC-0000-1997, Revision 0, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum," and ECN-0000-4862, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum – Impact Analysis" (which are the calculations of record for the analyses presented in FSAR Sections 15.2.1, 15.2.2, and 15.2.3), the staff notes that single failures were not examined for the LOEL cases because of their similarity to TT cases, which did examine single failures. The staff further notes that a failure of a feedwater isolation valve (FWIV) to close for a TT event was slightly more limiting than assuming no single failure. The same is true for the LOCV event. Because the peak RCS pressure case for the LOEL, TT, and LOCV events assumes a failure of a FWIV, the FSAR should reflect that assumption. Please update the FSAR as appropriate.

NuScale Response:

As indicated in the response to RAI 9407 question 15.02.01-1, the failure of one feedwater isolation valve (FWIV) to close is assumed for the limiting reactor coolant system (RCS) peak pressure during the loss of condenser vacuum (LOCV) transient and for the limiting steam generator system (SGS) peak pressure during the turbine trip (TT) event. In both cases, this assumption increased the peak pressure by a small fraction of a psi.

FSAR Sections 15.2.1.2, 15.2.2.2 and 15.2.3.2 are updated to clarify this assumed single failure for the LOCV and TT.

Impact on DCA:

FSAR Sections 15.2.1.2, 15.2.2.2 and 15.2.3.2 have been revised as described in the response above and as shown in the markup provided in this response.

pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOEL event.

RAI 15.02.01-3

For the limiting RCS pressure case, the module protection system (MPS) initiates a reactor trip and actuates DHRS on a high pressurizer pressure signal. In addition to opening the DHRS valves, actuating DHRS also closes the FWIVs, FW regulating valves, MSIVs and secondary MSIVs and deenergizes the pressurizer heaters. The FW regulating valves and secondary MSIVs are nonsafety-related and close in 30 seconds. These valves are credited as having redundant isolation capability in the case of the failure of the FWIVs or MSIVs to close. The limiting assumption for RCS pressure during an LOEL event is a loss of AC power, with DC power available, primarily due to the heatup caused by an immediate loss of feedwater. No single failure resulted in a more limiting RCS pressure for the LOEL event. The RCS pressure increase is mitigated by opening one of the two redundant RSVs.

RAI 15.02.01-3

For the limiting SG pressure case, the MPS initiates a reactor trip and actuates the DHRS on high steam pressure. The limiting single failure for an LOEL for peak SG pressure is the failure of a FWIV to close with AC and DC power available. Loss of AC and DC power would initiate MPS and ESF functions earlier in the event and therefore are not limiting. If a FWIV fails to close, FW flow will be provided to the SG until the FW pumps are secured or the FW regulating valves close. The feedwater regulating valves are nonsafety-related but are credited to close within 30 seconds in the event of a failure of the safety-related FWIV. The feedwater regulation valves get a close signal on DHRS actuation or containment isolation. The valves also close on a loss of DC power (EDSS). The FWIV failure results in the highest SG peak pressure and in the worst case would result in disabling one DHRS train due to overfilling. The remaining DHRS train is adequate for heat removal. The RCS pressure does not reach the RSV actuation setpoint ~~in this event~~for the LOEL event. The peak SG pressure is shown on Figure 15.2-8.

RAI 15.02.01-3

For the limiting MCHFR case, the MPS initiates a reactor trip and actuates the DHRS on high steam pressure. The limiting assumption for MCHFR during an LOEL event is a loss of AC power with DC power available. Loss of DC power would initiate MPS and ESF functions earlier in the event and, therefore, is not limiting. No single failure resulted in a more limiting MCHFR for the LOEL event. The RCS pressure does not reach the RSV actuation setpoint in this event. The limiting MCHFR versus time is shown in Figure 15.2-9.

For the peak RCS pressure and MCHFR cases, a loss of AC power provides the limiting results. Loss of DC power would initiate MPS functions earlier in the event and therefore is not limiting.

For the peak SG pressure case, normal AC and DC power are assumed to be available. A loss of AC or DC power is not conservative for this event because the loss of power would terminate feedwater flow and actuate MPS functions earlier in the event sequence.

RAI 15.02.01-3

The enveloping sequence of events for either the LOEL ~~events~~TT or LOCV transients are described in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure event, and Table 15.2-6 for the limiting MCHFR event.

15.2.1.3 Thermal Hydraulic and Subchannel Analyses

15.2.1.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a LOEL is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) 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 for a discussion of the VIPRE-01 code and evaluation model.

15.2.1.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator density and fuel temperature is taken from the beginning of cycle (BOC). The reactivity coefficients for moderator density are least negative at BOC. Thus, they are conservative for undercooling events such as LOEL, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102%. The most reactive rod is assumed to remain out of the core, with a delay of 2 seconds between the reactor trip signal and scram initiation. The most limiting combination of reactivity coefficients for moderator density and fuel temperature is applied. Instrument inaccuracy is accounted for by examining the sensitivity to the setpoints over the given margin of error.

The values for key input parameters and initial conditions for the evaluation of the LOEL event are listed in Table 15.2-1 for the maximum RCS pressure event, Table 15.2-2 for the maximum steam pressure event and Table 15.2-3 for the MCHFR event.

15.2.1.3.3 Results

As the RCS heats up, the expansion of water volume increases pressurizer level and pressure as shown in Figure 15.2-1 and Figure 15.2-7, respectively. Upon the reactor trip, power decreases as shown in Figure 15.2-2. Figure 15.2-3 presents the net reactivity from the control rod insertion. RCS temperature decreases due to the reactor trip and then increases due to the reduction in heat removal until DHRS begins to cool the primary system as plotted in Figure 15.2-4. RCS flow

RAI 15.02.01-2

Unless stated otherwise, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. For the turbine trip event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for a turbine trip event. The disabled PCSs are for RCS temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of a turbine trip event.

RAI 15.02.01-3

The description for the remaining sequence of the turbine trip event is the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios and also including the failure of one feedwater isolation value (FWIV) to close. The enveloping sequence of events for either the ~~turbine trip~~LOEP, TT. or LOCV transients are described in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure events and Table 15.2-6 for the limiting MCHFR event.

15.2.2.3 Thermal Hydraulic and Subchannel Analyses

15.2.2.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a turbine trip event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) 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 for a discussion of the VIPRE-01 code and evaluation model.

15.2.2.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator density and fuel temperature is taken from the Beginning of Cycle (BOC). The reactivity coefficients for moderator density are least negative at BOC. Thus, they are conservative for undercooling events such as turbine trip, since they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102%. The most reactive rod is assumed to remain out of the core, along with a delay of 2 seconds between the reactor trip signal and the scram initiation. The most limiting combination of reactivity coefficients for moderator density and fuel temperature is applied. Instrument inaccuracy is accounted for by examining the sensitivity to the setpoints over the given margin of error. The key parameters are listed in

A LOCV event is expected to occur one or more times in the life of the plant. Therefore, a LOCV event is an AOO as indicated in Table 15.0-1.

15.2.3.2 Sequence of Events and Systems Operation

RAI 15.02.01-12

The severity of a LOCV event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine stop valve~~s~~ (TSV~~s~~). Following the valve closure~~s~~, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event.

RAI 15.02.01-2

Unless stated otherwise, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. For the LOCV event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for an LOCV event. The disabled PCSs are for RCS temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOCV event.

RAI 15.02.01-3

The description for the remaining sequence of the LOCV event are the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios and also including the failure of one feedwater isolation value (FWIV) to close. However, for the limiting RCS pressure and MCHFR cases, the RSV does actuate for approximately 10 sec.The enveloping sequence of events for either the LOEL, TT, or LOCV transients are presented in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure events and Table 15.2-6 for the limiting MCHFR event.

15.2.3.3 Thermal Hydraulic and Subchannel Analyses

15.2.3.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a LOCV event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) 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 for a discussion of the VIPRE-01 code and evaluation model.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-4

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26, and guides the staff to review the extent to which plant and reactor protection systems are required to function.

FSAR Tier 2, Section 15.2.1 states that the RCS pressure does not reach the reactor safety valve (RSV) actuation setpoint for the limiting MCHFR case for the LOEL event. This appears to be true for LOEL based on a staff audit of EC-0000-1997, Revision 0. FSAR Tier 2, Sections 15.2.2 and 15.2.3 state that the events proceed the same as in Section 15.2.1, thus implying that RCS pressure does not reach the RSV actuation setpoint. However, the staff notes that EC-0000-1997 shows that the RSV does actuate for the overall limiting MCHFR event for LOEL, TT, and LOCV.

Therefore, please update the FSAR to state that the RSV actuates for the limiting MCHFR event.

NuScale Response:

The text in FSAR Sections 15.2.1, 15.2.2 and 15.2.3 have been revised in response to RAI 9407 question 15.02.01-1 to describe the enveloping analysis that is used for the transients presented in those sections. The specific discussion also clarifies that for the loss of external load (LOEL) event, the reactor safety valve (RSV) setpoint is not reached. Additionally, it is clarified in response to RAI 9407 question 15.02.01-3 that, for the loss of condenser vacuum (LOCV) event, the RSV setpoint is reached for the minimum critical heat flux ratio (MCHFR) case.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-5

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26, and guides the reviewer to evaluate the values of system parameters and initial core and system conditions as input to the model.

For the LOEL/TT/LOCV events, the staff notes that FSAR Tier 2, Table 15.2-1 does not list the steam generator heat transfer bias for the peak RCS pressure case, and FSAR Tier 2, Table 15.2-2 does not list the initial pressurizer pressure assumed for the peak SG pressure case. To enable the staff to make a finding with regard to the choice of initial system conditions, please update the FSAR to include the values for these biases.

In addition, the staff would expect biased-low SG heat transfer to reduce heat removal from the RCS and potentially be more challenging for RCS pressure and/or MCHFR. However, FSAR Tier 2, Table 15.2-3 indicates that biased-high SG heat transfer is limiting for MCHFR, and based on the staff audit of EC-0000-1997, a nominal value is used for peak RCS pressure. Explain why biased-low SG heat transfer is not limiting for these cases. Update the FSAR as appropriate.

NuScale Response:

For the loss of external load (LOEL), turbine trip (TT) and loss of condenser vacuum (LOCV) events, the peak reactor coolant system (RCS) pressure case applies a nominal steam generator (SG) heat transfer bias. The peak SG pressure case applies a nominal pressurizer pressure bias. These initial conditions have been added to FSAR Table 15.2-1 and Table 15.2-2 respectively.

Sensitivity cases performed for the peak RCS pressure case presented in FSAR Sections 15.2.1 to 15.2.3 indicate that SG heat transfer biasing has an insignificant influence on RCS pressure. Therefore a nominal SG heat transfer bias was selected for the limiting RCS pressure case. Sensitivity results are presented in Table 1. It is noted that the dominant contributor to

maximizing RCS pressure is the assumed loss of AC power at event initiation, resulting in a loss of secondary flow at time zero.

Table 1. Sensitivity to SG Heat Transfer Biasing

Description	Peak RCS Pressure (psia)
Limiting RCS Pressure Case	2158
w/ -30% SG Heat Transfer	2158

As discussed in Section 7.2.6.3 of "Non-Loss of Coolant Accident Analysis Methodology," TR-0516-49416-P Rev 1, sensitivity studies are performed to identify limiting primary and secondary side pressures for these heatup events. From Section 7.2 of TR-0516-49416-P Rev 1, “an event-specific parameter that is relevant to the acceptance criterion may be described as “challenging” in the event-specific summary, however, it is recognized that the parameter may not present the highest challenge for any event.” In this case, the minimum critical heat flux ratio (MCHFR) case provided in FSAR Section 15.2.1 to 15.2.3 is a representative example presented to demonstrate that MCHFR is sufficiently bounded by overcooling and reactivity events. Consideration of all possible sensitivity conditions are not evaluated for impact on MCHFR for these heatup events.

Initial conditions Tables 15.2-1 and 15.2-2 of the FSAR are modified to include SG heat transfer bias and pressurizer pressure, respectively.

Impact on DCA:

Tables 15.2-1 and 15.2-2 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.02.01-5

Table 15.2-1: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Peak RCS Pressure

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+100°F	200°F
Feedwater Temperature	+10°F	310°F
Moderator Temperature Coefficient	BOC	0.0 pcm/°F
Doppler Reactivity Coefficient	BOC	-1.4 pcm/°F
Decay Heat	EOC+10%	various
RCS Flow Rate	minimum	1179 lbm/s
Delayed Neutron Fraction	BOC	5.9154x10-3
Pressurizer Level	+8%	68%
RSV Lift Setpoint	+3%	2137 psia
RCS Average Coolant Temperature	-10°F	535°F
Steam Generator Pressure	+35 psia	535 psia
Pressurizer Pressure	+70 psia	1920 psia
Turbine Stop Valve Stroke Time	n/a	0.001 s
Time of Loss of AC Power	n/a	coincident
SG Heat Transfer Bias	nominal	various

RAI 15.02.01-5, RAI 15.02.01-12

Table 15.2-2: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - ~~LOCV -~~ Initial Conditions Peak SG Pressure

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+100°F	200°F
Feedwater Temperature	+10°F	310°F
Moderator Temperature Coefficient	BOC	0.0 pcm/°F
Doppler Reactivity Coefficient	BOC	-1.4 pcm/°F
Decay Heat	EOC+10%	various
RCS Flow Rate	minimum	1179 lbm/s
Delayed Neutron Fraction	BOC	5.9154x10-3
Pressurizer Level	+8%	68%
RSV Lift Setpoint	+3%	2137 psia
RCS Average Coolant Temperature	+10°F	555°F
Steam Generator Pressure	+35 psia	535 psia
SG Heat Transfer Bias	+30%	various
Turbine Stop Valve Stroke Time	n/a	0.001 s
Pressurizer Pressure	nominal	1850 psia

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-6

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26, and specifies that the applicant should analyze the events using an acceptable analytical model.

In auditing ECN-0000-4862 and EC-A010-1782, "NuScale NRELAP5 Module Basemodel" (the base analysis model used for the FSAR Chapter 15 analyses), the staff noted that the heat transfer option used to model heat transfer between the DHRS and the pool in the NRELAP5 basemodel and the LOEL, TT, and LOCV events appears to be different from what is stated to be used in TR-0516-49416-P, "Non-Loss-of-Coolant Accident Analysis Methodology," which is referenced in FSAR Chapter 15 as the evaluation model used for non-LOCA events. Please justify the differences in methodology between ECN-0000-4862/EC-A010-1782 and TR-0516-49416-P, and also clarify and justify the heat transfer option used for all other FSAR Chapter 15 analyses. Update the FSAR as appropriate.

NuScale Response:

As described in ECN-0000-4862, the limiting cases for the loss of external load (LOEL), turbine trip (TT) and loss of condenser vacuum (LOCV) events in FSAR Sections 15.2.1, 15.2.2 and 15.2.3 were revised to use the pool heat transfer model option prescribed by TR-0516-49416-P, "Non-Loss-of-Coolant Accident Analysis Methodology." The pool heat tranfser model had not been determined at the time of the development of the NRELAP5 model or at the time of the initial analysis. Therefore a sensitivity case was added to demonstrate this model change would not invalidate the sensitivity conclusions previously determined by the analysis. This methodology change was applied to the NRELAP5 models used throughout all of the Chapter 15 supporting analyses prior to completion of Revision 0 of the FSAR to ensure consistency with the non-LOCA Topical Report.

It is noted that the pool heat transfer option will not impact the peak RCS pressure or MCHFR scenarios as those occur well before DHRS actuation is completed. The peak steam pressure case showed little sensitivity (<2 psi) to the pool heat transfer option change.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-7

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26, and guides the reviewer to review, among other things, the sequence of events.

FSAR Tier 2, Tables 15.2-4 through 15.2-6 intend to provide the limiting sequence of events with respect to RCS maximum pressure, SG maximum pressure, and MCHFR for the LOEL, TT, and LOCV events. The staff notes that Tables 15.2-4 through 15.2-6 do not appear to be consistent with the sequence of events tables the staff audited in ECN-0000-4862, provided via EC-0000-1997, Revision 0. For example, Table 15.2-5 shows that the reactor trip and DHRS actuation analytical limit is reached at 5 seconds, and the trip and DHRS actuation occur at 7 seconds. However, ECN-0000-4862 shows these events occur at 7 and 9 seconds, respectively. Table 15.2-5 also shows that the RSV lift setpoint is reached for the SG pressure limiting case at 9 seconds, but both ECN-0000-4862 and FSAR Section 15.2.1.2 state that the RSV lift setpoint is not reached at all for the SG pressure case. Furthermore, based on Tables 15.2-4 through 15.2-6, the decay heat removal system (DHRS) valves fully open before the 30 second delay following DHRS actuation, which is not consistent with the analysis assumptions in EC-0000-1997 or the design basis description of the valves in FSAR Section 5.4.3.2.1, which states that the DHRS actuation valves are designed to fully open within 30 seconds from receipt of a DHRS actuation signal.

The limiting sequences of events should be accurately identified so the staff is able to assess the event response. Please identify whether any errors exist in Tables 15.2-4 through 15.2-6. If errors are present, update the FSAR as appropriate. If no errors are present, explain the apparent lack of consistency between the FSAR and ECN-0000-4862.

NuScale Response:

The timing presented in FSAR Tables 15.2-4 through 15.2-6 is incorrect. These tables are revised to correctly reflect the timing and sequence of events in ECN-0000-4862.

Impact on DCA:

Tables 15.2-4, 15.2-5 and 15.2-6 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.02.01-7

Table 15.2-4: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - RCS Maximum Pressure Sequence of Events

Event	Time (sec)
Event initiator -Turbine Trip and loss of FW flow	0
Turbine Stop Valves Fully Closed (assumption)	0
FW flow is secured (assumption)	0
Pressurizer heater power secured	0
CVCS flow secured (assumption)	0
Reactor Trip and DHRS Actuation analytical limit (High Pressurizer Pressure)	5
Reactor Trip issued	7
DHRS Actuation signal issued (including feedwater and steam isolation)	7
RSV opens	~~9~~10
Peak RCS Pressure	10
MSIVs are closed	14
FWIVs are closed	14
RSV Reseats	~~19~~20
DHRS actuation valves full open	~~35~~37

RAI 15.02.01-7

Table 15.2-5: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - SG Maximum Pressure Sequence of Events

Event	Time (sec)
Event initiator -Turbine Trip	0
Turbine Stop Valves Fully Closed (assumption)	0
~~FWIV Check valve seats~~	~~2~~
Reactor Trip and DHRS Actuation analytical limit (High Steam Pressure)	~~5~~7
Reactor Trip issued	~~7~~9
DHRS Actuation signal issued (including feedwater and steam isolation)	~~7~~9
MSIVs are closed	~~7~~16
~~Control rods fully inserted~~	~~9~~
~~RSV Lift setpoint (2137 psia)~~	~~9~~
~~Peak RCS pressure~~	~~10~~
FWIV train 1 is closed (train 2 fails to close)	~~14~~16
DHRS actuation valves full open	~~35~~39
FWRV train 2 is closed	39
Time of Peak Secondary Pressure	~~81~~82

RAI 15.02.01-7, RAI 15.02.01-9

Table 15.2-6: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - MCHFR Sequence of Events

Event	Time (sec)
Event initiator -Turbine Trip and loss of FW flow	0
Turbine Stop Valves Fully Closed (assumption)	0
FW flow is secured (assumption)	0
Reactor Trip and DHRS Actuation analytical limit (High Pressurizer Pressure)	~~13~~10
Reactor Trip and DHRS signal	~~15~~12
~~Time of MCHFR~~	~~16~~
~~Control Rods completely inserted~~	~~17~~
RSV Lift Point (2137 psia)	~~19~~18
MSIVs are closed	19
FWIVs are closed	~~22~~19
RSV reseats	~~30~~28
DHRS valves open	~~43~~42

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-8

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with GDC 15. GDC 15 requires that the reactor coolant system (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of several GDC, including GDC 15, and specifies that pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

The staff notes that FSAR Figure 15.2-7 shows pressurizer pressure for the LOEL, TT, and LOCV maximum RCS pressure case, but the peak pressure shown in the figure is less than the maximum RCS pressure in FSAR Table 15.2-7 (2158 psia), presumably because the maximum RCS pressure occurs at a different location in the RCS. To allow the staff to verify the RCS pressure response for these events, please provide and add to the FSAR a new figure that shows the pressure response for the location of maximum RCS pressure for the LOEL, TT, and LOCV events.

NuScale Response:

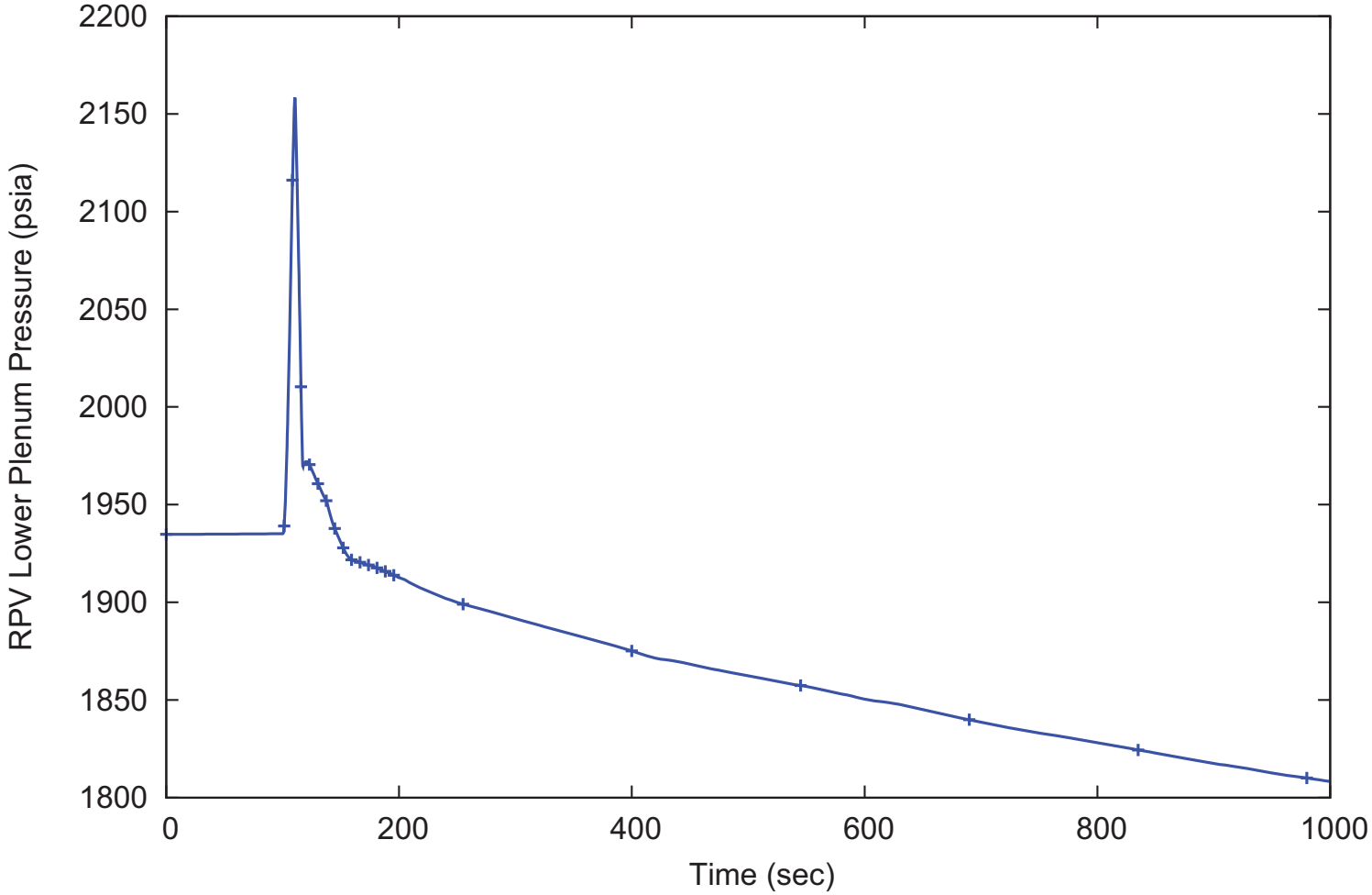
The requested figure has been added to the FSAR to show reactor pressure vessel (RPV) lower plenum pressure for the limiting loss of external load (LOEL), turbine trip (TT) and loss of condenser vacuum (LOCV) reactor coolant system (RCS) pressure transient scenario.

Impact on DCA:

FSAR Figure 15.2-7 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.02.01-8

Figure 15.2-7: Reactor Pressure Vessel Pressure - Peak RCS Pressure Case (15.2.1-15.2.3 LOEL-TT-LOCV)



Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-9

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of several GDC, including GDC 10, and specifies that the minimum departure from nucleate boiling ratio (DNBR) must remain above the 95/95 DNBR limit based on acceptable correlations and by satisfaction of any other SAFDL applicable to the particular reactor design. In this case, the SAFDL is MCHFR.

The timing of MCHFR in FSAR Tier 2, Figure 15.2-9, "Hot Channel Node MCHFR – Limiting MCHFR Case (15.2.1-15.2.3 LOEL-TT-LOCV)," is not consistent with the sequence of events in Table 15.2-6 (specifically, Figure 15.2-9 indicates that MCHFR occurs at about 13 seconds, while Table 15.2-6 states 16 seconds). The staff also notes that FSAR Tier 2, Figure 15.2-18, "Minimum Critical Heat Flux Ratio – Limiting MCHFR Case (15.2.4 MSIV Closure)," shows MCHFR at about 7 seconds, while FSAR Tier 2, Table 15.2-13, "Main Steam Isolation Valve Closure - Sequence of Events for MCHFR Limiting Case," reports 8 seconds.

Please update Figures 15.2-9 and 15.2-18 and/or Tables 15.2-6 and 15.2-13 as appropriate so the staff can be sure the correct MCHFR timing is plotted. In addition, correct any other MCHFR sequence of events tables and/or figures presented in FSAR Chapter 15 as necessary to reflect the correct timing of MCHFR.

NuScale Response:

FSAR Figures 15.2-9 and 15.2-18 where updated with the new NSP4 critical heat flux (CHF) correlation results upon Revision 1 of the FSAR and specifically delineate the timing of the minimum critical heat flux ratio (MCHFR) for the loss of external load (LOEL), turbine (TT), loss of condenser vacuum (LOCV), and main steam line isolation value (MSIV) closure events. These figures demonstrate that the heatup event sequences have little influence on MCHFR;

therefore, it is concluded that the timing of MCHFR is not important to the event sequence and is removed from Tables 15.2-6, 15.2-13, 15.2-17, 15.2-20, 15.2-26, and 15.2-30.

Impact on DCA:

Tables 15.2-6, 15.2-13, 15.2-17, 15.2-20, 15.2-26, and 15.2-30 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.02.01-7, RAI 15.02.01-9

Table 15.2-6: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - MCHFR Sequence of Events

Event	Time (sec)
Event initiator -Turbine Trip and loss of FW flow	0
Turbine Stop Valves Fully Closed (assumption)	0
FW flow is secured (assumption)	0
Reactor Trip and DHRS Actuation analytical limit (High Pressurizer Pressure)	~~13~~10
Reactor Trip and DHRS signal	~~15~~12
~~Time of MCHFR~~	~~16~~
~~Control Rods completely inserted~~	~~17~~
RSV Lift Point (2137 psia)	~~19~~18
MSIVs are closed	19
FWIVs are closed	~~22~~19
RSV reseats	~~30~~28
DHRS valves open	~~43~~42

RAI 15.02.01-9

Table 15.2-13: Main Steam Isolation Valve Closure - Sequence of Events for MCHFR Limiting Case

Event	Time (s)
Event initiator - MSIV Closure	0
High pressurizer pressure is reached	4
Loss of AC power occurs (FW pumps trip)	4
RTS and DHRS actuation initiates	6
~~MCHFR~~	~~8~~
FWIVs are fully closed	11
DHRS actuation valves are fully open	36

RAI 15.02.01-9

Table 15.2-17: Loss of Non-Emergency AC Power - Limiting MCHFR Sequence of Events

Event	Time [s]
Loss of AC power occurs	0
Turbine trip occurs	0
Feedwater pump trips.	0
CVCS pump trips (approximated as CVCS isolation)	0
High steam line pressure is reached	10
RTS actuation on high steam line pressure signal.	12
DHRS actuation on the high steam line pressure signal. DHRS actuation valves begin to open.	12
FWIVs and MSIVs begin to close.	14
~~MCHFR reached~~	~~14~~
RSV1 opens	16
Peak RPV pressure is reached	16
MSIVs are fully closed.	19
FWIVs are fully closed.	19
Peak steam generator pressure is reached.	42
DHRS actuation valves are fully open.	42

RAI 15.02.01-9

Table 15.2-20: Loss of Feedwater Event - RCS Overpressurization/Limiting MCHFR Sequence of Events

Event	Time [s]
Loss of Feedwater initiation	0
High pressurizer pressure (>2000 psia)	16
Turbine Trip	17
Loss of Normal AC	17
MSIVs signal	17
RTS actuation	18
DHRS actuation	18
~~Minimum CHF ratio~~	~~18~~
Rods fully inserted	20
RSV #1 lifts	20
Peak RCS pressure	21
DHRS valves complete opening	48
Peak secondary pressure	333

RAI 15.02.01-9

Table 15.2-26: Feedwater Line Break Sequence of Events - Limiting MCHFR Case

Event	Time [s]
Failure that initiates event.	0
AC power is lost resulting in turbine trip and FW pump trip	0
High PZR pressure analytical limit is reached (2000psia)	13
RTS and DHRS ESFAS actuated	15
MSIVs closed	15
~~MCHFR~~	~~16~~
Control rods fully inserted	17
RSV lift point is reached (2137 psia)	19
Peak RCS pressure reached	19
FWIVs Closed (check valves already seated)	22
RSV reseats	30
DHRS actuation valve fully open	43

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Table 15.2-30: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting MCHFR Case

Event	Time (s)
Transient initiation (spurious DHRS#1 valve actuation signal)	0.0
DHRS#1 actuation valve begins opening	29.7
DHRS#1 actuation valve fully open	30.0
High Hot Leg Temperature signal (610°F)	169.1
DHRS actuation (secondary isolation and DHRS #2 valve opens)	177.1
RTS actuation	177.1
~~Limiting MCHFR~~	~~177.6~~

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-10

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26. To meet these requirements, DSRS Section 15.2.1-15.2.5 states that the most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

FSAR Tier 2, Section 15.2.4, "Closure of Main Steam Isolation Valve(s)," states that no single failure results in more severe results for any of the acceptance criteria. However, the staff notes that a failure of the feedwater isolation valve (FWIV) to close is limiting for SG pressure in FSAR Tier 2, Sections 15.2.1-15.2.3 for the LOEL, TT, and LOCV events because of the additional feedwater provided to the SG. Justify why failure of a FWIV to close is not limiting for SG pressure for the MSIV closure event. Update the FSAR as appropriate.

NuScale Response:

Sensitivity cases performed for the limiting steam generator (SG) pressure case presented in FSAR Section 15.2.4 "Closure of Main Steam Isolation Valve(s)," show that SG pressure is maximized when no failure of a feedwater isolation valve (FWIV) to close is assumed. These results are presented in Table 1.

Table 1. Sensitivity to FWIV Failure for Closure of MSIVs Event

Description	Peak SG Pressure (psia)
Limiting SG Pressure Case	1481.0
w/ Failure of One FWIV	1480.3

For FSAR Sections 15.2.1 to 15.2.3, the failure of a FWIV to close was selected as limiting for SG pressure based on sensitivity results presented in rows 1 and 2 of Table 7-33 in "Non-Loss

of Coolant Accident Analysis Methodology" Topical Report, TR-0516-49416-P. These results are repeated in Table 2.

Table 2. Sensitivity to FWIV Failure for Turbine Trip Event

Description	Peak SG Pressure (psia)
w/ Failure of One FWIV	1472.3
Baseline	1472.1

It is noted that the sensitivity to a failure of a FWIV to close is small for peak SG pressure. The difference is less than 1.0 psia for both initiating events.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-11

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.1-15.2.5 provides guidance for meeting the requirements of GDC 10, 13, 15, 17, and 26, and guides the reviewer to evaluate the values of system parameters and initial core and system conditions as input to the model.
Several of the initial conditions selected for the MSIV closure event in FSAR Tier 2, Section 15.2.4 are not clear, because they either appear inconsistent with conclusions that can be drawn from the sensitivity studies in EC-0000-2995, Rev. 1, "Closure of Main Steam Isolation Valve Transient Analysis," or they deviate from the phenomenologically similar LOEL, TT, and LOCV event bias directions. Please explain the following points, and provide any updates to the FSAR as necessary:

- FSAR Tier 2, Table 15.2-8 lists initial conditions for the limiting RCS pressure case. The staff notes that the initial RCS average temperature (T_{ave}) is biased high, whereas the MSIV closure sensitivity studies and LOEL, TT, and LOCV analysis both seem to suggest a biased-low RCS T_{ave} is limiting for RCS peak pressure.

- Table 15.2-8 also shows a low SG pressure bias, whereas the LOEL, TT, and LOCV analysis used a high SG pressure bias.

- FSAR Tier 2, Table 15.2-9 lists initial conditions for the limiting SG pressure case. The initial pressurizer level and SG pressure are both biased low, whereas they are both biased high for the LOEL, TT, and LOCV analysis.

- FSAR Tier 2, Table 15.2-10 lists initial conditions for the MCHFR limiting case. The initial pressurizer pressure is biased high, although the MSIV closure sensitivity studies and the LOEL, TT, and LOCV analysis both suggest a low bias may be limiting.

Furthermore, FSAR Tier 2, Tables 15.2-8 and 15.2-9 for the MSIV closure event do not list the assumed RCS flow. Please update the tables to include the initial RCS flow.

NuScale Response:

Bullet 1

The inadvertent main steam isolation value (MSIV) closure analysis included sensitivities on reactor coolant system (RCS) temperature which showed for scenarios that did not lift the reactor safety values (RSVs) there was a small inverse sensitivity of RCS pressure to initial RCS temperature. An additional sensitivity with low RCS average temperature has been added to the analysis which shows little (~1psi) sensitivity for the limiting case which lifts the RSVs. This is because cases that lift the RSVs are less sensitive to initial conditions due to the RSV pressure relief capacity. The peak pressure reported in FSAR Table 15.2-14 has been updated to ensure a bounding value is presented.

Bullet 2

The inadvertent MSIV closure event analysis sensitivity results for steam pressure show a small sensitivity of about 3 psi on maximum RCS pressure from a low initial SG pressure condition. In this event the high main steam pressure trip can be reached and therefore lower initial steam pressure may delay reactor trip. For the loss of load, turbine trip, and loss of condenser vacuum events, there is very little sensitivity toward initial steam pressure for the peak pressure cases and therefore high steam pressure was generally applied. This is in part due to total volume of piping between the MSIV and turbine resulting in a different steam pressurization rates between the events.

Bullet 3

The inadvertent MSIV closure event analysis sensitivity results for PZR level and steam pressure show a slight delay (<1 second) in reactor trip with both biased low which results in slightly more severe peak steam pressure results. As described previously, the event sequence and timing for the turbine related transients is different than that of the MSIV closure event and therefore the sensitivity results may not be the same. Sensitivity results support high steam pressure bias as being more limiting and the PZR level is generally biased high per the methodology described in TR-0516-49416-P Rev . 1 "Non-Loss-Coolant Accident Analysis Methodology Topical Report."

Bullet 4

As discussed in Section 7.2.6.3 of TR-0516-49416-P Rev 1, sensitivity studies are performed to identify limiting primary and secondary side pressures for the heatup events. From Section 7.2 of TR-0516-49416-P Rev 1, “an event-specific parameter that is relevant to the acceptance criterion may be described as “challenging” in the event-specific summary, however, it is recognized that the parameter may not present the highest challenge for any event.” In this

case, the MCHFR case provided in FSAR Sections 15.2.1 to 15.2.4 is a representative example presented to demonstrate that MCHFR is sufficiently bounded by overcooling and reactivity events. Consideration of all possible sensitivity conditions are not evaluated for impact on MCHFR for these heatup events.

In addition, FSAR Tables 15.2-8 and 15.2-9 have been updated with the initial RCS flowrate.

Impact on DCA:

FSAR Tables 15.2-8, 15.2-9 and 15.2-14 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.02.01-11

Table 15.2-8: Main Steam Isolation Valve Closure - Initial Conditions for RCS Pressure Limiting Case

Parameter	Analysis Value
Initial Reactor Power (includes 2% uncertainty)	102% RTP (163.2 MW_{th})
Initial RCS T_{ave} (high bias)	555 °F
Initial Pressurizer Pressure (high bias)	1920 psia
Initial Pressurizer Level (high bias)	68%
Feedwater Temperature (high bias)	313 °F
Steam Generator Pressure (low bias)	465 psia
Steam Generator heat transfer (low bias)	70%
Initial RCS Flowrate (low bias)	1179 lbm/s
Moderator Temperature Coefficient (MTC) (BOC)	set to 0.0 at transient initiation
Doppler Reactivity Coefficient (BOC)	set to -1.4 at transient initiation
Delayed Neutron Fraction (ß) (BOC)	5.9154E-03
Safety Relief Valve Open Setpoint	2137 psia
Pool Temperature (high bias)	200°F

RAI 15.02.01-11

Table 15.2-9: Main Steam Isolation Valve Closure - Initial Conditions for SG Pressure Limiting Case

Parameter	Analysis Value
Initial Reactor Power (includes 2% uncertainty)	102% RTP (163.2 MW_{th})
Initial RCS Tave (high bias)	555 °F
Initial Pressurizer Pressure (high bias)	1920 psia
Initial Pressurizer Level (low bias)	52%
Feedwater Temperature (high bias)	313 °F
Steam Generator Pressure (low bias)	465 psia
SG Tube heat transfer (high bias)	130%
Initial RCS Flowrate (low bias)	1179 lbm/s
Moderator Temperature Coefficient (MTC) (BOC)	set to 0.0 at transient initiation
Doppler Reactivity Coefficient (BOC)	set to -1.4 at transient initiation
Delayed Neutron Fraction (ß) (BOC)	5.9154E-03
Safety Relief Valve Open Setpoint	2137 psia
Turbine Stop Valve Stroke Time	0.1 sec
Pool Temperature (high bias)	200°F

RAI 15.02.01-11

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

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2310 psia	2160~~58~~
Maximum SG Pressure	2310 psia	1481
MCHFR	1.284	2.567

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9407
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.01-12

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. Per DSRS Section 15.2.1-15.2.5, the information in the related FSAR sections is reviewed against GDC 10, 13, 15, 17, and 26. The information in the FSAR that supports meeting these regulations needs to be accurate and consistent so the staff is able to make a reasonable assurance finding.

The staff noted that FSAR Tier 2, Sections 15.2.1-15.2.5 and the related tables and figures contain apparent typographical errors that affect technical meaning or details. These errors are listed below:

- FSAR Section 15.2.1.3.3, “Results,” refers to limiting MCHFR in Figure 5.2-9, but the correct figure number is 15.2-9.
- FSAR Sections 15.2.2.2 and 15.2.3.2, both titled “Sequence of Events and System Operation,” refer to more than one turbine stop valve (TSV). However, the staff notes that FSAR Chapter 10 indicates there is only one TSV per NPM.
- The title of Table 15.2-3 indicates it is a sequence of events table, but it is actually an initial conditions table.
- FSAR Tier 2, Figure 15.2-12 is titled “Net Reactivity – Peak RCS Pressure Case (15.2.4 MSIV Closure)” but appears to show RCS flow rate instead.

Please address the above items by (1) justifying why they are correct as written or (2) by updating FSAR Tier 2 to correct them.

NuScale Response:

The FSAR Sections, Table and Figure noted above have been corrected. In addition, an editorial correction is made to the caption of Table 15.2-2.

Impact on DCA:

FSAR Sections 15.2.1.3.3, 15.2.2.2, 15.2.3.2, Tables 15.2-2, 15.2-3 and Figure 15.2-12 have been revised as described in the response above and as shown in the markup provided in this response.

(Figure 15.2-5) drops initially due to the reactor trip and is reestablished as DHRS flow is established (Figure 15.2-6). Steam generator pressure for the peak SG pressure case is presented in Figure 15.2-8.

RAI 15.02.01-12

LOEL results in increased temperature in the RCS which could potentially challenge fuel parameters. Although RCS fluid and fuel temperatures increase, the core remains covered throughout the event, such that the MCHFR limits are not challenged. The limiting MCHFR is demonstrated in ~~Figure 5.2-9~~Figure 15.2-9.

15.2.1.4 Radiological Consequences

The radiological consequences of an LOEL event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.2.1.5 Conclusions

RAI 15.02.01-1

The six DSRS acceptance criteria for this AOO are met for the enveloping analysis which includes: Loss of External Load, Turbine Trip and Loss of Condenser Vacuum cases. These acceptance criteria, followed by how the NuScale design meets them are listed below:

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

   - The limiting RCS pressure for this event, shown in Table 15.2-7, is below 110% of the design value for the reactor coolant system.
   - The limiting steam generator pressure, shown in Table 15.2-7, is below 110% of the design value for the main steam system up to the MSIVs.

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

   - The MCHFR for this event, shown in Table 15.2-7, is above the 95/95 limit.

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

   - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met. The LOEL event does not lead to a more serious event.

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the 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 GDCs 10 and 15.

- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

   - The limiting single failure for an LOEL event is a failure of a feedwater isolation valve to close for the limiting SG pressure case. Results from this scenario do not challenge the identified limits.

6) The guidance provided in SECY-77-439, SECY-94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

   - The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.2.1.2 and Section 15.2.1.3 and ensure the guidance provided is met.

15.2.2 Turbine Trip

15.2.2.1 Identification of Causes and Event Description

A turbine trip event is initiated by the closure of the turbine stop valves (TSVs). For the NPM design, the effect of a turbine trip is similar to LOEL and LOCV. No credit is taken for the turbine bypass system or other control systems and, therefore, numerous secondary side or electrical perturbations can result in a trip of the turbine generator. The turbine trip causes the primary and secondary side temperatures and pressures to increase because energy is not being removed through the steam generators to the condenser. The reactor trip signal and Decay Heat Removal System (DHRS) actuation signal are initiated on high pressurizer pressure or high steam line pressure. The reactor trip reduces power to decay heat levels. The DHRS actuates and transfers decay heat to the reactor pool. If offsite power is lost, with a coincident loss of DC power, the reactor trip, DHRS actuation and main steam isolation occur simultaneously.

A turbine trip event is expected to occur one or more times in the life of the plant. Therefore, a turbine trip event is an AOO as indicated in Table 15.0-1.

15.2.2.2 Sequence of Events and Systems Operation

RAI 15.02.01-12

The severity of a turbine trip is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of by the turbine stop valve~~s~~ (TSV~~s~~). Following the valve closure~~s~~, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event.

A LOCV event is expected to occur one or more times in the life of the plant. Therefore, a LOCV event is an AOO as indicated in Table 15.0-1.

15.2.3.2 Sequence of Events and Systems Operation

RAI 15.02.01-12

The severity of a LOCV event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine stop valve~~s~~ (TSV~~s~~). Following the valve closure~~s~~, SG pressure continues to increase until DHRS establishes natural circulation. DHRS establishes cooling and begins to depressurize the SGs and the RCS approximately 80 seconds into the event.

RAI 15.02.01-2

Unless stated otherwise, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. For the LOCV event, containment pressure control is enabled because its operation is benign with respect to the event consequences. In contrast, most PCSs are disabled because their operation is beneficial with respect to the consequences for an LOCV event. The disabled PCSs are for RCS temperature control, pressurizer pressure control, pressurizer level control, steam pressure control, and feedwater and turbine load control. No operator action is credited to mitigate the effects of an LOCV event.

The description for the remaining sequence of the LOCV event are the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios. The sequence of events for the LOCV are presented in Table 15.2-4 for the limiting RCS pressure event, Table 15.2-5 for the limiting secondary pressure events and Table 15.2-6 for the limiting MCHFR event.

15.2.3.3 Thermal Hydraulic and Subchannel Analyses

15.2.3.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a LOCV event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) 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 for a discussion of the VIPRE-01 code and evaluation model.

RAI 15.02.01-5, RAI 15.02.01-12

Table 15.2-2: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - ~~LOCV -~~ Initial Conditions Peak SG Pressure

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+100°F	200°F
Feedwater Temperature	+10°F	310°F
Moderator Temperature Coefficient	BOC	0.0 pcm/°F
Doppler Reactivity Coefficient	BOC	-1.4 pcm/°F
Decay Heat	EOC+10%	various
RCS Flow Rate	minimum	1179 lbm/s
Delayed Neutron Fraction	BOC	5.9154x10-3
Pressurizer Level	+8%	68%
RSV Lift Setpoint	+3%	2137 psia
RCS Average Coolant Temperature	+10°F	555°F
Steam Generator Pressure	+35 psia	535 psia
SG Heat Transfer Bias	+30%	various
Turbine Stop Valve Stroke Time	n/a	0.001 s
Pressurizer Pressure	nominal	1850 psia

RAI 15.02.01-12

Table 15.2-3: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions MCHFR ~~Sequence of Events~~

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+100°F	200°F
Feedwater Temperature	+10°F	310°F
Moderator Temperature Coefficient	BOC	0.0 pcm/°F
Doppler Reactivity Coefficient	BOC	-1.4 pcm/°F
Decay Heat	EOC+10%	various
RCS Flow Rate	minimum	1179 lbm/s
Delayed Neutron Fraction	BOC	5.9154x10-3
Pressurizer Level	+8%	68%
RSV Lift Setpoint	+3%	2137 psia
RCS Average Coolant Temperature	+10°F	555°F
SG Heat Transfer Bias	+30%	various
Pressurizer Pressure	-70 psia	1780 psia
Turbine Stop Valve Stroke Time	n/a	0.001 s

RAI 15.02.01-12

Figure 15.2-12: Net Reactivity - Peak RCS Pressure Case (15.2.4 MSIV Closure)