RAIO-0118-58343

January 25, 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. 297 (eRAI No. 9213) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 297 (eRAI No. 9213)," dated December 15, 2017

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 Question from NRC eRAI No. 9213 :

- 03.06.03-11

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 Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9213

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9213

NUSCALE POWER™

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

eRAI No.: 9213
Date of RAI Issue: 12/15/2017

NRC Question No.: 03.06.03-11

In response to RAI 8843 concerning the Leak-Before-Break (LBB) application for main steam system (MSS) and feedwater system (FWS) line breaks, the applicant stated that technical specifications (TS) limiting conditions for operation (LCO) 3.4.5 is for reactor coolant system (RCS) leakage and not for MSS/FWS piping leakage. The staff found the response to be unacceptable for the following reasons:

1. In order to meet the requirements of Title 10 of the Code of Federal Regulations Part 50, Appendix A, General Design Criterion 4 and the guidance of Standard Review Plan Section 3.6.3, the procedures under Regulatory Guide (RG) 1.45 "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" address the monitoring of the RCS leakage. The applicability of these procedures support the implementation of a leakage limit as specified in LCO 3.4.5 for the RCS. However, the applicant is proposing to use the RG 1.45 procedures to monitor leakage of MSS and FWS piping without a corresponding LCO leakage limit. To correctly use RG 1.45 procedures for MSS and FWS leakage, a limit similar to LCO 3.4.5 for RCS leakage must be specified for MSS and FWS leakage.
2. The proposed procedures do not require a licensee to take action in the event a MSS/FWS leak is identified to remediate the source of the leak to ensure that integrity of the MSS/FWS will be maintained under all design basis scenarios.

Equipment in the containment of a NuScale small modular reactor will not be protected from dynamic effects of main steam line and feedwater line breaks in the NuScale design, based on the assumption for the success of LBB

Standard Review Plan Section 3.6.3 states that, "The specifications for plant specific leakage detection systems inside the containment should be equivalent to those in RG 1.45," which in turn requires that under certain circumstances (e.g., to support LBB for smaller diameter pipes), leakage monitoring system specifications may need to exceed the quantitative criteria in RG 1.45. In addition, 10 CFR 50.36(a)(2) requires the applicant for a design certification under Part 52 of this chapter to include technical specifications in accordance with the requirements of this section for the portion of the plant that is within the scope of the design certification. The failure of the proposed LBB lines could lead to the failure of the instrumentation used to detect/indicate a significant abnormal degradation of the reactor coolant pressure boundary as indicated in Criterion 1 10 CFR 50.36(c)(2)(ii), or a failure to the integrity of a fission product barrier due to jet impingement and pipe whip (see Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Based on the above review, the applicant is requested to propose a TS LCO leakage limit consistent with the moment versus leak rate curves used for flaw stability criteria - especially since both the MSS and FWS involve small diameter piping. The staff has previously requested moment versus leak rate curves for the MSS and FWS piping in RAI 9113 Question 03.06.03-8

NuScale Response:

The Introduction of Regulatory Guide (RG) 1.45, Revision 1, *Guidance on Monitoring and Responding to Reactor Coolant System Leakage*, specifically states (**emphasis** added) that

> … Regulatory Guide 1.45 (Ref. 1) describes methods that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in implementing the regulatory requirements specified below…
>
> General Design Criterion (GDC) 14, "**Reactor Coolant Pressure Boundary**," as set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the Code of Federal Regulations (10 CFR Part 50), (Ref. 2), …
>
> Consequently, GDC 30, "Quality of **Reactor Coolant Pressure Boundary**" (Ref. 2), requires that plants provide the means for detecting and, to the extent practical, identifying the location of the source of **reactor coolant leakage**…
>
> Thus, the concept of defense in depth is applied to provide assurance that the structural integrity of the **RCPB** is maintained.

Regulatory Guide 1.45 does not address secondary side pressure boundary monitoring requirements. It does describe expectations for leakage detection systems, specifically those that monitor the reactor coolant pressure boundary, which has been extended to the application of monitoring LBB piping in industry practice.

The RAI indicates that the procedures described in the response to eRAI 8843 that are required to monitor any postulated RCS leakage as required by LCO 3.4.5 cannot monitor for LBB leakage 'without a corresponding LCO leakage limit'. However, plant procedures are routinely written and implemented to maintain systems "without a corresponding LCO." Systems and functions that are not within the scope of 10 CFR 50.36 are managed with other programs in accordance with the appropriate requirements. For example programs and procedures that implement the Maintenance Rule of 10 CFR 50.65, in combination with the corrective action and conduct of operations, monitor and assure the availability and reliability of the LBB leakage monitoring functionality.

These administrative means governed by licensing basis commitments are used to verify the functionality of the majority of safety and non-safety SSC in a nuclear power plant. Most criteria and operating limits that assure the availability and reliability of nuclear power plant systems are not the subject of an LCO.

Establishing limits and requirements for SSC that do not fall within the scope of 10 CFR 50.36 is consistent with Commission policy and rulemaking as described in 58 FR 39132, *Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors*, July 22, 1993, and 60 FR 36953, *Technical Specifications*, July 19, 1995. The Commission policy explicitly indicates that the FSAR describe the Commission-accepted safety basis of the facility for systems that are not within the Technical Specification scope as described and required by 10 CFR 50.36. The very impetus for the policy and rulemaking was to reduce the scope of the Technical Specifications to those that met the criteria of the rule.

The leakage limit calculation and construction of the LBB bounding curves are described in FSAR Section 3.6.3. When plant procedures are written, they will implement the requirements of the FSAR including the LBB detection and functionality requirements. Operating programs and procedures will ensure the availability and reliability of the function.

The statement that 'proposed procedures do not require a licensee to take action…' is inaccurate because the procedures have not been written.

The RAI indicates that failure of the LBB lines could lead to failure of the instrumentation used to satisfy Criterion 1 of 10 CFR 50.36(c)(2)(ii). If such an event were to occur, the appropriate Conditions of LCO 3.4.5 would be entered and complied with. The potential for a failure of a component to adversely affect a TS required system is the purpose for the Conditions and Required Actions of the LCO, not a basis for inclusion in the TS.

The RAI continues, indicating that failure of the LBB lines could lead to 'a failure to the integrity of a fission product barrier due to jet impingement and pipe whip' and implies that this could satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) and require an LCO for LBB monitoring. However this is inconsistent with the application of Criterion 2, which requires a TS limiting condition for operation for:

> A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Postulated 'jet impingement and pipe whip' does not satisfy this criterion. LBB monitoring is used to anticipate and assess the potential for failure of the LBB lines, thereby permitting evaluation of the condition and corrective measures to be implemented. It does not monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient that challenges the integrity of a fission product barrier (the reactor coolant pressure boundary or the containment in this case.)

In summary, and as explicitly stated in a similar situation addressed in the Bases for LCO 3.7.8, Main Steam Line Leakage of NUREG 2194, Volume 2: Bases, *Standard Technical Specifications Westinghouse Advanced Passive (AP1000) Plants*, Revision 0:

> … [T]he main steam line leakage limit is not required by the 10 CFR 50.36(c)(2)(ii) criteria….

NuScale has also revised FSAR 3.6.3.5, Leak Detection, to address the staff concern and desire for additional assurance of the reliability and availability of the LBB detection capability. The change appends a paragraph indicating that LBB monitoring availability and limits will be included in the owner-controlled requirements manual. Consistent with the Commission Policy and industry practice, this manual is to be prepared by a COL holder to describe limits and manage the availability and reliability of certain systems and components, including those relocated from the previous standardized versions of the Technical Specifications. A new COL item has been added to FSAR Section 16.1 that requires a COL applicant to prepare the owner-controlled requirements manual.

Impact on DCA:

The Technical Specifications and FSAR have been revised as described in the response above and as shown in the markup provided in this response.

3.6.3.4.3.2 Feedwater System Piping

The LBB allowable maximum axial and bending stress loads are compared against the actual normal operating plus SSE loadings of the FWS piping. The data for SBAC are summarized in Table 3.6-3b. The actual loads (the combined axial loads and the combined bending stresses as defined in SRP 3.6.3), for a given LBB location, fall within the SBAC depicted in Figure 3.6-29, Figure 3.6-30, Figure 3.6-31 and Figure 3.6-32. Therefore, it is concluded that the FWS piping meets the LBB criteria.

3.6.3.5 Leak Detection

Section 5.2.5 describes the leak detection system for inside the CNV. The SRP 3.6.3 states "The specifications for plant-specific leakage detection systems inside containment are equivalent to those in Regulatory Guide 1.45." As noted in Section 5.2.5, the reactor coolant pressure boundary leakage detection systems for the NPM conform to the sensitivity and response times recommended in RG 1.45, Revision 1.

This section describes the analysis methods used to support the application of LBB to high-energy piping in the NPM.

Regulatory Guide 1.45 Regulatory Position 2.1 states plant procedures should include the collection of leakage to the primary reactor containment from unidentified sources so that the total flow rate can be detected, monitored, and quantified for flow rates greater than 0.05 gpm. According to RG 1.45 Regulatory Position 2.2, the plant should use leakage detection systems with a response time of no greater than 1 hour for a leakage rate of 1 gpm.

Leakage monitoring is provided by two means, change in pressure within the CNV and collected condensate from the CES sample vessel.

The minimum detectable leak rate for the CES sample vessel is not easily quantified, since all liquid or vapor leaks within the CNV are eventually collected in the CES sample vessel. Once in the CES sample vessel, the minimum detectable volume is 0.042 gal or 0.333 lb of liquid. While there is theoretically no minimum detectable leak rate, main steam and feedwater system leak rates of 0.001 gpm or 0.01 lbm/min take less than 60 minutes to accumulate more than the minimum detectable volume.

To satisfy Regulatory Position 2.1 of RG 1.45, once the operators observe a pressure change in containment, a leak rate procedure is initiated to quantify the total leak rate. This, combined with other indications can aid in determining the leak source. In this instance, leaks can be detected using the CES sample vessel, where condensable fluids are collected after they are removed from containment via the vacuum pumps. The sample vessel level is configured to alarm the control room. Once a higher equilibrium pressure is reached during a leak scenario, leak rate measurements can be taken with the CES alone, using the CES sample tank.

RAI 03.06.03-11

The LBB leak detection availability and limits will be included in the owner-controlled requirements manual.

3.6.4 High Energy Line Break Evaluation (Non-LBB)

The GDC 4 requires that components be appropriately protected against the dynamic effects that may result from pipe ruptures. High-energy and moderate-energy piping systems that cannot be fully excluded using either the BTP 3-4, Section B.A.(ii) criteria, or LBB, must be designed for HELB. The specific locations for the postulated break locations are determined using the criteria in BTP 3-4. In general, welds meeting certain stress, fatigue and design requirements may be excluded and are not required to be postulated to rupture. Other locations, such as terminal ends or high stress locations, must be postulated to rupture.

At postulated rupture locations, the consequences of HELB can include pipe whip or jet impingement, both of which can potentially damage safety related equipment required for safe shutdown. At break locations, the pipe must either be located such that there is no essential equipment in the area, or the pipe must be restrained from whip, and equipment protected from jet impingement, as needed.

The piping systems that must be considered include the Class 1, Class 2, Class 3 and B31.1, high-energy and moderate-energy systems, located inside and outside of the CNV up to the reactor pool wall penetrations. Piping outside of the NPM is the responsibility of the COL applicant.

3.6.4.1 Postulation of Pipe Breaks in Areas Other than Containment Penetration

RAI 03.06.02-10

a) With the exceptions of those portions of piping identified in the first two paragraphs of ~~Section 3.6.2.1.1~~Section 3.6.2.1.2 and the portions of piping identified in Section 3.6.2.5, postulated pipe break locations are determined using the criteria described in Section 3.6.2.~~, breaks in Class 1 piping (ASME Code, Section III) are postulated at the following locations in each piping and branch run:at terminal ends~~

b) ~~at intermediate locations where the maximum stress range as calculated by Eq. (10) and either Eq. (12) or Eq. (13) exceeds 2.4~~ ~~S_m~~

c) ~~at intermediate locations where the cumulative usage factor exceeds 0.1~~

~~The RCS/CVCS discharge piping is representative of the NPM ASME Class 1 piping with respect to deadweight, seismic, thermal transient and fatigue loading. The discharge line is longer than other Class 1 lines, with more seismic supports and longer spans between thermal restraints. Therefore, this analysis presents the more challenging analysis case.~~

As a result of piping reanalysis, the highest stress locations may be shifted; however, the initially determined intermediate break locations need not be changed unless one of the following conditions exists:

Table 16.1-1 provides the initial surveillance test frequencies to be incorporated into the Surveillance Frequency Control Program (SFCP) required by NuScale GTS 5.5.11. The table identifies each GTS surveillance test requirement that references the SFCP, the base testing frequency for evaluation of future changes to the surveillance test frequency, and the basis for that test frequency.

Incorporation of Technical Specification Task Force Change Travelers

Technical Specification Task Force (TSTF) travelers issued since publication of Revision 4 of the ISTS were reviewed in the development of the NuScale GTS. Travelers were incorporated into the NuScale GTS or utilized as a basis for similar NuScale situations as described in the conformance report (Reference 16.1-1). The TSTF travelers considered in development of the NuScale GTS are listed in that report.

The GTS are intended to be used as a guide in the development of the plant-specific technical specifications. Preliminary information has been provided in single brackets [ ]. Combined license applicants referencing the NuScale Power Plant are required to provide the final plant-specific information.

COL Item 16.1-1: A COL applicant that references the NuScale Power Plant design certification will provide the final plant-specific information identified by [ ] in the generic Technical Specifications.

RAI 03.06.03-11

COL Item 16.1-2: A COL applicant that references the NuScale Power Plant design certification will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.

16.1.2 References

16.1-1 Technical Report TTR-1116-52011, "Technical Specifications Regulatory Conformance and Development Technical Report," Rev. 0.

16.1-2 NEI 04-10, Risk-Informed Technical Specifications Initiative 5b - Risk-Informed Method for Control of Surveillance Frequencies - Industry Guidance Document, Rev. 1, April 2007.

16.1-3 NEI 06-09, Risk-Informed Technical Specifications Initiative 4b - Risk-Managed Technical Specifications (RMTS) Guidelines - Industry Guidance Document, Rev. 0-A, November 2006.

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 [owner-controlled 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."

BASES

BACKGROUND (continued)

6. Demineralized Water Supply Isolation (DWSI) actuation;

7. Pressurizer Heater Trip (PHT) actuation; and

8. Low Temperature Overpressure Protection (LTOP) actuation.

Equipment actuated by each of the above signals is identified in the FSAR (Ref. 4). Setpoints are specified in the [owner-controlled requirements manual].

This LCO addresses the equipment from the MPS input sensors to the input to the RTS and ESFAS SVMs. The MPS RTS and ESFAS equipment from the inputs of the SVMs to the outputs of the equipment interface modules (EIMs) to the actuated devices is addressed in LCO 3.3.2, “Reactor Trip System (RTS) Logic and Trip Initiation,” and LCO 3.3.3, "Engineered Safety Features Actuation System (ESFAS) Logic and Actuation", respectively. Manual actuation of the RTS and ESFAS from the actuating switches to the backplane connections of the chassis are addressed in LCO 3.3.4, “Manual Actuation Functions.”

The roles of each of the MPS functions in the RTS and ESFAS, including the actuation logic of LCO 3.3.2, 3.3.3, and 3.3.4 are discussed below.

Measurement Channels

Measurement channels, consisting of field transmitters or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the process variable~~parameter~~ being measured. Some measurement channels that are processed by MPS are sent to MCS for control functions (e.g., pressurizer pressure and level).

The excore nuclear instruments are considered components in the measurement channels of the High Power Range Linear Power, High Power Range Positive and Negative Rate, Source Range Count Rate, Source Range Log Power Rate, and High Intermediate Range Log Power Rate Neutron Flux trips.

Four identical measurement channels (also designated separation group-A through D) with electrical and physical separation are provided for each variable~~parameter~~ used in the generation of trip and actuation signals.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~Design Basis Definition~~

The MPS is designed to ensure that the following operational criteria are met:

- The associated actuation will occur when the variable~~parameter~~ monitored by each channel reaches its setpoint and the specific coincidence logic is satisfied; and
- Separation and redundancy are maintained to permit a channel to be out of service for testing or maintenance while still maintaining redundancy within the MPS instrumentation architecture.

~~All design basis events can be mitigated by one or more MPS Functions. The accident analysis takes credit for most of the MPS trip Functions.~~Each of the analyzed accidents and transients which require a reactor trip or engineered safety feature can be detected by one or more MPS Functions. The MPS Functions that are credited to mitigate specific design basis events are described in the FSAR Chapter 15 (Ref. 9). Setpoints are specified in the [owner-controlled requirements manual].

Each MPS setpoint is chosen to be consistent with the function of the respective trip. The basis for each setpoint falls into one of three general categories:

- To ensure that the SLs are not exceeded during AOOs;
- To actuate the RTS and ESFAS during accidents; and
- To prevent material damage to major ~~MODULE~~ components (equipment protection).

The MPS maintains the SLs during AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

The Module~~MODULE~~ Protection System instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Permissive and interlock setpoints automatically provide, or allow manual or automatic blocking of trips during unit~~MODULE~~ evolutions. They are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the initial conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative initial conditions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

Operating~~onal~~ bypasses are addressed in the footnotes to Table 3.3.1-1. They are not otherwise addressed as specific Table entries.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Trip System (RTS) Logic and Actuation

BASES

BACKGROUND

The RTS portion of the Module Protection System (MPS) initiates a reactor trip to protect against violating the core fuel design limits and maintain reactor coolant pressure boundary integrity during anticipated operational occurrences (AOOs) and postulated accidents. By tripping the reactor, the RTS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

LCO 3.3.2 addresses only the logic and actuation portions of the MPS that perform the RTS function. The scope of this LCO begins at the inputs to the scheduling and voting modules (SVM) and extends through the actuated components. This includes the reactor trip breakers (RTBs). LCO 3.3.1, “Module Protection System (MPS),” LCO 3.3.3, "Engineered Safety Features Actuation System (ESFAS) Logic and Actuation," provide requirements on the other portions of the MPS that automatically initiate the Functions described in Table 3.3.1-1.

Details of the design~~,~~ and operation~~, and setpoints~~ of the entire MPS are provided in the Bases for LCO 3.3.1, “Module Protection System (MPS).” Setpoints are specified in the [owner-controlled requirements manual]. As noted there, the MPS transmits trip determination data to both divisions of the RTS SVMs. Redundant data from all four separation groups is received by each division of the RTS SVMs.

Logic for Reactor Trip Initiation

The MPS reactor trip initiation logic is implemented in two divisions of RTS. The three SVMs, in each division, generate a reactor trip signal when safety function modules (SFMs) in any two of the four separation groups determine a reactor trip is required. Each of the two RTS divisions evaluate the input signals from the SFMs from all four separation groups.

Each SVM compares the four inputs received from the SFMs, and generates a reactor trip signal if required by two of the four separation groups. The output of the three redundant SVMs is communicated via three independent safety data buses to the associated equipment interface modules (EIMs).

The EIMs compare inputs from the three SVMs and initiate an actuation if two out of three signals agree on the need to actuate.

B 3.3 INSTRUMENTATION

B 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation

BASES

BACKGROUND

The ESFAS portion of the Module~~MODULE~~ Protection System (MPS) protects against violating the core fuel design limits, ensures reactor coolant pressure boundary integrity during anticipated operational occurrences (AOOs) and postulated accidents, and ensures acceptable consequences during accidents by initiating necessary safety systems.

Details of the design and operation of the entire MPS are provided in the Bases for LCO 3.3.1, “Module Protection System (MPS).” Setpoints are specified in the [owner-controlled requirements manual]. As noted there, the MPS transmits trip determination data to both divisions of the ESFAS scheduling and voting modules (SVMs). Redundant data from all four separation groups is received by each division of the ESFAS SVMs.

LCO 3.3.3 addresses only the logic and actuation portions of the MPS that perform the ESFAS functions. The scope of this LCO begins at the inputs to the ~~scheduling and voting modules (~~SVMs~~)~~ and extends through the actuating contacts on the actuated components. This LCO also includes the pressurizer heater trip breakers. Component OPERABILITY and surveillance requirements are provided in the system LCOs and by programmatic requirements identified in Chapter 5, Administrative Controls.

LCO 3.3.1, “Module Protection System (MPS),” and LCO 3.3.2, "Reactor Trip System (RTS) Logic and Actuation," provide requirements on the other portions of the MPS that automatically initiate the Functions described in Table 3.3.1-1.

The ESFAS logic and actuation consists of:

1. Emergency Core Cooling System (ECCS) actuation;
2. Decay Heat Removal System (DHRS) actuation;
3. Containment Isolation System (CIS) actuation;
4. Demineralized Water Supply Isolation (DWSI) actuation;
5. Chemical Volume and Control System Isolation (CVCSI) actuation;
6. Pressurizer Heater Trip (PHT); and
7. Low Temperature Overpressure Protection (LTOP) actuation.