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10 CFR 50.90

July 12, 2016
GO2-16-087

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

SUBJECT: **Columbia Generating Station, Docket No. 50-397**
License Amendment Request for Changes to Technical Specification 2.1.1, Reactor Core SLs

References: GE Energy-Nuclear, "10 CFR Part 21 Communication, Reportable Condition [21.21(d)] Potential to Exceed Low Pressure Technical Specification Safety Limit", March 29, 2005

Dear Sir or Madam:

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Energy Northwest is submitting a request for an amendment to the Columbia Generating Station (Columbia) Technical Specifications (TS). The proposed amendment reduces the minimum reactor dome pressure associated with the critical power correlation from 785 psig to 685 psig in TS 2.1.1 Reactor Core Safety Limits (SL), and associated Bases.

The proposed changes have been evaluated in accordance with 10 CFR 50.92(c). It has been determined that the changes involve no significant hazards considerations. Attachment 1 provides the No Significant Hazards Consideration for the change and provides a description of the proposed change.

Attachment 2 provides the existing TS pages marked up to show the proposed changes. Attachment 3 provides the existing TS Bases (TSB) pages marked up to show the proposed changes (for information only). Attachment 4 contains the TS clean pages.

Energy Northwest requests approval of the proposed License Amendment one year from the date of submittal, with the amendment being implemented within 60 days of approval.

In accordance with 10 CFR 50.91, a copy of this application with attachments is being provided to the designated Washington State Official.

There are no new commitments being made by this submission.

If you should have any questions regarding this submittal, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

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

Executed on 12th day of July, 2016.

Respectfully,

for Alex Javorik

A. L. Javorik
Vice President, Engineering

Attachments: As stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Sr. Resident Inspector - 988C
CD Sonoda - BPA - 1399 (w/o enclosures)
WA Horin - Winston & Strawn (email)
RR Cowley - WDOH (email)
EFSECutc.wa.gov-- EFSEC (email)

DESCRIPTION AND ASSESSMENT

1.0 SUMMARY

On March 29, 2005, GE Energy-Nuclear (GE) issued a Safety Communication (SC 05-03) in accordance with 10 CFR 21.21(d) invoking a Reportable Condition for Potential Violation of Low Pressure Technical Specification (TS) Safety Limit (SL) (Reference 1). This condition is applicable to Columbia Generating Station (Columbia). GE identified a condition where a Pressure Regulator Failure Open (PRFO) – Maximum Demand Anticipated Operational Occurrence (AOO) may cause a TS SL to be violated since reactor dome pressure could drop below the current TS 2.1.1.1 SL pressure value of 785 psig while reactor power is above 25% of rated thermal power (RTP).

GE informed the affected licensees that their recent code calculations showed that during the PRFO transient, reactor pressure would fall below the current TS safety limit pressure. Depending upon the Low Pressure Isolation Setpoint (LPIS), the margin to the low pressure TS SL may not be adequate. GE recommended lowering the low pressure TS SL to 700 psia (685 psig), as supported by the expanded GEXL correlation applicability range for GE14 and GNF2 fuels that are currently installed in Columbia's reactor core.

GE advanced fuel designs have an NRC approved critical power correlation with a lower-bound pressure significantly below the 785 psig reactor steam dome pressure specified in TS SLs 2.1.1.1 and 2.1.1.2. Columbia proposes to utilize this fact and reduce the reactor steam dome pressure consistent with the approved lower-bound pressure for the GE fuel comprising Columbia's core. GE fuel utilizes the GEXL14 and GEXL17 critical power correlations, with an approved pressure range from 700 to 1400 psia (685 to 1385 psig). Columbia has determined that changing the pressure limits in TS SLs 2.1.1.1 and 2.1.1.2 to 685 psig as permitted by NEDC-32851P-A, Rev. 5, "GEXL14 Correlation for GE14 Fuel", (Reference 2) and NEDC-33292P, Rev. 3, "GEXL17 Correlation for GNF2 Fuel", (Reference 3), provides greater margin for the PRFO transient, such that the dome pressure will remain above the revised TS 2.1 safety limits.

Accordingly, pursuant to 10 CFR 50.90, Columbia hereby requests an amendment to its TS. The proposed amendment revises the reactor dome pressure from 785 psig to 685 psig in TS Reactor Core SLs 2.1.1.1 and 2.1.1.2 and TS Bases 2.1.1.1 and 2.1.1.2 to resolve the PRFO transient.

2.0 DETAILED DESCRIPTION

The proposed changes to the Technical Specifications would change the reactor dome pressure from 785 psig to 685 psig in TS SLs 2.1.1.1 and 2.1.1.2.

The referenced GESTAR II compliant reports and NRC approved correlations, References 2 and 3, are contained in fuel design information reports in accordance with GESTAR II Section 1.2.7. Therefore, these documents are included by reference.

3.0 BACKGROUND

References 2 and 3 document the expanded pressure range of GEXL correlations, for the currently installed fuel, GE14 and GNF2, in the Columbia reactor core. Reference 1 identified an AOO due to a PRFO transient that could potentially result in violation of the low pressure SLs in TS 2.1.1.1 and 2.1.1.2 as it is currently set at 785 psig.

Columbia reviewed the GEXL14 and GEXL17 correlations approved by the NRC in NEDC-32851P-A, Rev. 5 (Reference 2), NEDC-33292P, Rev. 3 (Reference 3) and NEDC-33270P, Rev. 5, "GNF2 Advantage Generic Compliance with NEDE- 24011-P-A (GESTAR II)", (Reference 4), and concluded that the GEXL14 and GEXL17 correlations apply to GE14 and GNF2 fuel, respectively. Since Columbia's core has both GE14 and GNF2 fuel, Columbia is proposing to change the current 785 psig reactor dome pressure limit in SLs 2.1.1.1 and 2.1.1.2 and associated TS Bases to 685 psig, using NRC approved GESTAR II methodology and approved documents (References 2, 3, 4 and 7). The proposed reduction in dome pressure is consistent with that used in the GESTAR II or NRC approved critical power correlations for GE14 and GNF2 fuel designs.

This change offers a greater pressure margin in TS 2.1.1.1 such that the reactor pressure remains above the proposed low pressure SL of 685 psig in the event of a PRFO transient. Thus the proposed change, in addition to compliance with the updated GEXL pressure range documented by Global Nuclear Fuel (GNF) for GE14 and GNF2 fuel designs, resolves the condition reported in Reference 1.

4.0 TECHNICAL EVALUATION

TS SLs are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs. SLs are set such that fuel cladding integrity is maintained and no significant fuel damage would occur if the SLs are not exceeded. In accordance with the Improved Standard Technical Specifications, Columbia specifies a SL 2.1.1.2 to require Minimum Critical Power Ratio (MCPR) be greater than the specified limit when reactor steam dome pressure and core flow are within the validity range of the GEXL correlation. GNF has updated the validity range of the GEXL14 and GEXL17 correlations via References 2, 3 and 4, which allows the pressure to be reduced to 685 psig instead of 785 psig for MCPR calculations to be valid.

Columbia relies upon the data used for development and verification of the GEXL critical power correlation based on NEDC-32851P-A, GEXL14 Correlation for GE14

Fuel, Rev. 5, (Refer to Table 2, Safety Evaluation Page 6) (Reference 2) and NEDC-33292P, GEXL17 Correlation for GNF2 Fuel, Rev. 3, (Refer to Table 3-1, Page 3-4) (Reference 3). These GE documents state that Stern Laboratory test data extended the pressure range from 700 to 1400 psia. This data were used to create the GEXL correlations. Therefore valid test data form the basis of the extended range of applicability for the GEXL correlation from 700 to 1400 psia. Therefore a greater pressure range is available for transients to demonstrate compliance with MCPR limits.

Thus, the proposed change offers a greater pressure margin in TS 2.1.1.1 for the PRFO transient than what is currently available such that the reactor pressure remains above the proposed low pressure SL of 685 psig.

The pressure regulator failure open event involves the failure of the pressure regulator in the open direction causing the turbine control valves and bypass valves to open. This causes the reactor to depressurize rapidly. When the reactor high water level setpoint is reached, the turbine stop (throttle) valves close and the reactor scrams on turbine stop (throttle) valve position. If the depressurization rate is slower, the reactor will depressurize down to the low pressure isolation setpoint which will cause the Main Steam Isolation Valves (MSIV) to close and the reactor will scram on MSIV position. The scram terminates the event and compliance with the SL is automatically restored as reactor power is quickly reduced to below 25% RTP.

Analysis of a PRFO transient is described in Columbia's updated Final Safety Analysis Report (FSAR), Section 15.1.3 (Reference 6 and 8). The sequence of events during the PRFO transient is as follows:

At the start of the event, a pressure regulator failure is assumed to occur with 130% steam flow demand signal. Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. The depressurization rate is large enough such that water level swells to the high water level setpoint, initiating main turbine and feedwater turbine trips. Position switches on the turbine stop (throttle) valves initiate reactor scram at 10% closed. After the turbine trip, the failed pressure regulator signals the bypass valves to allow maximum bypass flow. After the pressurization resulting from the turbine stop (throttle) valve closure, pressure again drops and continues to drop until turbine inlet pressure is below the low turbine pressure isolation setpoint and isolation is initiated with closure of the MSIVs (Reference 8). Note that this Columbia amendment request to lower the low pressure TS limit from 785 psig to 685 psig offers a greater range for pressure to reduce further while MSIVs are closing (Reference 6 and 8).

Depressurization rate has a proportional effect upon the voiding action in the core and the flashing in the vessel bulk water regions. If the rate is low enough, the water level may not swell to the high water level trip setpoint and the isolation will occur earlier when pressure at the turbine inlet decreases below the lower analytical limit, 795 psig. The reactor will scram as a result of the MSIV closure. Since power is being depressed as the pressure decreases due to additional voiding in the core, this transient is less severe when a slower depressurization rate is assumed. Therefore, the assumed Reactor Vessel Water Level - High, Level 8 trip provides the most restrictive margins on MCPR and peak vessel pressure (Reference 6).

The pressure reduction from this event causes creation of large voids in the core, which introduce negative reactivity and lower power. At lower pressure, the latent heat of vaporization is larger than at higher pressure. Therefore the critical heat flux is greater at lower pressure. Both of these factors cause the critical power ratio (critical power/actual power) to rise during depressurization. The fuel cladding integrity is never challenged because in pressure decrease events like this, fuel critical power is rising and therefore MCPR rises during the event. Therefore the PRFO transient has no adverse safety consequence as the critical power ratio actually rises during the event.

5.0 REGULATORY EVALUATION

5.1 Regulatory Requirement

Criterion 10 – Reactor Design

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 (GDC 10).

Compliance with the applicable 10 CFR 50, Appendix A, General Design Criteria (GDC) provides assurance that the integrity of the fuel and cladding will be maintained, thus preventing the potential for release of fission products during normal operation or AOOs. 10 CFR 50.36, Technical Specifications, requires safety limits to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. The proposed decrease in the reactor dome pressure for SLs in TS 2.1.1 follows the requirements cited and ensures the fuel cladding integrity is maintained.

5.2 Precedent

1. River Bend Station, Unit 1, License Amendment 182, December 11, 2014 (TAC NO. MF1948) (Reference 5). Like Columbia, River Bend is a Boiling Water Reactor (BWR). River Bend’s core makeup, at the time of the license amendment, was the same as Columbia’s, containing GNF2 and GE14 fuel.

5.3 No Significant Hazards Consideration

Columbia Generating Station (Columbia) requests a License Amendment to Technical Specification (TS) 2.1.1 to make the following change:

- Reduce the reactor dome pressure from 785 psig to 685 psig in TS 2.1.1.1 and 2.1.1.2.

Columbia has evaluated the proposed TS changes using the criteria in 10 CFR 50.92, and has determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration determination.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The change does not involve a modification of any plant hardware; the probability and consequence of the Pressure Regulator Failure Open (PRFO) transient are essentially unchanged. The reduction in the reactor dome pressure safety limit (SL) from 785 psig to 685 psig provides greater margin to accommodate the pressure reduction during the transient within the revised TS limit.

   The proposed change will continue to support the validity range for the correlations and the calculation of Minimum Core Power Ratio (MCPR) as approved. The proposed TS revision involves no significant changes to the operation of any systems or components in normal, accident or transient operating conditions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed reduction in the reactor dome pressure SL from 785 psig to 685 psig is a change based upon previously approved documents and does not involve changes to the plant hardware or its operating characteristics. As a result, no new failure modes are being introduced.

Therefore, the change does not introduce a new or different kind of accident from those previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The margin of safety is established through the design of the plant structures, systems, and components, and through the parameters for safe operation and setpoints for the actuation of equipment relied upon to respond to transients and design basis accidents. The proposed change in reactor dome pressure enhances the safety margin, which protects the fuel cladding integrity during a depressurization transient, but does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The change does not alter the behavior of plant equipment, which remains unchanged. The available pressure range is expanded by the change, thus offering greater margin for pressure reduction during the transient.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the above, Columbia concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATIONS

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. GE Energy-Nuclear, "10 CFR Part 21 Communication, Reportable Condition [21.21(d)] Potential to Exceed Low Pressure Technical Specification Safety Limit", March 29, 2005

2. NEDC-32851 P-A, Rev. 5, "GEXL 14 Correlation for GE14 Fuel", April 2011

3. NEDC-33292P, Rev 3, "GEXL 17 Correlation for GNF2 Fuel", June 2009

4. NEDC-33270P, Rev. 5, "GNF2 Advantage Generic Compliance with NEDE- 24011-P-A (GESTAR II)", May 2013

5. River Bend Station, Unit 1 - Issuance of Amendment Re: Technical Specification 2.1.1, "Reactor Core SLs", December 11, 2014 (TAC NO. MF1948)

6. Columbia Generating Station Updated Final Safety Analysis Report, Chapter 15, Accident Analysis, Amendment 63, December 2015

7. NEDE-24011-P-A and NEDE-24011-P-A-US, “General Electric Standard Application for Reactor Fuel (GESTAR II) and Supplement for the United States”, Revision 21, May 2015

8. GE-NE-208-08-039, “WNP-2 Power Uprate Transient Analysis Task Report, September 1993

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (MARK-UPS)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 685 psig and core flow ≥ 10% rated core flow:

The MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES
(FOR INFORMATION ONLY)

BASES

BACKGROUND (continued)

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures > ~~785~~ 685 psig and core flows > 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 E3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 E3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < ~~785~~ 685 psig is conservative.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level - Low Low, Level 2 and Low Low Low, Level 1 provide sufficient protection for level transients in all other MODES.

5. Main Steam Isolation Valve - Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the Nuclear Steam Supply System and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve - Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analyses of References 2 and 3, the Average Power Range Monitor Neutron Flux - High Function, along with the SRVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 5 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. The reactor scram resulting from an MSIV closure due to a Low Main Steam Line Pressure Isolation also ensures reactor power is less than 25% RTP before reactor pressure decreases below the Safety Limit 2.1.1 Low Pressure Limit of 685 psig.

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve - Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve - Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines (MSLs) must close in order for a scram to occur. In addition, certain combinations of valves closed in two lines will result in a half-scram.

The Main Steam Isolation Valve - Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the Main Steam Isolation Valve - Closure Function with eight channels in each trip system are required to be OPERABLE to

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Reactor Vessel Water Level - Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSL on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure transmitters with trip units that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure - Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 4). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs ~~prior to pressure decreasing~~ during the depressurization transient in order to maintain reactor steam dome pressure >~~below 785~~685 psig. The MSIV closure results in a scram~~, which results in a scram due to MSIV closure~~, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four sensors that are connected to the MSL header. The sensors are arranged such that, even though physically separated from each other, each sensor is able to detect low MSL pressure.

Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

PROPOSED TECHNICAL SPECIFICATIONS CLEAN PAGES

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

The MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.