July 18, 2006 TSTF-06-20

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

SUBJECT: TSTF-495, Revision 0, "Bases Change to Address GE Part 21 SC05-03"

Dear Sir or Madam:

Enclosed for NRC review is TSTF-495, "Bases Change to Address GE Part 21 SC05-03." The TSTF requests a meeting with the NRC at your earliest convenience to discuss this Traveler.

Any NRC review fees associated with the review of TSTF-495 should be billed to the Boiling Water Reactor Owners' Group.

TSTF-495 only affects the Technical Specification Bases and may be adopted by plants without requesting a license amendment from the NRC. Therefore, TSTF-495 is not a candidate for the Consolidated Line Item Improvement Process.

Should you have any questions, please do not hesitate to contact us.

Wesley Sparkman (PWROG/W)

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Enclosure

cc: Tim Kobetz, Technical Specifications Branch, NRC
David E. Roth, Technical Specifications Branch, NRC

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Bases Change to Address GE Part 21 SC05-03

NUREGs Affected: ☐ 1430 ☐ 1431 ☐ 1432 ☑ 1433 ☑ 1434

Classification: 2) Bases Only Change

Recommended for CLIIP?: No

Correction or Improvement: Correction

NRC Fee Status: Not Exempt

Benefit: Improves Bases

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1.0 Description

On March 29, 2005, GE issued Part 21 reportable condition SC05-03 under the provisions of 10 CFR 21.21(d) concerning BWR Standard Technical Specifications (TS) Safety limit (SL) 2.1.1.1. This Part 21 report described a previously analyzed event that could result in the Reactor Core Safety Limit being violated for a short period of time. This represents a Technical Specification compliance issue, but it was determined that this temporary condition does not represent a safety issue.

The Bases of 2.1.1.1 is revised to clarify the Applicability of the Safety Limit and eliminate the Technical Specification compliance issue.

2.0 Proposed Change

The Applicable Safety Analysis portion of the Reactor Core Safety Limit Bases is modified to clarify that the Safety Limit does not apply to momentary depressurization transients. The Applicable Safety Analysis portion of the Primary Containment Isolation Instrumentation specification is modified to delete a sentence that states that the Main Steam Line pressure - Low Function functions to ensure that Safety Limit 2.1.1.1 is not exceeded.

3.0 Background

On March 29, 2005, GE issued Part 21 reportable condition report SC05-03 under the provisions of 10 CFR 21.21(d) concerning BWR Standard Technical Specifications (TS) Safety limit (SL) 2.1.1.1, "Reactor Core SLs."

This SL states that when reactor steam dome pressure is less than 785 psig or when core flow is less than 10% of rated core flow, reactor thermal power shall be less than or equal to 25% of rated thermal power. When pressure and core flow are greater than the above listed values, Safety Limit 2.1.1.2 prohibits operation with a Minimum Critical Power Ratio (MCPR) less than [1.07] to prevent fuel cladding damage that could occur when a fuel bundle experiences Onset of Transition Boiling (OTB). The Critical Power Ratio (CPR) calculations performed by the fuel vendor and approved by the NRC are valid for reactor steam dome pressures greater than 785 psig and core flows greater than 10% of rated core flow. Consequently, below these values for dome pressure and core flow, SL 2.1.1.1 restricts operation to less than or equal to 25% of Rated Thermal Power where OTB conditions should not occur.

The BWR core is therefore protected from the type of fuel failure that could occur during OTB conditions by a combination of SL 2.1.1.1 and SL 2.1.1.2. That is, when valid analyses do not exist, by keeping the core power low enough such that the critical power which could result in OTB is not expected (SL 2.1.1.1) and, when valid analyses do exist, by requiring monitoring of the fuel bundle powers to ensure that no fuel bundles experience the critical power at which OTB would occur. Per the GE Part 21 Communication, SL 2.1.1.1 is "...intended to provide fuel integrity protection during startup conditions..."

The Part 21 report describes a revised accident analysis for the Pressure-Regulator Failure in the Open (PRFO) direction transient which results in a temporary pressure drop below 785 psig while reactor power is greater than 25% of Rated Thermal Power, resulting in a violation of Safety Limit 2.1.1.1. Under previous GE methodologies, the PRFO transient was terminated by the direct turbine trip and the subsequent reactor scram which resulted from the reactor water level swell following the PRFO. Specifically, in the postulated event the Electro Hydraulic Control (EHC) system pressure regulator fails in such a manner that it results in a demand to open the turbine steam admission valves, i.e., stop valves, control valves, and bypass valves. As a result, the reactor depressurizes which causes the formation of voids within the reactor core. The core voiding increases the reactor water level until it reaches the level of the main turbine trip setpoint. The turbine will trip, which in turn sends a direct signal (via the stop valve position switches) to the reactor protection system (RPS) and the reactor will automatically shut down, terminating the transient.

New analysis methodologies predict a different series of events: the PRFO occurs as before and the reactor depressurizes; however, the reactor level does not swell to the setpoint to cause a main turbine trip. Therefore, the transient is not terminated as quickly as the earlier methods predicted and the reactor depressurization continues until pressure reaches the setpoint of the Main Steam Line Isolation Valve (MSIV) Closure in MODE 1 containment isolation signal. The MSIV closure is a direct input, via their position switches, to the RPS. The reactor scrams and the transient is terminated. Under this series of events, the delay in the termination of the transient introduces the possibility for the reactor pressure to dip below 785 psig before the reactor has been shutdown. Therefore, the reactor power could still be above 25% when the reactor steam dome pressure is less than 785 psig, violating the Safety Limit. This condition would only be present for a matter of seconds before the RPS is initiated by the MSIV closure.

4.0 Technical Analysis

The revised accident analysis has not revealed a safety concern because the depressurization actually increases the margin to OTB. Critical Power is that fuel bundle power which could result in OTB conditions. OTB results in a "vapor blanket" over much of the fuel cladding which drastically reduces heat transfer between the fuel cladding and the flowing coolant in the channel. Consequently, the cladding temperature rises and fuel failure could result. In a BWR these conditions are monitored by the CPR which is a ratio of the critical power required to reach OTB conditions to the actual power of a specific fuel assembly (CPR = CP/AP where CP is the critical power and AP is the actual fuel assembly power). A ratio of one indicates that the actual assembly power is equal to the critical power and therefore OTB conditions could be occurring somewhere on the fuel cladding of that particular fuel assembly. Values greater than one indicate margin between actual power and critical power. The SR 2.1.1.2 minimum critical power ratio is [1.07].

In the case of the PRFO transient, or any other depressurization transient, the actual fuel assembly power is decreasing. The reduced pressure results in more voids being formed in the core, which results in reduced neutron moderation and lower power. At the same time, the critical power increases because as pressure decreases the heat of vaporization of water increases. Therefore, it takes more energy to change a certain mass of water from a liquid to a vapor and, consequently, the amount of power required to blanket a segment of fuel cladding in vapor will increase. Since critical power is increasing and actual power is decreasing, the critical power ratio is getting larger, which is the direction of more margin to the CPR.

This means that the reportable condition discovered by GE could result in the potential for a violation of SL 2.1.1.1 but does not represent an actual safety concern for a BWR plant.

In the Part 21 report, GE indicated that SL 2.1.1.1 was intended only to provide protection during startup conditions to insure that operation at less than 785 psig or less than 10% core flow while greater than 25% RTP would not occur. It was not intended to protect the core from possibly "passing through" these conditions during a depressurization transient. Indeed, as shown above, a depressurization would result in more margin to the OTB conditions.

Therefore, the SL 2.1.1.1 Bases are revised to state that the safety limit is not intended to apply to depressurization transients, such as PRFO, that may result in momentarily decreasing below 785 psig with thermal power above 25%, but which do not result in steady state operation under those conditions.

Additionally, GE stated in the Part 21 report that the Main Steam Line (MSL) Pressure - Low function should not be considered as a Limiting Safety System Setting to provide protection for SL 2.1.1.1 because the depressurization conditions do not challenge the physical barrier of the cladding that the SL is intended to protect. Therefore, the Bases for this Function are revised to eliminate a statement to that effect.

5.0 Regulatory Analysis

A regulatory analysis is not required for Bases changes.

6.0 Environmental Consideration

An environmental consideration is not required for Bases changes.

7.0 References

1. GE Energy - Nuclear, 10 CFR Part 21 Communication, SC05-03, "Potential to Exceed Low Pressure Technical Specifications Safety Limits," March 29, 2005.

Revision History

OG Revision 0 **Revision Status: Active**

Revision Proposed by: BWROG

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 26-Apr-06

Owners Group Comments
(No Comments)

Owners Group Resolution: Approved Date: 10-May-06

TSTF Review Information

TSTF Received Date: 19-Jun-06 Date Distributed for Review 19-Jun-06

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 17-Jul-06

NRC Review Information

NRC Received Date: 18-Jul-06

Affected Technical Specifications

S/A 2.1.1 Bases Reactor Core Safety Limits

Ref. 2.1.1 Bases	Reactor Core Safety Limits
S/A 3.3.6.1 Bases	Primary Containment Isolation Instrumentation

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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 an 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 the 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.

2.1.1.1a Fuel Cladding Integrity [General Electric Company (GE) Fuel]

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 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 x 10^3 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 x 10^3 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 psig is conservative.

SL 2.1.1.1 prohibits operation at a THERMAL POWER level greater than 25% RTP when reactor steam dome pressure is less than 785 psig or core flow is less than 10% of rated core flow. Certain depressurization events can result in transient operation at less than 785 psig when THERMAL POWER is greater than 25% RTP. The pressure regulator failure-open (PRFO) event is one such depressurization transient. In depressurization events, critical bundle power increases and actual bundle power decreases, which increases the Critical Power Ratio (Ref. 3). This provides more thermal margin. As a result, SL 2.1.1.1 does not apply during depressurization transients, such as PRFO, that may result in momentarily decreasing below 785 psig with THERMAL POWER above 25%, but do not result in steady state operation under those conditions.

2.1.1.1b Fuel Cladding Integrity [Advanced Nuclear Fuel Corporation (ANF) Fuel]

The use of the XN-3 correlation is valid for critical power calculations at pressures > 580 psig and bundle mass fluxes > 0.25 x 10^6 lb/hr-ft^2 (Ref. ~~3~~4). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow is > 30 x 103 lb/hr. For the ANF 8x8 fuel design, the minimum bundle flow is > 28 x 103 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are

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such that the mass flux is always > 0.25 x 106 lb/hr-ft2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 106 lb/hr-ft2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

SL 2.1.1.1 prohibits operation at a THERMAL POWER level greater than 25% RTP when reactor steam dome pressure is less than 785 psig or core flow is less than 10% of rated core flow. Certain depressurization events can result in transient operation at less than 785 psig when THERMAL POWER is greater than 25% RTP. The pressure regulator failure-open (PRFO) event is one such depressurization transient. In depressurization events, critical bundle power increases and actual bundle power decreases, which increases the Critical Power Ratio (Ref. 3). This provides more thermal margin. As a result, SL 2.1.1.1 does not apply during depressurization transients, such as PRFO, that may result in momentarily entering the conditions prohibited by the SL, but do not result in steady state operation under the prohibited conditions.

2.1.1.2a MCPR [GE Fuel]

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the

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procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the XN-3 critical power correlation. Reference ~~3~~ 4 describes the methodology used in determining the MCPR SL.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

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SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. ~~4~~5). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10. 2. NEDE-24011-P-A (latest approved revision). 3. 10 CFR 21 Reportable Condition Notification: "Potential to Exceed Low Pressure Technical Specification Safety Limit," MFN 05-021, March 29, 2005. ~~3~~4. XN-NF524(A), Revision 1, November 1983. ~~4~~5. 10 CFR 100.

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Main Steam Line Isolation

1.a. Reactor Vessel Water Level - Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. 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 MSLs 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 level transmitters 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 100 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/hr 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

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) 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 below 785 psig, 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 transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter 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.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

<u>1.c. Main Steam Line Flow – High</u>

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 1). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

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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 an 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.

2.1.1.1a Fuel Cladding Integrity [General Electric Company (GE) Fuel]

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 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 x 10^3 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 x 10^3 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 psig is conservative.

SL 2.1.1.1 prohibits operation at a THERMAL POWER level greater than 25% RTP when reactor steam dome pressure is less than 785 psig or core flow is less than 10% of rated core flow. Certain depressurization events can result in transient operation at less than 785 psig when THERMAL POWER is greater than 25% RTP. The pressure regulator failure-open (PRFO) event is one such depressurization transient. In depressurization events, critical bundle power increases and actual bundle power decreases, which increases the Critical Power Ratio (Ref. 3). This provides more thermal margin. As a result, SL 2.1.1.1 does not apply during depressurization transients, such as PRFO, that may result in momentarily decreasing below 785 psig with THERMAL POWER above 25%, but do not result in steady state operation under those conditions.

2.1.1.1b Fuel Cladding Integrity [Advanced Nuclear Fuel Corporation (ANF) Fuel]

The use of the XN-3 correlation is valid for critical power calculations at pressures > 580 psig and bundle mass fluxes > 0.25 x 10^6 lb/hr-ft^2 (Ref. ~~3~~4). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow

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APPLICABLE SAFETY ANALYSES (continued)

is > 30 x 10^3 lb/hr. For the ANF 8x8 fuel design, the minimum bundle flow is > 28 x 10^3 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

SL 2.1.1.1 prohibits operation at a THERMAL POWER level greater than 25% RTP when reactor steam dome pressure is less than 785 psig or core flow is less than 10% of rated core flow. Certain depressurization events can result in transient operation at less than 785 psig when THERMAL POWER is greater than 25% RTP. The pressure regulator failure-open (PRFO) event is one such depressurization transient. In depressurization events, critical bundle power increases and actual bundle power decreases, which increases the Critical Power Ratio (Ref. 3). This provides more thermal margin. As a result, SL 2.1.1.1 does not apply during depressurization transients, such as PRFO, that may result in momentarily entering the conditions prohibited by the SL, but do not result in steady state operation under the prohibited conditions.

2.1.1.2a MCPR [GE Fuel]

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical

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APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2b MCPR [ANF Fuel]

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the XN-3 critical power correlation. Reference ~~3~~ 4 describes the methodology used in determining the MCPR SL.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated

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cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. ~~4~~5). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

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

2. NEDE-24011-P-A, (latest approved revision).

3. 10 CFR 21 Reportable Condition Notification: "Potential to Exceed Low Pressure Technical Specification Safety Limit," MFN 05-021, March 29, 2005.

~~3~~4. XN-NF524(A), Revision 1, November 1983.

~~4~~5. 10 CFR 100.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Main Steam Line Isolation

1.a. Reactor Vessel Water Level - Low Low Low, Level 1

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. 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 level transmitters 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 100 limits.

This Function isolates the Group 1 and 5 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. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. ~~In addition,~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~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 below 785 psig, 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 transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter 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.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow - High

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is directly assumed in the analysis of the main stream line break (MSLB) accident (Ref. 1). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one steam line would be able to detect the high flow. Four channels of Main Steam Line Flow - High Function for each unisolated MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.