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SOUTHERN COMPANY

March 24, 2014

Docket Nos.: 50-321
50-366

NL-13-2035

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

Edwin I. Hatch Nuclear Plant
License Amendment Request to Reduce the Reactor Steam Dome Pressure
Specified in the Reactor Core Safety Limits

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90 Southern Nuclear Operating Company (SNC) is submitting a request for an amendment to the Technical Specifications (TS) for Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2.

The amendment will revise the HNP Unit 1 and Unit 2 Technical Specifications (TS) Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. This change to TS Section 2.1.1 became necessary as a result of General Electric (GE) Part 21 report SC05-03, Potential to Exceed Low Pressure Technical Specification Safety Limit. This change is consistent with the Nuclear Regulatory Commission (NRC) approved pressure range for the critical power correlations applied to the fuel types in use at HNP Units 1 and 2.

SNC requests approval of the proposed license amendments by March 15, 2015. The proposed changes would be implemented within 90 days of issuance of the amendment.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Mr. C. R. Pierce states he is Regulatory Affairs Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/RMJ/

Sworn to and subscribed before me this 24 *day of* March*, 2014.*

Laura L. Crumpton
Notary Public

My commission expires: 10/8/2017

Enclosures:
1. Description and Assessment
2. Marked Technical Specification Pages
3. Clean Typed Technical Specification Pages
4. Marked Technical Specification Bases Pages

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. D. R. Vineyard, Vice President – Hatch
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U. S. Nuclear Regulatory Commission
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State of Georgia
Mr. J. H. Turner, Environmental Director Protection Division

Edwin I. Hatch Nuclear Plant
License Amendment Request to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

Enclosure 1

Description and Assessment

1.0 INTRODUCTION

The Edwin I. Hatch Nuclear Plant (HNP) Technical Specifications (TS) Section 2.1.1 is revised to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. This change to TS Section 2.1.1 became necessary as a result of General Electric (GE) Part 21 report SC05-03, Potential to Exceed Low Pressure Technical Specification Safety Limit. This change is consistent with the Nuclear Regulatory Commission (NRC) approved pressure range for the critical power correlations applied to the fuel types in use at HNP Units 1 and 2.

The proposed changes are described in detail in Section 2.0.

2.0 PROPOSED CHANGE

Reduce the reactor steam dome pressure specified within Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 from 785 to 685 psig. The Reactor Core Safety Limits for HNP Unit 1 would then read:

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

THERMAL POWER shall be ≤ 24% RTP.

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

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

The Unit 2 Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 would then read:

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

THERMAL POWER shall be ≤ 24% RTP.

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

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

A marked-up copy of the proposed changes to the TS Reactor Core Safety Limits is provided in Enclosure 2. Clean-typed TS pages are provided in Enclosure 3. Enclosure 4 provides a copy of the associated marked-up TS Bases pages. The Bases changes will be issued in accordance with HNP Specification 5.5.11, "Technical Specification (TS) Bases Control Program," following NRC approval.

3.0 HISTORY

On March 29, 2005, GE submitted a 10 CFR Part 21 notification (Reference 1) identifying that, as a result of applying improved methodologies for licensing basis transient analyses, the anticipated operational occurrence (AOO) Pressure Regulator Failure Maximum Demand (Open) (PRFO) had been identified as an event in which Reactor Core Safety Limit 2.1.1.1 could potentially be violated. During the transient, the expected sequence of events predicted by the computer models could potentially change, and based upon this, the reactor steam dome pressure could momentarily decrease below 785 psig while thermal power was above the plant-specific thermal power limit specified in the Technical Specification 2.1.1.1, violating Reactor Core Safety Limit 2.1.1.1.

GE indicated that the approved methodology for modeling AOOs had evolved from REDY, to ODYN, to TRACG. Reactor depressurization transients, such as PRFO, are non-limiting for fuel cladding integrity because critical power ratio (CPR) increases during the PRFO event, and are not typically included in the scope of cycle-specific reload evaluations. GE determined that REDY, ODYN, and TRACG all show the CPR increasing during the PRFO transient, and hence fuel cladding integrity not being threatened,[1] and that the difference in reactor level swell predicted by REDY, versus ODYN and TRACG, can impact the predicted plant response to the PRFO.

GE indicated within the 10 CFR Part 21 notification letter that no clear compensatory action can be defined to appropriately mitigate this vulnerability, and since the condition does not challenge the physical barrier that the Safety Limit intends to protect (i.e., the fuel cladding integrity), there is no safety basis for a compensatory action. While this condition had been determined by GE to not involve an actual safety hazard, the potential for violation of a Reactor Core Safety Limit had been identified, and restoration to comply with the safety limit is required for the PRFO event. As a consequence, SNC is revising the reactor steam dome pressure TS Safety Limit consistent with the NRC approved pressure range of critical power correlations for the current and anticipated HNP fuel designs.[2]

4.0 BACKGROUND

A discussion providing background on the Reactor Core Safety Limits and a summary of the PRFO transient scenario considering the change in computer analysis codes is provided below.

4.1 Background on the Reactor Core Safety Limits

TS Safety Limits are specified to ensure that specified acceptable fuel design limits (SAFDLS) are not violated during steady state operation, normal operational transients,

[1] The Minimum Critical Power Ratio (MCPR) Safety Limit specified in Reactor Core Safety Limit 2.1.1.1 is established to protect fuel cladding integrity.

[2] The GE14 fuel type is currently in use at both HNP units, with the exception of four Westinghouse Lead Use Assemblies (LUAs) currently in the Unit 1 core. These LUAs are modeled as GE14, and are placed in non-limiting locations. Commencing Unit 1 Cycle 28 (spring 2016 startup) and Unit 2 Cycle 24 (spring 2015 startup), SNC intends to transition from GE14 fuel to GNF2 fuel. Both fuel types (GNF2 and GE14) have an approved pressure range from 700 to 1400 psia.

and anticipated operational occurrences. The Reactor Core Safety Limits are set such that fuel cladding integrity is maintained. No significant fuel damage is calculated to occur if the Safety Limits are not violated.

The Boiling Water Reactor (BWR) core is protected from the type of fuel failure that could occur during the Onset of Transition Boiling (OTB) by a combination of Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. Reactor Core Safety Limit 2.1.1.1 states that when the reactor steam dome pressure is less than 785 psig or when core flow is less than 10% of rated core flow, the reactor thermal power shall be less than or equal to 24% rated thermal power (RTP). When reactor pressure and core flow are greater than these specified values, Reactor Core Safety Limit 2.1.1.2 prohibits operation with a MCPR Safety Limit less than the values specified to prevent fuel cladding damage that could occur when a fuel assembly experiences the OTB.

As discussed in Section B 2.1.1 of the TS Bases, for operation at low pressures or low flows, such as during startup, an alternate basis is used to provide fuel cladding integrity protection. Reactor Core Safety Limit 2.1.1.1 precludes the need for CPR calculations when reactor steam dome pressure is less than 785 psig or when core flow is less than 10% rated core flow by ensuring that reactor power would remain well below the fuel assembly critical power for the conditions at which CPR calculations are not performed (i.e., Safety Limit 2.1.1.1 limits thermal power to less than or equal to 24% RTP to ensure OTB conditions will not occur).

4.2 Pressure Regulator Failure Maximum Demand (Open) Transient Analysis Background

The GE Part 21 report describes a revised transient analysis scenario for the PRFO event. A change in the predicted series of events for this transient was identified based upon a change in computer codes and the predicted results of this event.

Previous evaluations using the REDY methodology indicated the transient would be terminated by direct turbine trip and subsequent reactor scram resulting from the reactor water level swell following the event. Specifically, for the postulated event, the pressure regulator system fails in such a manner that a demand occurs to open the turbine steam admission valves, i.e., turbine stop valves (TSVs), turbine control valves, and turbine bypass valves. As a result, the reactor depressurization causes the formation of voids within the reactor core. The core voiding increases the reactor water level until the level reaches the main turbine trip (level) setpoint. The turbine trips, in turn sending a direct signal (via the TSV position switches) to the reactor protection system (RPS) resulting in the reactor automatically shutting down, terminating the transient.

A somewhat different series of events is predicted when the event was analyzed with improved transient methods. The transient occurs as before and the reactor depressurizes; however, the reactor level does not swell to the setpoint to cause a main turbine trip. Level swell is difficult to predict and the level swell portion of transient models have larger uncertainties than other portions of the transient models. In this case the depressurization could be terminated by Main Steam Isolation Valve

(MSIV) closure at the low-pressure isolation setpoint (LPIS)[3]. This results in the transient not being terminated as quickly as the earlier methods predicted. Reactor depressurization continues to occur until the pressure decreases to the MSIV closure (in MODE 1) containment isolation signal setpoint. The MSIV closure is a direct input, via position switches, to the RPS. The reactor scrams and the transient is terminated.

However, under this series of events, the delay in termination of the transient introduces the possibility for reactor pressure to decrease below the 785 psig TS limit while reactor power is still greater than 24% of RTP. Depending upon the plant-specific response to a PRFO event, including the value of the LPIS and the closure rate for the MSIV, reactor steam dome pressure could decrease to below 785 psig for a few seconds while thermal power exceeds 24% of RTP, which would violate the conditions in Reactor Core Safety Limit 2.1.1.1. Reactor Core Safety Limit 2.1.1.1, however, is overly conservative with respect to this event. During this event, CPR continues to increase and therefore does not threaten fuel cladding integrity. The pressure decrease, though, could result in violating the value specified in the safety limit specification, while having no actual safety significance.

5.0 TECHNICAL ANALYSIS

The purpose of Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 is to protect fuel cladding integrity. The fuel cladding integrity safety limit (MCPR Safety Limit) is defined as the CPR for which more than 99.9% of the fuel rods in the core are expected to avoid the OTB, considering the power distribution within the core and all uncertainties. The safety limit is set such that no significant fuel damage is calculated to occur if the limit is not violated. It is determined using a statistical model that combines the uncertainties in operating parameters and procedures used to calculate critical power.

The probability of the occurrence of OTB is determined using approved critical power correlations. Each fuel vendor has developed correlations valid over specified pressure and flow ranges (mass flow rates) that are approved by the NRC. The critical power correlations for some advanced fuel designs have received NRC approval down to a lower pressure than those approved previously. The lower-bound of the extended pressure ranges for these advanced fuel designs can be used to establish a lower reactor steam dome pressure than the 785 psig value currently specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. SNC proposes to utilize the fact that the GE14 and GNF2 fuel, comprising the HNP Unit 1 and Unit 2 cores[4,5], utilize critical power correlations that have an approved pressure range from 700 to 1400 psia[6]. Revising the Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 reactor steam dome

[3] The Main Steam Line Pressure – Low Function (Function 1.b in TS Table 3.3.6.1-1) for HNP Units 1 and 2 corresponds to the LPIS in the 10 CFR Part 21 notification.
[4] Commencing Unit 1 Cycle 28 (spring 2016 startup) and Unit 2 Cycle 24 (spring 2015 startup), SNC intends to transition from GE14 fuel to GNF2 fuel.
[5] In addition, there are currently four Westinghouse Optima 2 Lead Use Assemblies (LUAS) in the Unit 1 core. These LUAs are modeled as GE14, and are placed in non-limiting locations.
[6] In accordance with 10 CFR 50.59, only fuel which has an NRC approved critical power correlation with a lower-bound pressure less than or equal to the reactor steam dome pressure specified in the safety limit may be loaded into the core.

pressure from 785 to 685 psig resolves the reported 10 CFR Part 21 condition concerning the potential to violate Reactor Core Safety Limit 2.1.1.1 during a PRFO transient.

5.1 No Impact on the Main Steam Line Pressure- Low Function

The Main Steam Line Pressure- Low Function is directly assumed in the analysis of the pressure regulator failure. The Allowable Value for Main Steam Line Pressure- Low (TS Table 3.3.6.1-1, Function 1.b) for HNP is greater than or equal to 825 psig for Units 1 and 2. The current Nominal Trip Setpoint for this function is 864 psig and 855 psig for Units 1 and 2, respectively.

For the PRFO event, closure of the Main Steam Isolation Valves (MSIVs) ensures that the reactor pressure vessel temperature rate of change TS cooldown limit (100°F/hr) is not reached. Also, as discussed in the TS Bases, this Function supports actions to ensure that Reactor Core Safety Limit 2.1.1.1 is not violated. This Function is described as closing the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to less than 24% of RTP. The proposed change to reduce the reactor steam dome pressure in Safety Limit 2.1.1.1 and 2.1.1.2 from 785 psig to 685 psig does not change the function of the Main Steam Line Pressure – Low scram, but does increase the margin between the setpoint and the safety limit the setpoint is protecting.

No changes are required or proposed to any instrumentation settings associated with the Main Steam Line Pressure- Low Function, including the TS Allowable Value. The TS Bases description for this function is revised to indicate the change in reactor steam dome pressure.

The trip on low main steam line pressure will occur as previously specified, at the assumed instrument settings discussed above. These Main Steam Line Pressure – Low Function setpoints provide added assurance that with the revised reactor steam dome pressure of 685 psig, that Reactor Core Safety Limit 2.1.1.1 would not be violated.

5.2 Application of these Extended Pressure Ranges to Resolution of the 10 CFR Part 21 Concerning the Potential for Violation of Reactor Core Safety Limit 2.1.1.1 for a Pressure Regulator Failure Maximum Demand (Open) Transient

As discussed previously, each fuel vendor has critical power correlation(s) which are valid over established pressure ranges and flows (mass flow rates), approved by the NRC, which may or may not be fuel design specific. These critical power correlations have become increasingly fuel design dependent as advanced fuel designs evolved. This has resulted in an extension of the NRC approved pressure range to lower pressures as additional test data became available to demonstrate the validity of revised or new correlation(s) for performance of critical power calculations.

These reduced lower-bound pressures associated with the newer critical power correlations, as discussed previously, can be utilized to reduce the reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2, consistent with the NRC approved pressure range for these correlation(s). Lowering the reactor steam dome pressure specification in this fashion provides margin to ensure Reactor Core Safety Limit 2.1.1.1 is not violated and resolves this 10 CFR Part 21 issue involving a potential to violate the low pressure TS Safety Limit during a PRFO transient.

SNC has determined that with the value of 685 psig proposed for the reactor steam dome pressure, a PRFO transient would not result in a violation of Reactor Core Safety Limit 2.1.1.1. Since this approach follows, and is consistent with, the NRC-approved pressure range for the critical power correlations applied to the GE14 and GNF2 fuel designs, it addresses the potential for violating the TS Safety Limits discussed in the 10 CFR Part 21 report SC05-03 issued by GE.

5.3 Conclusion

In summary, it has been identified that a PRFO transient could potentially violate Reactor Core Safety Limit 2.1.1.1. Reducing the reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 from the current value of 785 psig to 685 psig, in accordance with the NRC approved lower-bound pressure for the critical power correlations applicable to HNP fuel (GE14 and GNF2) eliminates the potential for violating the Safety Limits during this event. This TS change resolves an industry 10 CFR Part 21 condition, which identified that during a PRFO transient Reactor Core SL 2.1.1.1 might be violated.

6.0 REGULATORY ANALYSIS

6.1 Applicable Regulatory Requirements

10 CFR 50.36, "Technical Specifications," provides the regulatory requirements for the content required in the TSs. As stated in 10 CFR 50.36, the TSs will include Safety Limits for nuclear reactors which are stated to be "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down."

The proposed TS change revises the reactor steam dome pressure stated in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 to remove the potential to violate Reactor Core Safety Limit 2.1.1.1 during a Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient. The HNP Unit 1 construction permit was received under the 70 general design criteria issued for comment in July 1967. The HNP Unit 2 construction permit was received under the current 10 CFR 50 Appendix A General Design Criteria (GDC).

The applicable 70 draft AEC General Design Criterion (AEC-GDC) and applicable GDC for Unit 1 and Unit 2, respectively, are listed below, along with a discussion regarding how this criteria is still met.

- AEC-GDC 6 – Reactor Core Design (Category A)

  The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of off-site power.

- GDC 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.

SNC has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. As long as the core pressure and flow are within the range of validity of the specified critical power correlation, the proposed reactor steam dome pressure change to Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 will continue to ensure that 99.9% of the fuel rods in the core are expected to avoid the onset of boiling transition. This satisfies the requirements of AEC-GDC 6 and GDC 10 regarding specified acceptable fuel design limits, and continues to assure that the underlying criteria of the safety limit is met. Based on this, there is reasonable assurance that the health and safety of the public, following approval of this TS change, is unaffected.

6.2 No Significant Hazards Determination

SNC has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the HNP Units 1 and 2 in accordance with the proposed amendment presents no significant hazards. SNC's evaluation against each of the criteria in 10 CFR 50.92 follows.

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

   Response: No.

   The proposed change to the reactor steam dome pressure in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 does not alter the use of the analytical methods used to determine the safety limits that have been previously reviewed and approved by the NRC. The proposed change is in accordance with an NRC approved critical power correlation methodology, and as such, maintains required safety margins. The proposed change does not adversely affect accident initiators or precursors, nor does it alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained.

The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not require any physical change to any plant SSCs nor does it require any change in systems or plant operations. The proposed change is consistent with the safety analysis assumptions and resultant consequences.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

There are no hardware changes nor are there any changes in the method by which any plant systems perform a safety function. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

The proposed change does not introduce any new accident precursors, nor does it involve any physical plant alterations or changes in the methods governing normal plant operation. Also, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in the safety analysis.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

Margin of safety is related to confidence in the ability of the fission product barriers (fuel cladding, reactor coolant system, andprimary containment) to perform their design functions during and following postulated accidents. Evaluation of the 10 CFR Part 21 condition by General Electric determined that since the Minimum Critical Power Ratio improves during the PRFO transient, there is no decrease in the safety margin and therefore there is not a threat to fuel cladding integrity.

The proposed change to Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 is consistent with and within the capabilities of the applicable NRC approved critical power correlation for the fuel designs in use at HNP Units 1 and 2. No setpoints at which protective actions are initiated are altered by the proposed change. The proposed change does not alter the manner in which the safety limits are determined. This change is consistent with plant design and does not change the TS operability requirements; thus, previously evaluated accidents are not affected by this proposed change.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, SNC has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

7.0 ENVIRONMENTAL EVALUATION

A review has determined that the proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change 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 change.

8.0 REFERENCES

1. GE letter to the NRC GENE SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," to the NRC informing them of this reportable condition pursuant to 10 CFR 21, dated 3/29/2005.

Edwin I. Hatch Nuclear Plant
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Enclosure 2

Marked Technical Specification 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 < ~~785~~ 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

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

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 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 (RCS) 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.

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

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

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License Amendment Request to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

Enclosure 3

Clean Typed Technical Specification 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 ≤ 24% RTP.

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

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

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

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

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

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

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Enclosure 4

Marked Technical Specification Bases Pages

BASES

BACKGROUND (continued)	to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant. 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 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 SL.

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 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 24% RTP for reactor pressure < ~~785~~ 685 psig is conservative.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure with the reactor at power 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, 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~~ 685 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 24% RTP.)

The MSL low pressure signals are initiated from four switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each switch 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 steam line break (MSLB) (Ref. 2). 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 50.67 limits.

(continued)

BASES

BACKGROUND (continued)

to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

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 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 Safety 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 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 24% RTP for reactor pressure < ~~785~~ 685 psig is conservative.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

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