Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. FitzPatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315-349-6024 Fax 315-349-6480

Lawrence M. Coyle
Site Vice President - JAF

JAFP-13-0115
October 8, 2013

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit

James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

References:

1. GE Energy - Nuclear, 10 CFR Part 21 Communication, Potential to Exceed Low Pressure Technical Specification Safety Limit, SC05-03, dated March 29, 2005
2. Letter, USNRC to Entergy, Grand Gulf Nuclear Station Unit 1- Issuance of Amendment RE: Extended Power Uprate (TAC No. ME4679), ML121210020, dated July 18, 2012

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) is submitting a request for an amendment to the Technical Specifications (TS) for the James A. FitzPatrick Nuclear Power Plant (JAF).

The proposed amendment modifies the TS to reduce the reactor pressure associated with the Reactor Core Safety Limit from 785 psig to 685 psig in TS 2.1.1.1 and TS 2.1.1.2. The proposed change would address the potential to not meet the pressure/Thermal Power/MCPR TS safety limit during a Pressure Regulator Failure-Maximum Demand (Open) (PRFO) transient as reported by General Electric Nuclear Energy in Reference 1. The proposed changes are consistent with similar change approved for Grand Gulf Nuclear Station in pages 324-325 of Reference 2.

Attachment 1 provides a description and assessment of the proposed changes. Attachment 2 provides the existing TS pages marked up to show the proposed changes. Attachment 3 provides revised (clean) TS pages. Attachment 4 contains the proposed TS Bases changes for information only.

Approval of the proposed amendment is requested by October 8, 2014. Once approved, the amendment shall be implemented within 30 days.

Entergy has reviewed the proposed amendment in accordance with 10 CFR 50.92 and concludes it does not involve a significant hazards consideration. In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated New York State Official.

No commitments are contained in this submittal. If you should have any questions regarding this submittal, please contact Mr. Chris M. Adner, Licensing Manager, at 315-349-6766.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 8, 2013.

Sincerely,

Lawrence M. Coyle
Site Vice President

LMC/CMA/mh

Attachments: 1. Description and Assessment
2. Proposed Technical Specification Changes (Markup)
3. Revised Technical Specification Changes (Clean)
4. Proposed Technical Specification Bases Changes (Information Only)

cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
2100 Renaissance Boulevard, Suite 100
King of Prussia, PA 19406-2713

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
James A. FitzPatrick Nuclear Power Plant
P.O. Box 136
Lycoming, NY 13093

Mr. Mohan Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Ms. Bridget Frymire
New York State Department of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Mr. Francis J. Murray Jr., President
New York State Energy and Research Development Authority
17 Columbia Circle
Albany, NY 12203-6399

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Attachment 1

Description and Assessment

(6 Pages)

Description and Assessment

1. DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating License DPR-59 for James A. FitzPatrick Nuclear Power Station (JAF).

The proposed change would revise the JAF Technical Specifications (TS) to reduce the reactor pressure associated with the Reactor Core Safety Limits (SLs) from 785 psig to 685 psig in TS 2.1.1.1 and TS 2.1.1.2. The proposed changes would address the potential to not meet the pressure/Thermal Power/MCPR TS safety limit during a Pressure Regulator Failure-Maximum Demand (Open) (PRFO) transient as reported by General Electric Nuclear Energy (GE) [Reference 1]. The proposed change is consistent with similar change approved for Grand Gulf Nuclear Station in pages 324-325 [Reference 2].

On March 29, 2005, GE issued a Safety Communication (SC 05-03) [Reference 1] in accordance with 10 CFR 21.21(d). SC 05-03 documented a reportable condition for a potential to not meet the low pressure/Thermal Power/MCPR TS SL. GE identified an unanalyzed condition where a PRFO may cause a TS SL to be violated since reactor pressure could drop below the current JAF TS SL 2.1.1.1 and TS 2.1.1.2 limit of 800 psia (785 psig) for a few seconds while reactor power is above 25% of rated thermal power. GE identified that even plants with a main steam isolation valve (MSIV) low pressure isolation setpoint ≥ 785 psig may experience an anticipated operational occurrence (AOO) that potentially could violate the SL. GE considers a PRFO to be an AOO.

GE informed the affected licensees that recent calculations showed that during the PRFO transient, reactor pressure could fall below the TS Reactor Core safety limits. Depending upon the low pressure isolation setpoint (LPIS), the margin to the low pressure TS SLs may not be adequate. GE recommended lowering the low pressure TS safety limit to 700 psia (685 psig), as supported by the expanded GEXL correlation (used by GE to perform Critical Power Ratio, CPR, calculations) applicability range for GNF2 fuel that is currently operated in the JAF reactor core.

Entergy reviewed the GEXL17 correlation in NEDC-33292P, Rev. 3 [Reference 3], for GNF2 fuel and has determined that it is applicable to the GNF2 fuel used in the JAF core. The proposed reduction in the current 785 psig reactor pressure limit in TS SL 2.1.1.1 and TS 2.1.1.2 to 685 psig is within the range of applicable pressures in the GEXL correlations.

The GEXL17 correlation for GNF2 fuel, NEDC-33292P, is approved for use per NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel (GESTAR II)" by reference. NEDE-24011 specifically states:

> *Fuel design compliance with the fuel licensing acceptance criteria constitutes USNRC acceptance and approval of the fuel design without specific USNRC review. The fuel licensing acceptance criteria are presented in the subsections that follow.*

The fuel licensing acceptance criteria for a new critical power correlation can be found in GESTAR II subsection 1.1.7. NEDC-33270P [Reference 4] documents that GESTAR II subsection 1.1.7 criteria for a new correlation are met. Therefore, per GESTAR II, the GEXL17 correlation is approved for use.

Entergy has determined that changing the pressure limit in TS 2.1.1.1 and TS 2.1.1.2 to 685 psig provides greater margin for the PRFO transient, such that reactor pressure is expected to remain above the revised TS 2.1.1.1 and TS 2.1.1.2 limit if the transient were to occur.

Description and Assessment

2. Detailed Description

The following change is proposed to TS SL 2.1.1 “Reactor Core SLs”:

Current TS 2.1.1.1	Proposed TS 2.1.1.1
With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow: *THERMAL POWER shall be ≤ 25% RTP.*	*With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:* *THERMAL POWER shall be ≤ 25% RTP.*
Current TS 2.1.1.2 *With the reactor steam dome pressure is ≥ 785 psig and core flow ≥ 10% of rated flow:* *MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.*	Proposed TS 2.1.1.2 *With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% of rated flow:* *MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.*

3. TECHNICAL EVALUATION

The Pressure Regulator and Turbine-Generator Control System protects the turbine from overpressure or excessive speed by controlling steam flow and pressure. The main turbine generator controls work in conjunction with the Nuclear Steam System controls to maintain essentially constant reactor pressure and limit reactor transients during load variations. During normal planned operation, the steam admitted to the turbine is controlled by the pressure regulator which maintains essentially constant pressure at the turbine inlet, thus controlling reactor vessel pressure. This control scheme forces turbine generator output to follow reactor steam output. Changing recirculation flow or moving control rods changes the steam flow available from the reactor. The change in recirculation flow or rod motion directly changes the reactor steaming rate, and the controlling pressure regulator reacts by appropriately opening or closing the turbine admission or bypass valves. Thus, the turbine and/or main condenser absorbs any change in reactor power.

Two pressure regulators are provided, one intended for use as a backup to the controlling regulator, either one of which can be used for control purposes with the unit at rated turbine inlet pressure. The controlling pressure regulator is used to control both the turbine admission valves and the turbine bypass valves.

Normally the bypass valves are held closed and the pressure regulator controls the admission valves using all the steam production to make electrical power. If the speed controls or electrical load demand a reduced steam flow to the turbine, the pressure regulator functions to open the turbine bypass valves to send the excess steam flow to the main condenser. The backup pressure regulator functions to assure pressure control in the event of failure of the controlling regulator; its setpoint is normally a few psi above that of the controlling regulator.

PFRO transient analysis, FSAR Section 14.5.5.1, a potential system pressure increases or decreases can be produced by pressure regulator failure.

If the controlling regulator fails in a closed direction, the backup regulator takes over control of the turbine admission valves, preventing a serious transient. The disturbance is mild, similar to a pressure setpoint change, and no significant thermal margin reductions occur. If either the controlling regulator or the backup regulator fails in an open direction, the turbine admission valves can be fully opened and the turbine bypass valves can be partially opened. This potential reactor depressurization threatens to impose serious stresses on the Reactor Coolant System.

A maximum flow limit of 110 percent is imposed at the turbine controls to limit the total valve opening. The main steam isolation valves are closed when pressure at the turbine drops, in order to shut off this uncontrolled release of steam and scram the reactor. Isolation valve closure stops the vessel depressurization and produces a normal shutdown of the isolated reactor.

A regulator failure to 115 percent steam flow demand was simulated as a worst case since 110 percent is the normal maximum flow limit. The depressurization results in the formation of voids in the reactor coolant, causing a rapid rise in reactor vessel water level, and also reduces the reactor power level. It should be noted here that a high level turbine trip would have occurred at about 3.55 seconds. Scram occurred with the isolation at about 12.5 seconds. Steam line and vessel pressures drop slightly over 100 psi before the main steam line isolation becomes effective. The safety/relief valves open partially to dissipate the stored heat and then close as they follow the decay heat characteristics. No reduction in fuel thermal margins occurs. The isolation limits the duration of the depressurization so that no significant thermal stresses are imposed on the Reactor Coolant Pressure Boundary.

In SC05-03, GE concluded that since during the PRFO, the Critical Power Ratio (CPR) increases during depressurization, so that the initial CPR is the limiting CPR condition during the entire transient, and that the conditions that exceed the low pressure TS safety limit exist for only a few seconds, fuel cladding integrity is not threatened. Nevertheless, GE considers the PRFO to be a known AOO that could contribute to the exceeding of a safety limit.

TS SLs are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs. Reactor Core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage would occur if the SLs are not exceeded. For JAF TS 2.1.1.1 and TS 2.1.1.2, the GE critical power correlation (also known as the GEXL critical power correlation) is applicable for operation at pressures greater than or equal to 785 psig and core flows greater than or equal to 10% of rated flow. A core thermal power limit of 25% rated thermal power ensures consistency with the threshold for requiring thermal limit monitoring (i.e., average planar linear heat generation rate, linear heat generation rate, and minimum critical power ratio (MCPR)). This assures that for those power levels where thermal limit monitoring is required, the GE critical power correlation is applicable. This SL was introduced to ensure the validity of MCPR calculations when power is > 25% and the reactor pressure is within the validity range of the GEXL correlation. GE has updated the validity range of GEXL Correlations via Reference 3, which allows the pressure to be reduced to 685 psig (700 psia) from 785 psig (800 psia). Therefore a wider pressure range is available for transients to demonstrate compliance with MCPR limits. Thus, the proposed change offers a greater pressure margin for a PRFO transient than what is currently available.

4. REGULATORY ANALYSIS

4.1 APPLICABLE REGULATORY REQUIREMENT/CRITERIA

The proposed change addresses an issue identified in a 10 CFR Part 21 communication regarding the potential for boiling water reactors to experience reactor pressure below the low pressure SL of 785 psig defined in standard Improved Technical Specifications SL 2.1.1 under certain transient conditions.

According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design", the reactor coolant pressure boundary shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and AOOs. The SL on reactor pressure protects the reactor coolant system (RCS) against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor pressure ensures continued RCS integrity. The proposed change will continue to ensure that RCS integrity is maintained because the GEXL correlations have been shown to be valid down to 685 psig for type of fuel in use at JAF.

10 CFR 50.36(c)(1) requires that SLs be included in the TS. SLs for nuclear reactors are 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. The proposed change modifies existing SLs.

4.2 <u>PRECEDENT</u>

The proposed change is consistent with similar change approved for Grand Gulf Nuclear Station in pages 324-325 [Reference 2].

4.3 <u>NO SIGNIFICANT HAZARDS CONSIDERATION</u>

Pursuant to 10 CFR 50.92, Entergy Nuclear Operations, Inc. (ENO) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10 CFR 50.92(c). These criteria require that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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.

The proposed change would revise the JAF TS to reduce the reactor pressure associated with the Reactor Core Safety Limit (SLs) from 785 psig to 685 psig in TS 2.1.1.1 and TS 2.1.1.2. The proposed changes would address the potential to exceed the low pressure TS safety limit associated with a Pressure Regulator Failure-Maximum Demand (Open) (PRFO) transient as reported by General Electric Nuclear Energy (GE) in Reference 1. The proposed change is consistent with similar change approved for Grand Gulf Nuclear Station in pages 324-325 of Reference 2.

The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

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

Response: No.

Decreasing the reactor pressure in TS Safety Limit 2.1.1.1 or 2.1.1.2 for reactor rated thermal power ranges effectively expands the validity range for GEXL correlation and the calculation of Minimum Critical Power Ratio Safety Limit (MCPR). The CPR rises during the pressure reduction following the scram that terminates the PRFO transient. Since the change does not involve a modification of any plant hardware, the probability and consequence of the PRFO transient are essentially unchanged. The reduction in the reactor dome pressure value in the safety limit from 800 psia (785 psig) to 700 psia (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 GEXL correlation and the calculation of MCPR as approved. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or 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 amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed reduction in the reactor pressure value in the safety limit from 800 psia (785 psig) to 700 psia (685 psig) reflects a wider range of applicability for the GEXL correlation for fuels in use at JAF 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 amendment 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 the reactor pressure safety limit 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 on the above, Entergy 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. ENVIRONMENTAL CONSIDERATIONS

This amendment request meets the eligibility criteria for categorical exclusion from environmental review set forth in 10 CFR 51.22(c)(9) as follows:

(i) The amendment involves no significant hazards determination.

As described in Section 4 of this evaluation, the proposed change involves no significant hazards consideration.

Description and Assessment

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed amendment does not involve any physical alterations to the plant configuration that could lead to a change in the type or amount of effluent release offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above, JAF concludes that the proposed change meets the eligibility criteria for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6. References

1. SC05-03, “10CFR21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit,” dated March 29, 2005
2. ML121210020, “Grand Gulf Nuclear Station Unit 1-Issuance of Amendment RE: Extended Power Uprate (TAC No. ME4679),” dated July 18, 2012
3. NEDC-33292P, Rev 3, “GEXL17 Correlation for GNF2 Fuel”, dated June 2009
4. NEDC-33270P, Rev. 4, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)”, dated October 2011

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Attachment 2

Proposed Technical Specification Changes (Markup)

<u>Page</u>

FOL Page 3

TS Page 2.0-1

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use, at any time, any byproduct, source and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus, components or tools..

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 2536 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~304~~, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protections program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated November 20, 1972; the SER Supplement No. 1 dated February 1, 1973; the SER Supplement No. 2 dated October 4, 1974; the SER dated August 1, 1979; the SER Supplement dated October 3, 1980; the SER Supplement dated February 13, 1981; the NRC Letter dated February 24, 1981; Technical Specification Amendments 34 (dated January 31, 1978), 80 (dated May 22, 1984), 134 (dated July 19, 1989), 135 (dated September 5, 1989), 142 (dated October 23, 1989), 164 (dated August 10, 1990), 176 (dated January 16, 1992), 177 (dated February 10, 1992), 186 (dated February 19, 1993), 190 (dated June 29, 1993), 191 (dated July 7, 1993), 206 (dated February 28, 1994) and 214 (dated June 27, 1994); and NRC Exemptions and associated safety evaluations dated April 26, 1983, July 1, 1983, January 11, 1985, April 30, 1986, September 15, 1986 and September 10, 1992 subject to the following provision:

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:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 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.

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Attachment 3

Revised Technical Specification Pages (Clean)

<u>Pages</u>

FOL Page 3

TS Page 2.0-1

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use, at any time, any byproduct, source and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus, components or tools..

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 2536 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

ENO shall implement and maintain in effect all provisions of the approved fire protections program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated November 20, 1972; the SER Supplement No. 1 dated February 1, 1973; the SER Supplement No. 2 dated October 4, 1974; the SER dated August 1, 1979; the SER Supplement dated October 3, 1980; the SER Supplement dated February 13, 1981; the NRC Letter dated February 24, 1981; Technical Specification Amendments 34 (dated January 31, 1978), 80 (dated May 22, 1984), 134 (dated July 19, 1989), 135 (dated September 5, 1989), 142 (dated October 23, 1989), 164 (dated August 10, 1990), 176 (dated January 16, 1992), 177 (dated February 10, 1992), 186 (dated February 19, 1993), 190 (dated June 29, 1993), 191 (dated July 7, 1993), 206 (dated February 28, 1994) and 214 (dated June 27, 1994); and NRC Exemptions and associated safety evaluations dated April 26, 1983, July 1, 1983, January 11, 1985, April 30, 1986, September 15, 1986 and September 10, 1992 subject to the following provision:

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:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 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.

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Attachment 4

Proposed Technical Specification Bases Changes (Information Only)

Pages

B 2.0 SAFETY LIMITS (SLs)
B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

JAFNPP design criteria (Ref. 1) requires. and SLs ensure. that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormal operational transients.

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during abnormal operational transients, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding

(continued)

BASES

BACKGROUND (continued)

to a structurally weaker form. This weaker form may lose its integrity. resulting in an uncontrolled release of fission products 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 ANALYSIS

The fuel cladding must not sustain damage as a result of normal operation and abnormal operational transients. 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 limit.

2.1.1.1 Fuel Cladding Integrity

The GEXL17 critical power is applicable for all critical power calculations at pressure ≥ 685 psig and core flows ≥10% of rated flow (References 5 and 6). ~~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 103 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 103 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

2.1.1.2 MCPR

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 uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

2.1.1.3 Reactor Vessel Water Level

The reactor vessel water level is required to be above the top of the active irradiated fuel. The top of the active irradiated fuel is the top of a 150 inch fuel column which includes both the enriched and the natural uranium. During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated 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 (Ref. 3). 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.

(continued)

BASES

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent 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.

APPRLICABILITY

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

SAFETY LIMIT VIOLATIONS

Exceeding a SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100. "Reactor Site Criteria," limits (Ref. 4). 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 UFSAR, Section 16.6.

2 NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, (Revision specified in the COLR).

3 NEDC-31317P, Revision 2. James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA. Loss-of-Coolant Accident Analysis, April 1993.

4 10 CFR 100.

5 NEDC-33292P, Rev 3, "GEXL17 Correlation for GNF2 Fuel", dated June 2009.

~~4~~6 NEDC-33270P, Rev. 4, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)", dated October 2011.