Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
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Robert G. Smith, P.E.
Site Vice President

April 5, 2013

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

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Proposed License Amendment: Revision to Technical Specification (TS) 2.1, Safety Limits To Resolve Pressure Regulator Fail-Open (PRFO) Transient Reported by General Electric Nuclear Energy In Accordance with 10 CFR 21.21(d)

REFERENCES:

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

2. Grand Gulf Nuclear Station Unit 1, Issuance of Amendment No. 191, RE.: Extended Power Uprate (pages 324-325), dated July 18, 2012 (TAC NO. ME 4679)

LETTER NUMBER: 2.13.009

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) requests an amendment to the Pilgrim Operating License Technical Specifications (TS). The proposed amendment reduces the reactor dome pressure from 785 psig to 685 psig in TS 2.1.1 and 2.1.2 Safety Limits, and associated Bases, and adds NRC approved References to resolve the Pressure Regulator Failure-Open (PRFO) transient reported by GE Nuclear Energy in Reference 1.

Attachment 1 to this letter provides "Evaluation of Proposed TS Changes" and Attachment 2 provides "Marked-up Pages of the Current TS and Bases". The marked-up Technical Specification Bases pages are provided for information only.

The proposed changes have been evaluated in accordance with 10 CFR 50.92(c) and determined that the changes involve no significant hazards considerations. Attachment 1 provides the No Significant Hazards Consideration Determination for the proposed change. Attachment 2 provides marked-up pages of the current TS and Bases. Marked-up Bases pages are provided for information only.

ADD1
NRR

The proposed TS Amendment follows the Grand Gulf License Amendment 191, dated July 18, 2012 (Reference 2).

Entergy requests NRC approval of the proposed Pilgrim TS amendment by April 30, 2014. Once approved, the amendment will be implemented within 60 days.

This letter contains no new regulatory commitments.

If you have any questions regarding the subject matter, please contact Joseph R. Lynch at (508) 830-8403.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 5th day of APRIL, 2013.

Sincerely,

Robert G. Smith

Robert G. Smith, P.E.
Site Vice President

Attachment 1: Evaluation of Proposed TS Changes (6 pages).
Attachment 2: Marked-up Pages of the Current TS and Bases (6 pages).

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Evaluation of Proposed TS Changes

1.0 DESCRIPTION

On March 29, 2005, GE Nuclear Energy (GE) issued a Safety Communication (SC 05-03) in accordance with 10 CFR 21.21(d) to Entergy, invoking a Reportable Condition for Potential Violation of Low Pressure Technical Specification (TS) Safety Limit (SL) (Reference 1). GE identified an unanalyzed condition where a Pressure Regulator Failure Open (PRFO) – Maximum Demand Abnormal Operation Occurrence (AOO) may cause a TS SL to be violated since reactor dome pressure could drop below the current TS SL 2.1 value of 785 psig while reactor power is above 25% of rated thermal power (RTP). GE identified that even plants with an MSIV low pressure isolation setpoint ≥ 785 psig may experience an AOO that potentially could violate the SL.

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

GE advanced fuel designs have an NRC approved critical power correlation with a lower-bound pressure significantly below the 785 psig reactor steam dome pressure specified in TS Reactor Core Safety Limits 2.1.1 and 2.1.2. Entergy proposes to utilize this fact and reduce the reactor steam dome pressure safety limit consistent with the approved lower-bound pressure for the GE fuel comprising the Pilgrim reactor core. GE fuel utilizes the GEXL critical power correlation, with an approved pressure range from 700 to 1400 psia (685 to 1385 psig). Entergy has determined that changing the pressure limit in TS Safety Limit 2.1 to 685 psig as permitted by References 2 and 3, provides adequate margin for the PRFO transient, such that the dome pressure will remain above the revised TS 2.1 safety limit.

Accordingly, pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) hereby requests an amendment to the Pilgrim Operating License Technical Specifications (TS). The proposed amendment revises the reactor dome pressure from 785 psig to 685 psig in TS Safety Limits 2.1.1 and 2.1.2 and TS Bases 2.1.1, 3.1 and 3.2 to resolve the potential to violate these limits during a pressure regulator failure open (PRFO) transient.

Pilgrim proposed TS change follows the NRC approved TS change for Grand Gulf by License Amendment No. 191, dated July 18, 2012 (TAC NO. ME 4679).

2.0 PROPOSED CHANGES

2.1 The proposed changes to the Technical Specifications are as follows:

The reactor dome pressure 785 psig is revised to 685 psig in TS Safety Limits 2.1.1 and 2.1.2.

2.2 The current reactor dome pressure value of 785 psig is revised to 685 psig in the TS Bases Pages B2-1, B2-2, and B3/4.2-2.

The Bases on B3/4.1-5 is revised to describe the role of the MSIV closure in preventing power operation with reactor pressure below 685 psig and core thermal power greater than 25% of rated thermal power.

Additionally, Reference (5) is deleted as it is no longer applicable and the following three References are added on TS Bases Page B2-4:

NEDC-32851P-A, Rev. 4, "GEXL14 Correlation for GE 14 Fuel", dated September 15, 2008

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

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

The BASES pages are provided for NRC information only. TS BASES changes will be incorporated into TS upon receipt of the NRC approved License Amendment in accordance with TS 5.5.6, Bases Control Program. The above referenced documents are contained in fuel design information reports in accordance with GESTAR-II Section 1.2.7. Therefore, these documents are included by reference.

3.0 BACKGROUND

References 2 and 3 documented the expanded pressure range for GEXL correlations for the current co-resident fuels, GE14 and GNF2 in the Pilgrim Reactor Core. Subsequently, GE Part 21 (Reference 1) identified an Anticipated Operational Occurrence (AOO) due to a Pressure Regulator Maximum Demand Open (PRFO) transient that could potentially result in violations of the low pressure Safety Limits in TS 2.1.1 and 2.1.2 as it is currently set at 785 psig.

Entergy reviewed the GEXL14 and GEXL17 correlations approved by the NRC in NEDC-32851P-A, Rev. 4 and NEDC-33292P, Rev. 3 (References 2 and 3), and concluded that the GEXL14 and GEXL17 correlations apply for GE14 and GNF2 fuel respectively. Since the Pilgrim core has both GE14 and GNF2 fuel, Pilgrim is proposing to reduce the current 785 psig reactor dome pressure limit in Safety Limits 2.1.1 and 2.1.2 and associated TS Bases to 685 psig, using GESTAR II, NRC approved documents (References 2, 3 and 5). The proposed reduction in the safety limit dome pressure is consistent with the pressure range specified in the NRC approved critical power correlations for GE14 and GNF2 fuel designs.

Entergy reviewed the GEXL14 and GEXL17 correlations in NEDC-32851P-A, Rev. 4 and NEDC-33292P, Rev. 3 for GE14 and GNF2 fuel respectively and has determined that they are applicable to the GE14 and GNF2 fuel used in the Pilgrim core. The proposed reduction in the current 785 psig reactor pressure limit in TS SL 2.1. to 685 psig is within the range of applicable pressures in the GEXL correlations.

The NRC has approved NEDC-32851P-A for use. 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 5) 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.

This change ensures reactor pressure remains above the proposed low pressure limit of 685 psig with reactor power greater than 25% of rated during a Pressure Regulator Fail Open (PRFO) transient. Thus the proposed change based on the updated GEXL pressure range documented by GEH, resolves the concern reported by GEH in Safety Communication SC05-03.

The proposed TS change follows the TS changes previously approved for Grand Gulf by License Amendment 191, dated July 18, 2012 (TAC NO. ME 4679) (Reference 4).

4.0 TECHNICAL ANALYSIS

The potential pressure regulator failure open event involves the failure of the pressure regulator in the open direction causing the turbine control valves to fully open, including the bypass valves. This causes the reactor to depressurize rapidly. When, the Main Steam Line (MSL) low pressure setpoint is reached, the Main Steam Isolation Valves (MSIVs) start to close and a reactor scram occurs. As the MSIVs approach full closure, reactor depressurization terminates and pressure commences to rise to the safety-relief valve setpoint, thus preventing reactor pressure from decreasing below the proposed safety limit of 685 psig while core thermal power is still above 25% of rated thermal power. The fuel cladding integrity is never challenged because in pressure decrease events like this in a BWR, fuel critical power is rising and therefore MCPR rises during the event.

Technical Specification (TS) Safety Limits are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). Reactor core Safety Limits are set such that fuel cladding integrity is maintained and no significant fuel damage would occur if the Safety Limits are not exceeded. Consistent with the Improved Technical Specifications (ITS), Pilgrim specifies Safety Limits in TS 2.1.1 and 2.1.2 to require that thermal power shall be ≤ 25% Rated Thermal Power (RTP) when reactor steam dome pressure is < 785 psig or core flow is < 10% of RTP. This Safety Limit 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. GEH has updated the validity range of GEXL Correlations via References 2 and 3, which allows the pressure to be reduced to 685 psig from 785 psig. 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.

The Entergy proposed changes to the TS safety limits are based upon the NRC approved topical reports. The proposed change has already been authorized for Grand Gulf Nuclear Station Unit 1 (Reference 4). The Pilgrim proposed TS change follows NRC approved methodology and is acceptable.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy requests a License Amendment to Pilgrim Operating License Technical Specifications (TS) to make the following changes:

- To reduce the reactor dome pressure from 785 psig to 685 psig in TS 2.1.1 and 2.1.2.

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

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

   Response: No

   Decreasing the reactor dome pressure in Technical Specification Safety Limits 2.1.1 and 2.1.2 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). MCPR 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 785 psig to 685 psig provides adequate margin to accommodate the pressure reduction during the transient within the revised TS limit.

   The expanded GEXL correlation range supports Pilgrim's revised low pressure safety limit of 685 psig. 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 change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed reduction in the reactor dome pressure value in the safety limit from 785 psig to 685 psig reflects a wider range of applicability for the GEXL correlation which is approved by the NRC for fuels in use at Pilgrim and does not involve changes to the plant hardware or its operating characteristics. As a result, no new failure modes are being introduced.

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

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

   Response: No

   The margin of safety is established through the design of the plant structures, systems, and components, and through the parameters for safe operation and setpoints for the actuation of equipment relied upon to respond to transients and design basis accidents. The proposed change in reactor dome pressure restores 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 reduction in the reactor dome pressure value in the safety limit from 785 psig to 685 psig provides adequate margin to accommodate the pressure reduction during the transient within the revised TS limit.

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

Based upon the above, Pilgrim 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.

5.2 Applicable Regulatory Requirements and Criteria

10 CFR 50, Appendix A provides criteria for Emergency Core Cooling System (ECCS) performance and 10 CFR 50.36, Technical Specifications, requires safety system settings to ensure the integrity of the reactor pressure boundary during normal and abnormal operations and to mitigate transient and accident conditions. The proposed decrease in the reactor dome pressure limit in TS 2.1.1 and 2.1.2 follows the requirements cited above and ensures the fuel cladding integrity.

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

6.0 ENVIRONMENTAL CONSIDERATION

Entergy has determined that the proposed amendment 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, "Standards for Protection Against Radiation." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for Categorical Exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review", Paragraph (c)(9). Therefore, in accordance with 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 COORDINATION WITH PENDING TS CHANGES

There are no pending proposed TS changes that are being filed for license amendment that would impact the proposed TS changes.

8.0 REFERENCES

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

2. NEDC-32851P-A, Rev. 4, "GEXL14 Correlation for GE14 Fuel", dated September 15, 2008.

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

4. Grand Gulf Nuclear Station Unit 1, Issuance of Amendment No. 191, RE.: Extended Power Uprate (pages 324-325), dated July 18, 2012 (TAC NO. ME 4679).

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

Attachment 2 to Entergy Letter 2.13.009

Marked-up Pages of the Current TS and Bases
(6 Pages)

1. TS Page 2-1
2. TS Bases Page B2-1
3. TS Bases Page B2-2
4. TS Bases Page B2-4
5. TS Bases Page B3/4.1-5
6. TS Bases Page B3/4.2-2

2.0 SAFETY LIMITS

2.1 Safety Limits

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

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

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

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1340 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met within two hours the following actions shall be met:

2.2.1 Restore compliance with all Safety Limits, and

2.2.2 Insert all insertable control rods.

BASES:

2.0 SAFETY LIMITS

INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no 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 a Safety Limit such that the Minimum Critical Power Ratio (MCPR) is not less than the limit specified in Specification 2.1.2. 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 which 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 Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling (i.e., MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, 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 to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

FUEL CLADDING INTEGRITY (2.1.1)

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

(Cont)

BASES:

2.0 SAFETY LIMITS (Cont)

FUEL CLADDING INTEGRITY (2.1.1) (Cont)	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 greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/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 greater than 28 x 10^3 lbs/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 of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below ~~785~~ 685 psig is conservative.
MINIMUM CRITICAL POWER RATIO (2.1.2)	The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (2), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. Instead of the standard GETAB model uncertainties, revised uncertainties in accordance with references 3 and 4 were used to calculate the SLMCPR. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at

(Cont)

BASES:

2.0 SAFETY LIMITS (Cont)

REACTOR STEAM DOME PRESSURE (2.1.4)

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the reactor coolant system is not endangered. The reactor pressure limit of 1340 psig as measured in the vessel steam dome was derived from the design pressure of the reactor vessel. The peak pressures for the piping systems connected to the reactor vessel have been recalculated based on a reactor steam dome peak pressure of 1340 psig. These peak pressures are below the lowest of the transient pressures permitted by the applicable design code: ASME Boiler and Pressure Vessel (B&PV) Code (1965 Edition, including the January 1966 Addendum) for the pressure vessel, USAS Piping Code B31.1 for the steam space piping and ASME Section III for the reactor coolant system recirculation piping. The ASME B&PV Code permits pressure transients up to 10% over the design pressure (110% x1250=1375 psig). The USAS Piping Code and ASME Section III permit pressure transients and other occasional loads whose combined effect do not exceed stress levels based on the duration of the loads and the applicable service limit.

REFERENCES

1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (through the latest approved amendment at the time the reload analyses are performed as specified in the CORE OPERATING LIMITS REPORT).

2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

3) "Methodology & Uncertainties for SLMCPR Evaluations," NEDC-32601-P-A (August 1999).

4) "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694-P-A (August 1999).

5) "GE 11 Compliance with Amendment 22 of GESTAR II," NEDE-31917P (April 1991).

5 ~~6~~) "GE 14 Compliance with Amendment 22 of GESTAR II," NEDC-32868P (December 1998).

6 ~~7~~) "Pilgrim Nuclear Power Station Safety Valve Setpoint Increase," GE Hitachi Nuclear Energy Report, NEDC-33532P, Rev. 2 (January 2011)

7) NEDC-32851P-A, Rev. 4, GEXL14 Correlation for GE 14 Fuel, dated September 15, 2008

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

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

BASES:

3.1 REACTOR PROTECTION SYSTEM (Cont)

Turbine Stop Valve Closure

The turbine stop valve closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the safety limit MCPR even during the worst case transient that assumes the turbine bypass is closed.

Turbine Control Valve Fast Closure

The turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the bypass valves. The reactor protection system initiates a scram when fast closure of the control valves is initiated by the acceleration relay. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop valve closure. MCPR remains above the safety limit MCPR.

Main Steam Line Isolation Valve Closure

The low pressure isolation of the main steam lines at 810 psig (as specified in Table 3.2.A) was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. ~~Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit.~~ Operation of the reactor at pressures lower than 785 psig requires the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram and APRM 15% scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

Advantage is taken of both the reactor scram that occurs when the main steam isolation valves close and the limited depressurization allowed by the rapid steam line isolation valve closure time, so that high power operation at low reactor pressure does not occur, thus providing protection of the fuel cladding protection safety limit.

BASES:

3.2 PROTECTIVE INSTRUMENTATION (Cont)

up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation, and primary system isolation are initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of Group 1 isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, steam flow trip setting in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain approximately 1000°F and release of radioactivity to the environs is well below 10CFR100 guidelines.

Temperature monitoring instrumentation is provided in the main steam line tunnel and the turbine basement to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 175° F for the main steam line tunnel detector is low enough to detect leaks on the order of ≥ 20 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

Pressure instrumentation is provided to close the main steam isolation valves in the RUN mode before the reactor pressure drops below 785 psig. This function is primarily intended to prevent excessive vessel depressurization in the event of a malfunction of the nuclear system pressure regulator. This function also provides automatic protection of the low-pressure core-thermal-power safety limit (~~25% of rated core thermal power for reactor pressure < 785 psig~~). In the Refuel or Startup Mode, the inventory loss associated with such a malfunction would be limited by closure of the Main Steam Isolation Valves due to either high or low reactor water level; no fuel would be uncovered. This function is not required to satisfy any safety design bases.

(< 25% of rated core thermal power with reactor pressure < 685 psig)