Attachment I to JPN-96-025

REVISED TECHNICAL SPECIFICATION PAGES

PROPOSED TECHNICAL SPECIFICATION CHANGES
REGARDING MINIMUM CRITICAL POWER RATIO SAFETY LIMIT

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications:

A. Reactor Pressure >785 psig and Core Flow >10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.09 shall constitute violation of the fuel cladding integrity safety limit, hereafter called the Safety Limit. An MCPR Safety Limit of 1.10 shall apply during single-loop operation.

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. IRM - The IRM flux scram setting shall be set at ≤120/125 of full scale.

1.1 BASES

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit minimum critical power ratio (MCPR). This Safety 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 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 and the protection system safety settings. 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 margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure >785 psig and Core Flow>10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variable, i.e., the operating domain. The current load line limit analysis contains the current operating domain map. The Safety Limit MCPR has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the MCPR operating limit in the Core Operating Limits Report, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The MCPR fuel cladding safety limit is increased by 0.01 for single-loop operation as discussed in Reference 2. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including the uncertainty in the boiling transition correlation. The method of determining the Safety Limit is described in Reference 1. The boiling transition correlation and the uncertainties employed in deriving the Safety Limit are

1.1 (cont'd)

provided in Reference 3. Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of fuel assembly at the Safety Limit would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to FitzPatrick operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (Safety Limit), operation is constrained by the maximum LHGR identified in the Core Operating Limits Report.

At 100% power, this limit is reached with a maximum fraction of limiting power density (MFLPD) equal to 1.00. In the event of operation with MFLPD greater than the fraction of rated power (FRP), the APRM scram and rod block settings shall be adjusted as specified in Tables 3.1-1 and 3.2-3 respectively.

B. Core Thermal Power Limit (Reactor Pressure $<$785 psig)

At pressures below 785 psig the core elevation pressure drop is greater than 4.56 psi for no boiling in the bypass region. At low powers and flows, this pressure drop is due to the elevation pressure of the bypass region of the core. Analysis shows that for bundle power in the range of 1-5 MWt, the channel flow will never go below 28×10^3 lb/hr. This flow results from the pressure differential between the bypass region and the fuel channel. The pressure differential is primarily a result of changes in the elevation pressure drop due to the density difference between the boiling water in the fuel channel and the non-boiling water in the bypass region. Full scale ATLAS test data taken at pressures from 0 to 785 psig indicate that the fuel assembly critical power at 28×10^3 lb/hr is approximately 3.35 MWt. With the design peaking factors, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

1.1 BASES (Cont'd)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the Safety Limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

E. References

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P, latest approved revision and amendments.

2. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO 24281, August 1980.

3. GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-32417P, December 1994.

Attachment II to JPN-96-025

SAFETY EVALUATIC · FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING MINIMUM CRITICAL POWER RATIO SAFETY LIMIT

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333

SAFETY EVALUATION
PROPOSED TECHNICAL SPECIFICATION CHANGES
REGARDING MINIMUM CRITICAL POWER RATIO SAFETY LIMIT

I. DESCRIPTION OF THE PROPOSED CHANGES

The following proposed changes to the James A. FitzPatrick Technical Specifications establish a revised Minimum Critical Power Ratio (MCPR) safety limit and associated basis. The changes are required to support introduction of GE12, 10x10 fuel into the Cycle 13 core.

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Change "1.07" in specification 1.1.A to "1.09." Change "1.08" in specification 1.1.A to "1.10."

Pages 12 and 13

Change "The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including the uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided in Reference 1." to "The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including the uncertainty in the boiling transition correlation. The method of determining the Safety Limit is described in Reference 1. The boiling transition correlation and the uncertainties employed in deriving the Safety Limit are provided in Reference 3."

Page 14

Add new reference, 1.1.Bases.E.3:

"3. GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-32417P, December 1994."

II. PURPOSE OF THE PROPOSED CHANGES

The purpose of the proposed changes is to provide the appropriate MCPR safety limit for the Reload 12 / Cycle 13 core. Reload 12 will consist of GE12 fuel, which utilizes a 10x10 mechanical design.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGES

The proposed changes revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) to be 1.09 for two-loop operation and 1.10 for single-loop operation, and make associated changes to the bases. These changes are required to support loading GE12 fuel bundles in the Cycle 13 core.

GE12 fuel was demonstrated to meet the fuel licensing acceptance criteria of amendment 22 of NEDE-24011-P-A (GESTAR II), General Electric Standard Application for Reactor Fuel as described in the reference 1 report. Changes to GESTAR II to address GE11, GE12 and GE13 fuel designs have been incorporated in NEDE-24011-P-A-11 (reference 2). The fuel licensing acceptance criteria from reference 2 discussed below are the same as those described in reference 1.

Subsection 1.1.5.A of reference 2 requires "Safety Limit MCPR shall be recalculated following steps in 1.1.5.B or reconfirmed when a new fuel design or new critical power correlation is introduced." Subsection 1.1.7.A of reference 2 requires "The currently approved critical power correlation will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry; this specifically includes fuel and water rod diameter, channel sizing and spacer design." Subsection 1.1.7B of reference 2 allows "A new correlation may be established if significant new data exists for a fuel design(s)."

Reference 1 discusses the derivation, applicability and uncertainty of the GEXL10 critical power correlation as applied to GE12 fuel. This correlation was used as a basis for determination of the SLMCPR for GE12 fuel. The SLMCPR is also influenced by bundle design parameters which affect the bundle R-Factor distribution and core radial power distribution. These parameters include spacer design, assembly dimensional geometry, bundle radial power distribution, and fuel discharge exposure. The SLMCPR calculated for GE12 fuel is 1.09, which is being conservatively applied to the entire core.

Reference 3 describes operation of the FitzPatrick Plant with a single Reactor Water Recirculation loop in service (single-loop operation, SLO). This mode of operation requires raising the SLMCPR by 0.01 to account for changes in core flow and Traversing Incore Probe (TIP) uncertainties. Therefore, the SLMCPR for SLO applicable to GE12 fuel is 1.10.

The changes to the bases are administrative in nature. They reflect relocation of details of the critical power correlation from the General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (GESTAR II) to documents which report fuel bundle design compliance with the fuel licensing acceptance criteria of GESTAR II (reference 2). Since this change does not affect the criteria to be satisfied by a new fuel design, there is no effect on nuclear safety.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated because:

   A change in the SLMCPR does not affect initiation of any accident. Operation in accordance with the revised SLMCPR ensures the consequences of previously analyzed accidents are not changed.

2. create the possibility of a new or different kind of accident from any accident previously evaluated because:

   The SLMCPR establishes a performance limit for the fuel. Therefore changing the limit will not initiate any accident.

3. involve a significant margin of safety because:

   The analyses performed to determine the revised SLMCPR assure maintenance of the same margin of safety as presently exists for the prevention of onset of transition boiling.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not adversely affect the ALARA or Fire Protection Program at the FitzPatrick plant, nor will the changes impact the environment.

VI. CONCLUSION

Based on the discussions above, implementation of a SLMCPR of 1.09 (1.10 for SLO) does not involve a significant hazards consideration, or an unreviewed safety question, and will not endanger the health and safety of the public. The Plant Operating Review Committee and Safety Review Committee have reviewed this proposed Technical Specification change and agree with this conclusion.

VII. REFERENCES

(1) GE12 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-32417P, December 1994.

(2) General Electric Standard Application for Reactor Fuel, (GESTAR II), NEDE-24011-P-A, November 1995.

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MARKUP OF TECHNICAL SPECIFICATION PAGE CHANGES

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Specifications:

A. Reactor Pressure >785 psig and Core Flow >10% of Rated

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1.09

1.10

2.1 FUEL CLADDING INTEGRITY

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The method of determining the Safety Limit is described in Reference 1.

1.1 (cont'd)

provided in Reference 1. (3) Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of fuel assembly at the Safety Limit would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

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B. Core Thermal Power Limit (Reactor Pressure <785 psig)

At pressures below 785 psig the core elevation pressure drop is greater than 4.56 psi for no boiling in the bypass region. At low powers and flows, this pressure drop is due to the elevation pressure of the bypass region of the core. Analysis shows that for bundle power in the range of 1-5 MWt, the channel flow will never go below 28×10^3 lb/hr. This flow results from the pressure differential between the bypass region and the fuel channel. The pressure differential is primarily a result of changes in the elevation pressure drop due to the density difference between the boiling water in the fuel channel and the non-boiling water in the bypass region. Full scale ATLAS test data taken at pressures from 0 to 785 psig indicate that the fuel assembly critical power at 28×10^3 lb/hr is approximately 3.35 MWt. With the design peaking factors, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

1.1 BASES (Cont'd)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

D. Reactor Water Level (Hot or Cold Shutdown Condition)

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the Safety Limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

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