SUMMARY OF CHANGES TO ITS CHAPTER 2.0

SUMMARY OF CHANGES TO ITS SECTION 2.0 - REVISION D

Source of Change	Summary of Change	Affected Pages
RAI 2.0-1	The original version of the ITS submittal retained a cycle specific CTS Safety Limit note that the SLMCPR is applicable to cycle 14 only. The Staff indicated that this note is not in the ISTS, and further indicated that the SLMCPR note was not necessary and that its treatment was being addressed by TSTF-357. In the RAI response, JAF responded that TSTF-357 was not yet approved by the NRC and that it was also unrelated to the note. However, consistent with the Staff's comment, JAF committed to remove the cycle specific SLMCPR note by a CTS License Amendment and to revise the ITS submittal accordingly after an approved CTS License Amendment was received. This change was received as License Amendment 266. Revision D revises ITS Section 2.0 to reflect the resulting ITS submittal changes, as per the RAI 2.0-1 response.	CTS mark-up, p 2 of 6 ITS mark-up, p 2.0-1 JFD CLB1 (JFDs p 1 of 1) ITS Bases mark-up, p B 2.0-6 Bases JFD CLB1 (JFDs p 1 of 2) Retyped ITS pp 2.0-1, B 2.0-4
License Amendment Number 266	As discussed under RAI 2.0-1 above, this amendment removed the cycle specific note which stated that "TS 1.1.A is applicable for Cycle 14 only." The ITS is revised to reflect the amendment. Since Amendment 266 and associated changes are the result of RAI 2.0-1, marginal change annotation reflects the RAI number instead of the Amendment.	Same as for RAI 2.0-1
Editorial Correction	Changed the ITS Bases markup JFD reference for Bases 2.1.1 Reference 1 from CLB1 to the more appropriate DB4.	ITS Bases mark-up, p B 2.0-7
Editorial Correction	Changed the revision number for Reference 3, NEDC-31317P, SAFER/GESTR-LOCA, from Rev. 3 to Rev.2, the currently approved and accepted revision for FitzPatrick.	ITS Bases mark-up, p B 2.0-7 Retyped ITS p B 2.0-5

ITS CONVERSION PACKAGE

SECTION 2.0 - SAFETY LIMITS (SLS)

JAFNPP
IMPROVED TECHNICAL SPECIFICATION (ITS)

CONVERSION PACKAGE

Section 2.0 - SAFETY LIMITS (SLS)

Table of Contents

The markup package for each Specification contains the following:

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

A1

L4

2.1.1.1

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

When the reactor pressure is ≤ 785 psig or core flow is less than ~~or equal to~~ 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power. <

C. **~~Power Transient~~**

~~To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.~~

L1

b. **APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)**

APRM - The APRM flux scram setting shall be ≤ 15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

See ITS: 3.3.1.1

c. **APRM Flux Scram Trip Settings (Run Mode)**

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be less than or equal to the limit specified in Table 3.1-1. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

A1

2.0 Safety Limits (SLs)
2.1 SLs
2.1.1 Reactor Core Safety Limits

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:

≥ L4 ≥

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

[2.1.1.2]

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.

Chapter 2.0 A1

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.

See ITS: 3.3.1.1

RAI 2.0-1

AMD #266

Chapter 2.0

~~1.1 (cont'd)~~

~~D. Reactor Water Level (Hot or Cold Shutdown Conditions)~~ M1

[2.1.1.3] ~~Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel~~, the water level shall ~~no~~ be ~~less than that corresponding to 18 inches above the Top of Active Fuel when it is seated in the core.~~

greater than the top of active irradiated fuel L2

LA2

2.1 (cont'd)

A1

~~(2) Fixed High Neutron Flux Scram Trip Setting~~

~~When the Mode Switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:~~

~~S ≤120% Power~~

See ITS: 3.3.1.1

~~d. APRM Rod Block Setting~~

~~The APRM Rod block trip setting shall be less than or equal to the limit specified in Table 3.2-3. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.~~

See ITS: 3.3.2.1

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Chapter 2.0 A1

2.0 Safety Limits (SL)
2.1 SLs
A1

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

[2.1.2] 1. The reactor vessel dome pressure shall not exceed 1,325 psig ~~at any time when irradiated fuel is present in the reactor vessel~~ A3

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

See ITS: 3.3.1.1 3.4.3

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

1. The Limiting Safety System setting shall be specified below:

   A. Reactor coolant high pressure scram shall be ≤ 1,080 psig.

   See ITS: 3.3.1.1

   B. At least 9 of the 11 reactor coolant system safety/relief valves shall have a nominal setting of 1,145 psig with an allowable setpoint error of ± 3 percent.

   See ITS: 3.4.3

Chapter 2.0 A1

JAFNPP see CTS Chapter 6.0

6.6 REPORTABLE EVENT ACTION

The following actions shall be taken for Reportable Events:

(A) The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

(B) Each Reportable Event shall be reviewed by the PORC, and the results of this review shall be submitted to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC.

[2.2] 6.7 SAFETY LIMIT VIOLATION within 2 hours M2

A2

[2.2] [2.1.1] [2.2.2] (A) If a safety limit is exceeded, the reactor shall be shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36 (c)(1)(i). L3

(B) An immediate report of each safety limit violation shall be made to the NRC by the Site Executive Officer. The Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC will be notified within 24 hours.

(C) The PORC shall prepare a complete investigative report of each safety limit violation and include appropriate analysis and evaluation of: (I) applicable circumstances preceding the occurrence, (2) effects of the occurrence upon facility component systems or structures and (3) corrective action required to prevent recurrence. The Site Executive Officer shall forward this report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, the Chairman of the SRC, and the NRC.

LA1

6.8 PROCEDURES

(A) Written procedures and administrative policies shall be established, implemented, and maintained that:

1. meet or exceed the requirements and recommendations of Section 5 of ANSI 18.7-1972 "Facility Administrative Policies and Procedures."
2. are recommended in Appendix A of Regulatory Guide 1.33, November 1972.
3. implement the Fire Protection Program.
4. include programs specified in Appendix B of the Radiological Effluent Technical Specifications, Section 7.2.

See ITS: 5.4

(B) Each procedure of Specification 6.8.(A), and changes thereto, shall be approved prior to implementation by the appropriate responsible member of management as specified in Specification 6.5.0.

[2.2.1] Within 2 hours, restore compliance with all SLs M2

Amendment No. ~~50, 60, 65, 78, 93, 110, 220, 222, 228~~, 240

A1

1.2 (cont'd)

2. The reactor vessel dome pressure shall not exceed 75 psig at any time when operating the Residual Heat Removal pump in the shutdown cooling mode.

2.2 (cont'd)

2. Action shall be taken to decrease the reactor vessel dome pressure below 75 psig or the shutdown cooling isolation valves shall be closed.

L5

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS CHAPTER: 2.0 - SAFETY LIMITS (SLs)

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 6.7.(A), in the event of a SL violation, specifies reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36(c)(1)(i). ITS does not retain this specification. 10 CFR 50.36(c)(1)(i) contains the requirements for reactor operation following a Safety Limit violation. This change deletes requirements from the Technical Specifications that duplicate other regulations. Consequentially, this is an administrative change. This change is consistent with NUREG-1433 as modified by TSTF-5.

A3 CTS 1.2.1 requires reactor vessel steam dome pressure to be within limits "at any time when irradiated fuel is present in the reactor vessel." The applicability for the Safety Limits in the proposed Specifications is "in all MODES". Since all MODES are defined as having irradiated fuel in the reactor vessel, the proposed change does not involve a technical change and is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 1.1.D, the Safety Limit for reactor water level, is currently only required when the reactor is in the shutdown condition (in startup and run the reactor would scram before the reactor vessel water level drops to the Safety Limit level). The ITS Safety Limit for reactor water level (ITS 2.1.1.3) proposes to make this requirement applicable in all Modes. This change is consistent with NUREG-1433, Revision 1. This represents a more restrictive change because the Applicability of this Safety Limit has been expanded to address all Modes. This change helps to ensure that sufficient reactor water level is available to provide adequate margin for effective action in the event of a level drop.

M2 A new requirement has been added to CTS 6.7 (Safety Limit Violation). This new requirement will stipulate that "compliance with all SLs" be restored within 2 hours (ITS 2.2.1). In addition, a 2 hour time limit is being added to CTS 6.7.A to "insert all insertable control rods" if a safety limit is exceeded (ITS 2.2.2). The present

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

requirement (CTS 6.7.A) requires the reactor to be shutdown, but does not stipulate a time. By adding the 2 hour time requirement, an additional requirement is being proposed. Exceeding a Safety Limit could cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100 limits. These new more restrictive requirements help ensure that the operators take prompt remedial actions and that the probability of an accident occurring when a Safety Limit is violated is minimal.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 6.7.B requires the notification within 24 hours of a safety limit violation to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, and the Chairman of the SRC. In addition, CTS 6.7.C requires the PORC to prepare a complete investigative report of each safety limit violation and include appropriate analysis and evaluation. The Site Executive Officer shall forward this report to the Chief Nuclear Officer, the Director Regulatory Affairs and Special Projects, the Chairman of the SRC, and the NRC. These requirements are proposed to be relocated to the Technical Requirements Manual (TRM). Given that the notification occurs following the SL violation and that the SL Violation Report is an after-the-fact report, the proposed relocated requirements are clearly not necessary to assure operation of the plant in a safe manner. Additionally, in the event of a SL violation, 10 CFR 50.36(c)(1)(i) does not allow operation of the plant to be resumed until authorization is received from the NRC. As such it is not required to be in the ITS to provide adequate protection of the public health and safety. At ITS implementation, the relocated requirements will be incorporated by reference into the UFSAR. As such changes to the relocated requirements in the TRM will be controlled by provisions of 10 CFR 50.59.

LA2 The detail in CTS 1.1.D that the reactor water level Safety Limit (SL) applies when irradiated fuel is in the reactor vessel is proposed to be relocated to the Bases. The requirement in ITS 2.1.1.3 that the reactor vessel water level shall be greater than the top of active irradiated fuel is adequate to define the Applicability of the limit. With no irradiated fuel in the reactor vessel the safety limit will be met at any reactor vessel water level since there is no fuel to cover. As such, these relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA2 (continued)

Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 1.1.C contains a "Power Transient" Safety Limit. This change proposes to delete the "Power Transient" Safety Limit requirements. The intent of this requirement is to ensure that Safety Limits are not exceeded. This requirement states Safety Limits are assumed to be exceeded when a scram is accomplished by means other than the expected scram signal. The scram setpoints are established in order to ensure margin to the Safety Limits. Exceeding the scram setpoint, in and of itself, does not necessarily indicate that a Safety Limit has been exceeded. CTS 2.1.A and 2.2.1.A contain seven trip settings that initiate a reactor scram. These scram settings are included in ITS Table 3.3.1.1-1. The Surveillance Requirements imposed on these scram setpoints in Table 3.3.1.1-1 help to ensure that the margin to a safety limit is preserved. The redundancy built into the RPS is maintained by the ACTIONS of ITS 3.3.1.1. Therefore, the intent of current Power Transient Safety Limit requirements in CTS 1.1.C is adequately maintained by the provisions in ITS 3.3.1.1 for the RPS.

L2 The current Safety Limit (SL; CTS 1.1.D) for the reactor vessel water level requires that the level be not less than 18 inches above the Top of Active Fuel when it is seated in the core during operations in hot and cold shutdown. The CTS definition of Top of Active Fuel (CTS 1.0.Z) is the top of the enriched fuel column of each fuel bundle, which is located at a maximum level of 352.5 inches above vessel zero (which is the lowest point in the inside bottom of the reactor vessel). ITS 2.1.1.3 requires that the reactor vessel water level be greater than the top of the active irradiated fuel in all MODES. This is a more restrictive change for MODE applicability (M1). However, it is less restrictive because the proposed reactor vessel water level SL is 12 inches less than the CTS limit. The CTS definition of "Top of Active Fuel" has a different meaning in the ITS with respect to the reactor vessel water level SL. A fuel rod may contain 6 inches of fuel with a natural enrichment of U_{235} above the 352.5 inch level. In the ITS, active fuel (enriched and natural) is considered to exist up to 358.5 inches above vessel zero. Since the active fuel becomes irradiated whenever the reactor has become critical all the active fuel becomes active irradiated fuel. The CTS limit of 18 inches above the Top of Active Fuel was established to ensure cooling to both the enriched and

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

natural portions of the fuel rod. The proposed limit continues to ensure this cooling. The CTS and ITS Bases state that (and plant design and operating license bases conservatively confirm) below 2/3-core height is where elevated cladding temperature and clad perforation would occur from decay heat without adequate cooling capability. With the reactor vessel water level above the top of active irradiated fuel, the fuel will be adequately cooled and there is no reason to require all rods to be inserted, and to remain shutdown to analyze whether continued operations should be permitted.

The current and proposed Technical Specifications impose requirements to help ensure the reactor fuel is adequately cooled at all times. In addition, Plant Emergency Operating Procedures require entry when level is reduced below reactor vessel water Level 3 which is significantly higher than the top of active irradiated fuel. The plant emergency core cooling systems (ECCS) are required to initiate automatically prior to reaching the proposed reactor vessel water level SL. The proposed ITS automatic actuation level (Allowable Value) for the low pressure ECCS is 12 inches above the top of active irradiated fuel which is 62 inches above 2/3-core height in all MODES. The High Pressure Injection System will initiate prior to this Allowable Value. Therefore, if and when a loss of vessel water level occurs, there is an overhead water level of 12 inches above the top of active irradiated fuel (Level 1) and yet 62 inches more before getting to the 2/3-core height level. This Allowable Value and the requirement that ECCS must be Operable will ensure, in all MODES, the accident analysis can be met by maintaining the reactor water level at a minimum level of 2/3-core height, thereby precluding any core damage. Whenever the reactor vessel water level is at or below the Level 3 scram limit, entry into the Emergency Operating Procedures will be required which will require the level to be recovered to above this level which is 15 feet above the top of irradiated fuel. Therefore, plant emergency operating procedures will require the recovery process to begin prior to the Technical Specification SL. This recovery can be accomplished by using all available water injection methods and sources.

Therefore, this 12 inch reduction in the reactor vessel water level SL is considered acceptable since the fuel is adequately cooled when the reactor vessel water level is above 2/3 core height and the current and proposed Specifications as well as plant emergency operating procedures will help ensure the limit is not violated. If the reactor water level remains above the top of active irradiated fuel, there is no reason to require all rods to be inserted, and to remain shutdown to analyze whether continued operations should be permitted since analysis indicate

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

that fuel damage would not have occurred.

L3 The requirement in CTS 6.7.B to immediately report each safety limit violation to the NRC by the Site Executive Officer has been extended to 1 hour in accordance with 10 CFR 50.72. 10 CFR 50.72 requires notification of the NRC Operations Center within 1 hour for events requiring the initiation of a nuclear plant shutdown required by the Technical Specifications. In addition, the explicit requirement to report a safety limit violation is not retained in the ITS since the requirement is duplicative of other regulations.

L4 CTS 1.1.A and 1.1.B have restrictions on the minimum critical power ratio (MCPR) and percent rated thermal power respectively, based on the existing reactor pressure or core flow. In the CTS, MCPR is the limit if reactor pressure is greater than 785 psig and if core flow is greater than 10% of rated. In addition, 25% rated thermal power is the limit if reactor pressure is less than <u>or equal to</u> 785 psig or if core flow is less than <u>or equal to</u> 10% of rated. In the ITS (2.1.1.1 and 2.1.1.2), MCPR is the limit if reactor pressure is greater than <u>or equal to</u> 785 psig and core flow is greater than <u>or equal to</u> 10% of rated, and 25% rated thermal power is the limit if reactor pressure is less than 785 psig or core flow is less than 10% of rated. This change was made since the GE critical power correlations are accurate at pressures greater than or equal to 785 psig and at core flows greater than or equal to 10% of rated. Justification for operation without thermal margin monitoring below 25% RTP is that the individual assembly power is conservatively estimated to be ≤ 3.35 MWt where design basis peaking factors are considered and since the individual assembly flow with natural circulation induced by normal water level and power in the range of 20 to 25% RTP is sufficient to maintain adequate thermal margin for the fuel. The proposed change is slightly less restrictive since the Applicability of CTS 1.1.B (the low pressure or low core flow SL limit) is reduced, however to compensate the Applicability of CTS 1.1.A (high pressure and high core flow SL) has been increased. This change is acceptable since it is consistent with GE critical power correlations.

L5 The CTS 1.2.2 SL, when operating the RHR System in the Shutdown Cooling Mode, is proposed to be incorporated into ITS 3.3.6.1 (Table 3.3.6.1-1 for Primary Containment Isolation Instrumentation). The RHR Shutdown Cooling System is designed with an interlock in the logic for the system isolation valves, which are normally closed during power operation, to prevent opening of the valves above a preset pressure setpoint (Allowable Value) of 75 psig. This setpoint is selected to assure that

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 (continued

pressure integrity of the RHR System is maintained. The high pressure interlock is only provided for equipment protection to prevent an intersystem LOCA and, as such, this function should not be considered a SL on plant operation.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 2.0 - SAFETY LIMITS (SLs)

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change deletes the Power Transient Safety Limit requirement. The intent of this requirement is to ensure that Safety Limits are not exceeded. This requirement states Safety Limits are assumed to be exceeded when a scram is accomplished by means other than the expected scram signal. The scram setpoints are established in order to ensure margin to the Safety Limits. Exceeding the scram setpoint, by itself, does not necessarily indicate that a Safety Limit has been exceeded. CTS 2.1.A and 2.2.1.A contain trip settings that initiate a reactor scram. These scram settings are included in ITS Table 3.3.1.1-1. The surveillance requirements imposed on these scram setpoints in Table 3.3.1.1-1 help to ensure that the margin to a Safety Limit is preserved. The redundancy built into the RPS is maintained by the Actions of ITS 3.3.1.1. The intent of current Power Transient Safety Limit requirement in CTS 1.1.C is maintained by the provisions in ITS 3.3.1.1 for the RPS. Therefore, since the intent of the Power Transient Safety Limit requirement has been retained in the ITS, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

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

Deletion of the Power Transient Safety Limit has no impact on any safety analysis assumptions. The proposed change does not remove or change the margin to the remaining Safety Limits. The scram signals generated by the current TS required trip equipment are retained in proposed RPS Table 3.3.1.1-1. ITS 3.3.1.1 for RPS will also set minimum operability requirements for these scram signals to ensure that design redundancy, including single failure criteria, is maintained. The Operability of the equipment that generates these signals is also ensured by the required surveillances, including calibrations and functional tests. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change relaxes the reactor vessel water level safety Limit requirements from the current Technical Specifications. These requirements do not result in any operation that will increase the probability of initiating an analyzed event. This change will not alter assumptions relative to mitigation of an accident or transient event. The analysis assumes that water level above the top of the active irradiated fuel is a point that can be monitored and also provides adequate margin above 2/3 core height to allow effective action to be taken prior to reaching the 2/3 core height. This change will not alter process variables, or operation of structures, systems, or components as described in the safety analysis. This change will not alter any analysis assumptions. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   This proposed change relaxes the reactor vessel water level Safety Limit in the current Technical Specifications. This change will not alter the plant configuration (no new or different types of equipment will be installed). It also will not change the methods governing normal plant operation. This change imposes different requirements for reactor vessel water level than exist in the current Safety Limits. However, the change still ensures that water level is adequately maintained. The safety analysis assumes that water level does not drop below 2/3 core height. The proposed change requires water level to be maintained above the top of the active irradiated fuel. This proposed level is greater than the level assumed in the safety analysis; thus it is encompassed by the current analysis. Therefore, this change will not create the possibility of a new or different kind of accident from any previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

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

The proposed change relaxes the current Safety Limit requirement on the reactor vessel water level. This change will not significally affect the margin of safety. This change is consistent with the current safety analysis assumptions that indicate the reactor vessel water level will not go below 2/3 core height. The proposed Safety Limit will require reactor vessel water level be maintained above the top of active irradiated fuel. The proposed Safety Limit has been established at the top of active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action to be taken. The margin of safety would only be affected if the water level is maintained below 2/3 core height. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The requirement in CTS 6.7.B to immediately report each safety limit violation to the NRC by the Site Executive Officer has been extended to 1 hour in accordance with 10 CFR 50.72. 10 CFR 50.72 requires notification of the NRC Operations Center within 1 hour for events requiring the initiation of a nuclear plant shutdown required by the Technical Specifications. In addition, the explicit requirement to report a safety limit violation is not retained in the ITS since the requirement is duplicative of other regulations. The time provided to report safety limit violations do not cause or influence the consequences of accidents. Restoring the safety limits to within limits and inserting all insertable control rods is important for safe operation of the plant. These requirements have been retained in the Technical Specifications, therefore this change will not significantly increase the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The requirement in CTS 6.7.B to immediately report each safety limit violation to the NRC by the Site Executive Officer has been extended to 1 hour in accordance with 10 CFR 50.72. 10 CFR 50.72 requires notification of the NRC Operations Center within 1 hour for events requiring the initiation of a nuclear plant shutdown required by the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

3. (continued)

Technical Specifications. In addition, the explicit requirement to report a safety limit violation is not retained in the ITS since the requirement is duplicative of other regulations. Times to report safety limit violations do not cause or influence the consequences of accidents. Restoring the safety limits to within limits and inserting all insertable control rods is important for safe operation of the plant. These requirements have been retained in the Technical Specifications. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that itdoes not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change reduces the Applicability of the low steam dome pressure or low core flow Reactor Core SL. However, the high steam dome pressure and high core flow Reactor Core SL Applicability is increased to be consistent with GE critical power correlations. Safety Limit Applicability is not considered in the initiation of any accident previously evaluated. Therefore, this change will not significantly increase the probability of an accident previously evaluated. In the CTS, MCPR is the limit if reactor pressure is greater than 785 psig and if core flow is greater than 10% of rated. In addition, 25% rated thermal power is the limit if reactor pressure is less than <u>or equal to</u> 785 psig or if core flow is less than <u>or equal to</u> 10% of rated. In the ITS (2.1.1.1 and 2.1.1.2), MCPR is the limit if reactor pressure is greater than <u>or equal to</u> 785 psig and core flow is greater than <u>or equal to</u> 10% of rated, and 25% rated thermal power is the limit if reactor pressure is less than 785 psig or core flow is less than 10% of rated. This change was made since the GE critical power correlations are accurate at pressures greater than or equal to 785 psig and at core flows greater than or equal to 10% of rated. Justification for operation without thermal margin monitoring below 25% RTP is that the individual assembly power is conservatively estimated to be ≤ 3.35 MWt where design basis peaking factors are considered and since the individual assembly flow with natural circulation induced by normal water level and power in the range of 20 to 25% RTP is sufficient to maintain adequate thermal margin for the fuel. The proposed change is slightly less restrictive since the Applicability of CTS 1.1.B (the low pressure or low core flow SL limit) is reduced, however to compensate the Applicability of CTS 1.1.A (high pressure and high core flow SL) has been increased. This change is acceptable since it is consistent with GE critical power correlations. These safety limits will help ensure the fuel will operate within design criteria that 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling as a result of normal operation and abnormal operational transients. Therefore, this change will not significantly increase the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

1. (continued)

consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change reduces the Applicability of the low steam dome pressure or low core flow Reactor Core SL. However, the high steam dome pressure and high core flow Reactor Core SL Applicability is increased to be consistent with GE critical power correlations. In the CTS, MCPR is the limit if reactor pressure is greater than 785 psig and if core flow is greater than 10% of rated. In addition, 25% rated thermal power is the limit if reactor pressure is less than <u>or equal to</u> 785 psig or if core flow is less than <u>or equal to</u> 10% of rated. In the ITS (2.1.1.1 and 2.1.1.2), MCPR is the limit if reactor pressure is greater than <u>or equal to</u> 785 psig and core flow is greater than <u>or equal to</u> 10% of rated, and 25% rated thermal power is the limit if reactor pressure is less than 785 psig or core flow is less than 10% of rated. This change was made since the GE critical power correlations are accurate at pressures greater than or equal to 785 psig and at core flows greater than or equal to 10% of rated. Justification for operation without thermal margin monitoring below 25% RTP is that the individual assembly power is conservatively estimated to be ≤ 3.35 MWt where design basis peaking factors are considered and since the individual assembly flow with natural circulation induced by normal water level and power in the range of 20 to 25% RTP is sufficient to maintain adequate thermal margin for the fuel. The proposed change is slightly less restrictive since the Applicability of CTS 1.1.B (the low pressure or low core flow SL limit) is reduced, however to compensate the Applicability of CTS 1.1.A (high pressure and high core flow SL) has been increased. This change is acceptable since it is consistent with GE critical power correlations. This change is acceptable since it is consistent with GE critical power correlations. These safety limits will help ensure the fuel will

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

3. (continued)

operate within design criteria that 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling as a result of normal operation and abnormal operational transients. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The CTS contains a SL for reactor steam dome pressure when operating in the RHR shutdown cooling mode. The function associated with this SL is provided to isolate the shutdown cooling portion of the RHR System for equipment protection in order to prevent an intersystem LOCA. This function is not taken credit for in the UFSAR as a SL and is more appropriately relocated to ITS 3.3.6.1. The placement of this function in ITS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," will ensure that this function is Operable when the reactor is pressurized in Modes 1, 2, and 3. Since this function is retained in the ITS and intersystem LOCA protection for the shutdown cooling portion of the RHR System is maintained, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not introduce a new mode of plant operation and does not involve physical modifications to the plant. The Shutdown Cooling System isolation function remains in the ITS with the nominal trip setpoint specified in the CTS being replaced with an Allowable Value in the ITS. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The CTS SL when operating the RHR System in the shutdown cooling mode will be relocated to ITS 3.3.6.1. The UFSAR does not take credit for this function as a plant SL, but as protection for the RHR System from an intersystem LOCA. The placement of this function in ITS Table 3.3.6.1-1 for the "Primary Containment Isolation Instrumentation" will continue to ensure its Operability when required by design. ITS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

3. (continued)

Table 3.3.6.1-1 is more explicit in listing the Operability requirements than CTS SL 1.2.2. ITS Table 3.3.6.1-1 provides necessary MODES of Operation, Required Channels per Trip System, Actions for inoperable equipment, and SRs. The ITS will require the reactor steam dome pressure interlock to be Operable in Modes 1, 2, and 3 when the reactor can be pressurized and, thus, when equipment protection is needed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

MARKUP OF NUREG-1433, REVISION 1 SPECIFICATION

[1.1][1.2]

2.0 SAFETY LIMITS (SLs)

[1.1][1.2] 2.1 SLs

[1.1] 2.1.1 Reactor Core SLs

[1.1.B] 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.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow: RAI 2.0-1

[1.1.A] MCPR shall be ≥ 1.09 ~~1.07~~ for two recirculation loop operation or ≥ 1.10 ~~1.08~~ for single recirculation loop operation. DB1

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

[1.2.1] 2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

[6.7] 2.2 SL Violations

within 2 hours

[6.7.A] [M2] With any SL violation, the following actions shall be completed:

~~2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.2 Within 2 hours:~~

TA1

[M2] 2.2.~~2~~.1 Restore compliance with all SLs; and

[6.7.A] 2.2.~~2~~.2 Insert all insertable control rods.

~~2.2.3 Within 24 hours, notify the [General Manager—Nuclear Plant and Vice President—Nuclear Operations].~~

(continued)

2.0 SLs

2.2 SL Violations (continued)

2.2.4 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC and the [General Manager—Nuclear Plant and Vice President—Nuclear Operations].

2.2.5 Operation of the unit shall not be resumed until authorized by the NRC.

TA 1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 2.0 - SAFETY LIMITS (SLs)

RAI 2.0-1

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 Not Used.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 5, Revision 1 have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

MARKUP OF NUREG-1433, REVISION 1, BASES

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES JAFNPP design criteria DB4

BACKGROUND

~~GDC 10~~ (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and ~~anticipated~~ abnormal operational ~~occurrences (AOOs)~~ transients. PA1

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 ~~[for [both General Electric Company (GE) and Advanced Nuclear Fuel Corporation (ANF) fuel]~~. DB1 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 ~~AOOs~~ abnormal operational transients, at least 99.9% of the fuel rods in the core do not experience transition boiling. PA1

(continued)

BASES

BACKGROUND (continued)

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.

PA2

fission products

INSERT BKGD

PA3

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 an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

PA4

abnormal operational transients

PA1

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 ~~[General Electric Company (GE) Fuel]~~

DB1

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

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia

(continued)

PA3

INSERT BKGD

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.

BASES

APPLICABLE SAFETY ANALYSES

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

DB1

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.

PA4

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

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

~~Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow is > 30 x 10^3 lb/hr. For the ANF 8x8 fuel design, the minimum bundle flow is > 28 x 10^3 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.~~

DB1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2a MCPR [GE Fuel] DOC 1

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.

~~2.1.1.2b MCPR [ANF Fuel]~~

~~The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent~~ DOC 1

(continued)

BASES

APPLICABLE SAFETY ANALYSES

DB1

~~2.1.1.2b MCPR [ANF Fuel] (continued)~~

~~in the XN-3 critical power correlation. Reference 3 describes the methodology used in determining the MCPR SL.~~

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

PA2

The reactor vessel water level is required to be above the top of active irradiated fuel.

The top of active irradiated fuel is the top of a 150 inch fuel column which includes both the enriched and the natural uranium.

PA5

2.1.1.3 Reactor Vessel Water Level

irradiated PA5

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

DB2

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

PA4

SAFETY LIMITS

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

RAI 2.0-1

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

~~2.2.1~~

~~If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 4).~~

2.2.2

PA4 4 1B1

TA1

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

(continued)

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If any SL is violated, the [senior management of the nuclear plant and the utility Vice President—Nuclear Operations] shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the appropriate utility management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the [senior management of the nuclear plant and the utility Vice President—Nuclear Operations].

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

TA1

edit

REFERENCES

1. UFSAR, Section 16.6 ~~10 CFR 50, Appendix A, GDC 10.~~ DB4

2. NEDE-24011-P-A ~~(latest approved revision)~~ -13, General Electric Standard Application for Reactor Fuel, August 1996. DB1 DB3

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

~~4. 10 CFR 50.72.~~

4 ~~5.~~ 10 CFR 100.

~~6. 10 CFR 50.73.~~

DB1 numbering

TA1

DB2

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

tip

REVISION D

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the 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 steam dome pressure ensures continued RCS integrity. According to ~~10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design"~~ (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and ~~anticipated~~ operational ~~occurrences (AOOs)~~.

JAF NPP design criteria

DB4

abnormal

transients

PA1

During normal operation and ~~AOOs~~, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

abnormal operational transients

PA1

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, ~~1971~~ 1965 Edition, including Addenda through the ~~winter of 1972~~ 1966 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1, ~~1969~~ 1967 Edition, including Addenda through ~~July 1, 1970~~ 1969 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of ~~110~~ 120% of design pressures of ~~1250~~ 1148 psig for suction piping and ~~1500~~ 1274 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

DB3 A

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is ~~110~~ 120% of design pressures of ~~1250~~ 1148 psig for suction piping and ~~1500~~ 1274 psig for discharge piping. The most limiting of these allowances is the 110% of the ~~suction piping~~ reactor pressure vessel design pressure~~s~~; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

DB3

APPLICABILITY

SL 2.1.2 applies in all MODES.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

2.2.1

If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure 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 SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

TAI

2.2.3

If any SL is violated, the appropriate [senior management of the nuclear plant and the utility Vice President—Nuclear Operations] shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to appropriate utility management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the [senior management of the nuclear plant and the utility Vice President—Nuclear Operations].

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.5 (continued)

actions are completed before the unit begins its restart to normal operation.

TA1

REFERENCES

USFAR, Section 16.6

1. ~~10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.~~ DB4

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000. A PA4

4. 10 CFR 100.

5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 ~~1971~~ Edition, Addenda (winter of ~~1970~~) 1966. DB3

6. ASME, USAS, Nuclear Power Piping Code, Section B31.1, 1967 ~~1969~~ Edition, Addenda (July 1, 1970). with Addendum A. 1969.

~~7. 10 CFR 50.72.~~

~~8. 10 CFR 50.73.~~ TA1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RAI 2.0-1

<u>RETENTION OF EXISTING REQUIREMENT</u> (CLB)

CLB1 Not Used.

<u>PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT</u> (PA)

PA1 Changes have been made (additions, deletions, and/or plant specific value changes to the NUREG) to reflect the plant specific nomenclature.

PA2 Changes have been made to be consistent with the Specification or with other places in the Bases.

PA3 A description of the reactor vessel water level SL has been added, consistent with the background description of the other SLs.

PA4 Correction of typographical/grammatical error.

PA5 Editorial change made for clarity and consistency with the actual safety limit.

<u>PLANT-SPECIFIC DIFFERENCE IN THE DESIGN</u> (DB)

DB1 JAFNPP core does not contain Advanced Nuclear Fuel Corporation (ANF) Fuel. Therefore, the Bases have been modified accordingly to remove discussions and references to any fuel vendor. The Bases Sections have been renumbered, as required. References have been renumbered as required.

DB2 The proper Reference has been included for the Reactor Vessel Water Level Safety Limit.

DB3 Changes have been made (additions, deletions) to reflect the plant specific References.

DB4 JAFNPP was designed and under construction prior to the promulgation of Appendix A to 10 CFR 50 - General Design Criteria for Nuclear Power Plants. The JAFNPP Construction Permit was issued on May 20, 1970. The proposed General Design Criteria (GDC) were published in the Federal Register on July 11, 1967 (32 FR 10213) and became effective on February 20, 1971 (36 FR 3256). UFSAR, Section 16.6 - Conformance to AEC Design Criteria, describes the JAFNPP current licensing basis with regard to the GDC. ISTS statements concerning the GDC are modified in the ITS to reference UFSAR, Section 16.6.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 5, Revision 1 have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 2.0

SAFETY LIMITS (SLS)

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

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 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

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

MCPR shall be ≥ 1.09 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.

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.

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.

(continued)

BASES

BACKGROUND (continued)

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 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 ANALYSES

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

> 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

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.

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.

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APPLICABILITY

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

(continued)

BASES (continued)

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-13, General Electric Standard Application for Reactor Fuel, August 1996.

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.

EDITS

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the 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 steam dome pressure ensures continued RCS integrity. According to JAFNPP design criteria (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and abnormal operational transients.

During normal operation and abnormal operational transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME, Boiler and Pressure Vessel Code, 1965 Edition including Addenda through the winter of 1966 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to the USAS Nuclear Power Piping Code, Section B31.1.0, 1967 Edition, including Addendum A through 1969 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig for suction piping and 1274 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 120% of design pressures of 1148 psig for suction piping and 1274 psig for discharge piping. The most limiting of these allowances is the 110% of the reactor pressure vessel design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding the RCS pressure SL may cause immediate RCS failure 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 SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also assures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. UFSAR, Section 16.6.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWA-5000.
4. 10 CFR 100.
5. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda winter of 1966.
6. ASME, USAS, Nuclear Power Piping Code, Section B31.1.0, 1967 Edition, with Addendum A, 1969.