ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS

JPTS-88-020

New York Power Authority

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

LIST OF FIGURES

1.0 (cont'd)

surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ±25 percent. The interval as pertaining to instrument and electric surveillance shall never exceed one operating cycle. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

1. Minimum critical power ratio (MCPR)- Minimum value of the ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power for all fuel assemblies in the core.

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR.

3. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

4. Transition Boiling - Transition boiling means the boiling region between nucleate and film boiling. Transition boiling is the region in which both nucleate and film boiling occur intermittently with neither type being completely stable.

V. Electrically Disarmed Control Rod

To disarm a rod drive electrically, the four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred. Electrical disarming does not eliminate position indication.

W. High Pressure Water Fire Protection System

The High Pressure Water Fire Protection System consists of: a water source and pumps; and distribution system piping with associated post indicator valves (isolation valves). Such valves include the yard hydr[illegible]h valves and the first valve ahead of the water flow alarm de[illegible] on each sprinkler or water spray subsystem.

X. Staggered Test Basis

A Staggered Test Basis shall consist of:

a. A test schedule for "n" systems, subsystems, trains or other designated components obtained by dividing the specified test interval into "n" equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

Y. Rated Recirculation Flow

That drive flow which produces a core flow of 77.0×10^6 lb/hr.

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Z. Top of Active Fuel

The Top of Active Fuel, corresponding to the top of the enriched fuel column of each fuel bundle, is located 352.5 inches above vessel zero, which is the lowest point in the inside bottom of the reactor vessel. (See General Electric drawing No. 919D690BD.)

AA. Rod Density

Rod density is the number of control rod notches inserted expressed as a fraction of the total number of control rod notches. All rods fully inserted is a condition representing 100 percent rod density.

AB. Purge-Purging

Purge or Purging is the controlled process of discharging air or gas from a confinement in such a manner that replacement air or gas is required to purify the confinement.

AC. Venting

Venting is the controlled process of releasing air or gas from a confinement in such a manner that replacement air or gas is not provided or required.

AD. Core Operating Limits Report (COLR)

This report is the plant-specific document that provides the core operating limits for the current operating cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.A.4. Plant operation within these operating limits is addressed in individual Technical Specifications.

1.1 (cont'd)

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.

2.1 (cont'd)

A.

1.

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.

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 established in the Core Operating Limits Report (COLR). This limit must be adjusted for single loop operation as specified in the COLR.

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.

1.1 (cont'd)

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

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 18 inches above the Top of Active Fuel when it is seated in the core.

2.1 (cont'd)

(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:

$S \leq 120\%$ Power

d. APRM Rod Block Setting

The APRM Rod block trip setting shall be less than or equal to the limit specified in the Core Operating Limits Report (COLR). The setting must be adjusted for single loop operation as specified in the COLR.

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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 such that the minimum critical power ratio (MCPR) is no less than 1.04. MCPR >1.04 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 of 1.04) 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 as described in Reference 1. The uncertanties employed in deriving the Safety Limit are

1.1 (cont'd)

provided in Reference 1. 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 a MFLPD greater than the fraction of rated power (FRP), the APRM scram and rod block settings shall be adjusted as required in the Core Operating Limits Report.

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 x 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 x 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.

BASES

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the FitzPatrick Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2436 MWt. The analyses were based upon plant operation in accordance with the operating map given in the current load line limit analysis. In addition, 2436 MWt is the licensed maximum power level of FitzPatrick, and this represents the maximum steady-state power which shall not knowingly be exceeded.

The transient analyses performed for each reload are given in Reference 2. Models and model conservatism are also described in this reference. As discussed in Reference 4, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis, and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

Fuel cladding integrity is assured by the applicable operating limit MCPR for steady state conditions given in the Core Operating Limits Report (COLR). These operating limit MCPR's are derived from the established fuel cladding integrity Safety Limit, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient.

The most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO. The type of transients evaluated were increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit, the required operating limit MCPR in the Core Operating Limits Report is obtained.

The evaluation of a given transient begins with the system initial parameters shown in the current reload analysis and Reference 2 that are input to the core dynamic behavior transient computer programs described in Reference 2. The output of these programs along with the initial MCPR form the input for the further analyses of the thermally limited bundle with a single channel transient thermal hydraulic code. The principal result of the evaluation is the reduction in MCPR caused by the transient.

2.1 BASES (Cont'd)

The MCPR operating limits in the COLR are conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation is not permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

- The abnormal operational transients were analyzed to the licensed maximum power level.
- The licensed maximum power level is 2436 MWt.
- Analyses of transients employ adequately conservative values of the controlling reactor parameters.
- The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

2.1 BASES (Cont'd)

c. APRM Flux Scram Trip Setting (Run Mode) (cont'd)

rated power. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 80°F feedwater heating event, than would result with the 120% fixed high neutron flux scram trip. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower Operating Limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the cycle.

The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow referenced scram.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in the Core Operating Limits Report, when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by either (1) reducing the APRM scram and rod block settings or (2) adjusting the indicated APRM signal to reflect the high peaking condition.

Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR will be greater than the Safety Limit when the transient is initiated from the MCPR operating limits provided in the Core Operating Limits Report.

d. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus provides an added level of protection before APRM Scram. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control withdrawal. The flow variable trip setting parallels that of the APRM Scram and provides margin to scram, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

2. Reactor Water Low Level Scram Trip Setting

The reactor low water level scram is set at a point which will assure that the water level used in the Bases for the Safety Limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

2.1 BASES (Cont'd)

C. References

1. (Deleted)

2. "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

3. (Deleted)

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

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 50 msec.

B. Minimum Critical Power Ratio (MCPR)

During reactor power operation, the MCPR operating limit shall not be less than that shown in the Core Operating Limits Report.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type of frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

B. Maximum Fraction of Limiting Power Density (MFLPD)

The MFLPD shall be determined daily during reactor power operation at $\geq$25% rated thermal power and the APRM high flux scram and Rod Block trip settings adjusted if necessary as required by Specifications 2.1.A.1.c and 2.1.A.1.d, respectively.

3.1 (cont'd)

1. If anytime during reactor operation at greater than 25% of rated power it is determined that the operating limit MCPR is being exceeded, action shall then be initiated within fifteen (15) minutes to restore operation to within the prescribed limits. If the MCPR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall begin immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the MCPR is returned to within the prescribed limits.

4.1 (cont'd)

C. MCPR shall be determined daily during reactor power operation at $\geq$25% of rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

D. When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

E. Verification of the MCPR operating limits shall be performed in accordance with the Core Operating Limits Report.

3.1 BASES (cont'd)

Turbine control valves fast closures initiates a scram based on pressure switches sensing electrohydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative ($500 < P < 850$ psig) to the normal (EHC) oil pressure of 1,600 psig so that based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the start-up and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis in the current reload submittal for various core exposures are given in the Core Operating Limits Report (COLR).

The ECCS performance analyses assumed reactor operation will be limited to MCPR = 1.20, as described in NEDO-21662 and NEDC-31317P. The Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as given in the COLR.

Amendment No. ~~48~~, ~~64~~, ~~109~~

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Mode in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)(16)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Selections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	≤ 120/125 of full scale	X	X		8 Instrument Channels	A
3	IRM Inoperative		X	X		8 Instrument Channels	A
2	APRM Neutron Flux-Startup (15)	≤ 15% Power	X	X		6 Instrument Channels	A
2	APRM Flow Referenced Neutron Flux (Not to exceed 117%) (13)(14)	(12)			X	6 Instrument Channels	A or B
2	APRM Fixed High Neutron Flux (14)	≤ 120% Power			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1- (cont'd)

C. High Flux IRM.

D. Scram Discharge Volume High Level when any control rod in a control cell containing fuel is not fully inserted.

E. APRM 15% Power Trip.

7. Not required to be operable when primary containment integrity is not required.
8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.
10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.
11. See Section 2.1.A.1.
12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit established in the Core Operating Limits Report.
13. The Average Power Range Monitor scram functions varied as a function of recirculation flow (W). The trip setting of this function must be maintained in accordance with Specification 2.1.A.1.c.
14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.
15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.
16. *During the proposed Hydrogen Addition Test, the normal background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to $\leq$ three times the increased radiation levels. The test will be conducted at power levels > 80% of normal rated power. During controlled power reduction, the setpoint will be readjusted prior to going below 20% rated power without the setpoint change, control rod withdrawal will be prohibited until the necessary trip setpoint adjustment is made.

* This specification is in effect only during Operating Cycle 7.

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Figures 3.1-1 and 3.1-2 have been deleted

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
2	APRM Upscale (Flow Biased)	(8)	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	(8)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Downscale)	$\geq$ 2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	$\geq$ 2% of full scale	8 Inst. Channels	(1)
3	IRM Detector not in Start-up Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$ 86.4% of full scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Start-up Position	(3)	4 Inst. Channels	(1)
2 (4)(5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Water Level	$\leq$ 26.0 gallons per instrument volume	2 Inst. Channels	(9)(10)

NOTES FOR TABLE 3.2-3

1. For the Start-up and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM block need not be operable in run mode, and

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2.-3

the APRM and RBM rod blocks need not be operable in start-up mode. From and after the time it is found that the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. From and after the time it is found that the first column cannot be met for both trip systems, the systems shall be tripped.

2. IRM downscale is bypassed when it is on its lowest range.
3. This function is bypassed when the count rate is $\geq$ 100 cps.
4. One of the four SRM inputs may be bypassed.
5. This SRM Function is bypassed when the IRM range switches are on range 8 or above.
6. The trip is bypassed when the reactor power is $\leq$ 30%.
7. This function is bypassed when the Mode Switch is placed in Run.
8. The Flow Biased APRM Upscale and Rod Block Monitor trip level setpoint shall be less than or equal to the limit established in the Core Operating Limits Report.
9. When the reactor is subcritical and the reactor water temperature is less than 212°F, the control rod block is required to be operable only if any control rod in a control cell containing fuel is not fully inserted.
10. When one of the instruments associated with scram discharge instrument volume high water rod blocks is not operable, the trip system shall be tripped.

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3.3 and 4.3 BASES (cont'd)

rods have been withdrawn (e.g., groups A_{12} and A_{34}, it is demonstrated that the Group Notch made for the control drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 20% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

4. The Source Range Monitor (SRM) System performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per sec. assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transient cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of squattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage.

This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (e.g., MCPR limit). During use of such patterns, it is judged that testing of the RBM System prior to withdrawal of such rods to assure its operability will assure that improper withdraw does not occur. It is the responsibility of the Reactor Analyst to identify these limiting patters and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other qualified personnel may perform this function.

C. Scram Insertion Times

The Control Rod System is designated to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. Scram insertion time test criteria of Section 3.3.C.1 were used to generate the generic scram reactivity curve shown in NEDE-24011-P-A. This generic curve was used in analysis of non-pressurization transients to determine MCPR limits. Therefore, the required protection is provided.

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes of satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. These values are provided in the Core Operating Limits Report. If at anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been maintained in a filled condition; the discharge piping of the affected subsystem shall be vented from the high point of the system and water flow observed.

3. Whenever the HPCI or RCIC System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI or RCIC shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested each month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

3.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR given in the Core Operating Limits Report.

If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the LHGR is returned to within the prescribed limits.

4.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The LHGR shall be determined daily during reactor operation at $\geq$25% rated thermal power.

3.5 BASES (cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, RCIC, and HPCI are not filled, a water hammer can develop in this piping when the pump(s) are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is required to be operable. If a discharge pipe is not filled, the pumps the supply that line must be assumed to be inoperable for technical specification purposes. However, if a water hammer were to occur, the system would still perform its design function.

H. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50 Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting values for APLHGR are given in the Core Operating Limits Report. During Single Loop Operation a multiplier is applied to these values. The derivation of this multiplier can be found in Bases 3.5.K, Reference 1.

I. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

The LHGR shall be checked daily during reactor operation at 25% rated thermal power to determine if fuel burnup, or control rod movement, has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the ratio of local LHGR to average LHGR would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

Figures 3.5-3 through 3.5-14 have been deleted

5.0 DESIGN FEATURES

5.1 SITE

A. The James A. FitzPatrick Nuclear Power Plant is located on the PASNY portion of the Nine Mile Point site, approximately 3,000 ft. east of the Nine Mile Point Nuclear Station, Unit 1. The NPP-JAF site is on Lake Ontario in Oswego Country, New York, approximately 7 miles northeast of Oswego. The plant is located at coordinates north 4,819, 545.012 m, east 386, 968.945 m, on the Universal Transverse Mercator System.

B. The nearest point on the property line from the reactor building and any points of potential gaseous effluents, with the exception of the lake shoreline, is located at the northeast corner of the property. This distance is approximately 3,200 ft. and is the radius of the exclusion areas as defined in 10 CFR 100.3.

5.2 REACTOR

A. The reactor core consists of not more than 560 fuel assemblies. The fuel types present in the core are listed in the Core Operating Limits Report.

B. The reactor core contains 137 cruciform-shaped control rods as described in Section 3.4 of the FSAR.

5.3 REACTOR PRESSURE VESSEL

The reactor pressure vessel is as described in Table 4.2-1 and 4.2-2 of the FSAR. The applicable design codes are described in Section 4.2 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters and characteristics for the primary containment are given in Table 5.2-1 of the FSAR.

B. The secondary containment is as described in Section 5.3 and the applicable codes are as described in Section 12.4 of the FSAR.

C. Penetrations of the primary containment and piping passing through such penetrations are designed in accordance with standards set forth in Section 5.2 of the FSAR.

5.5 FUEL STORAGE

A. The new fuel storage facility design criteria are to maintain a K_{eff} dry <0.90 and flooded <0.95. Compliance shall be verified prior to introduction of any new fuel design to this facility.

(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle. The analytical bases used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

a. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P, latest approved version and amendments.

b. "James A. FitzPatrick Nuclear Power Planr SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, October, 1986 including latest errata and addenda.

c. "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions and supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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ATTACHMENT II

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS

JPTS-88-020

New York Power Authority

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

I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes to the James A. FitzPatrick Technical Specifications are contained in Attachment I and are described below. In addition to these changed pages, all text from the following pages has been either relocated or deleted, and these pages should be removed from the Technical Specifications: 8a, 10a, 31a, 43a, 47b, 135a, 135b, 135c, 135d, 135e, 135f, 135g, 135h, 135i, 135j, 135k, and 135l.

Page vii, List of Figures

Figures 3.1-1 and 3.1-2 are deleted and the pages combined.

Figures 3.5-3 through 3.5-14 are deleted and the pages combined.

Page 6, Specifications 1.0.U.1 and 2

Insert "Minimum value of the" to the definition of Minimum Critical Power Ratio.

Replace "as calculated by application of the GEXL correlation (Reference NEDE-10958)" with, "for all fuel assemblies in the core."

Delete "The design LHGR is 14.4 KW/ft for GE8x8EB fuel and 13.4 KW/ft for the remainder."

Page 6a, Specifications 1.0.AD

A new definition is added to read:

AD. Core Operating Limits Report (COLR)

This report is the plant-specific document that provides the core operating limits for the current operating cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.A.4. Plant operation within these operating limits is addressed in individual Technical Specifications.

Page 8, Specification 2.1.A.1.c.(1)

The scram trip setting formula and remainder of the column are replaced with, "less than or equal to the limit established in the Core Operating Limits Report (COLR). This limit must be adjusted for single loop operation as specified in the COLR."

Page 8a, Specification 2.1.A.1.c.(1) (cont'd)

Relocate this specification to page 8 and remove this page.

Page 9, Specification 2.1.A.1.c.(1) (cont'd)

Delete this specification in its entirety.

Page 10, Specification 2.1.A.1.d

The Rod Block trip setting formula and remainder of the column are replaced with, "less than or equal to the limit established in the Core Operating Limits Report (COLR). This limit must be adjusted for single loop operation as specified in the COLR."

This specification is relocated to page 9 and page 10 is now intentionally blank.

Page 12, Bases 1.1.A

Replace "MCPR operating conditions in specification 3.1.B" with, "MCPR operating limit in the Core Operating Limits Report."

Page 13, Bases 1.1.A

FIRST PARAGRAPH

Replace "at the beginning of each fuel cycle" with, "in Reference 1."

FOURTH PARAGRAPH

Replace "to a maximum LHGR of 14.4 KW/ft for GE8x8EB fuel and 13.4 KW/ft for the remainder" with, "by the maximum LHGR identified in the Core Operating Limits Report."

FIFTH PARAGRAPH

Insert an additional "0" into "1.0" to make the number of significant figures consistent with the specifications.

Replace "specifications 2.1.A.1.c and 2.1.A.1.d" with, "the Core Operating Limits Report."

Page 14, Bases 1.1.E

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

Delete Reference 3.

Page 15, Bases 2.1

FIRST PARAGRAPH

Replace "2535" with "of 2436."

Insert "MWt" after "2436."

THIRD PARAGRAPH

Insert "applicable" before "operating Limit."

Replace "MCPR's" with "MCPR."

Replace "Specification 3.1.B" with, "the Core Operating Limits Report COLR."

FOURTH PARAGRAPH

Replace "Specification 3.1.B" with, "the Core Operating Limits Report."

FIFTH PARAGRAPH

Replace "a" with "the," replace "program" with "programs," and replace "References 1 and 3" with "Reference 2."

Page 16, Bases 2.1 (cont'd)

FIRST PARAGRAPH

Replace "of specification 3.1.B" with, "in the COLR."

THIRD PARAGRAPH

Replace "will not be" with "is not."

In subparagraph (i), replace "a power level of 2535 MWt" with, "the licensed maximum power level."

Replace the roman numerals of subparagraphs i through iv with bullets (•).

Page 18, Bases 2.1.A.1.c

THIRD PARAGRAPH

Replace "Specification 2.1.A.1.c" with, "the Core Operating Limits Report."

FOURTH PARAGRAPH

Replace "Specification 3.1.B" with, "the Core Operating Limits Report."

Page 20, Bases 2.1.C

Delete References 1. and 3.

In Reference 2, replace "Fuel Application" with, "Standard Application for Reactor Fuel."

Page 30f, Specification 3.1.B and 3.1.B.1

Replace "Below:" and the subsequent specification 3.1.B.1 with, "in the Core Operating Limits Report."

Page 31, Specification 3.1.B.1 (cont'd) and 3.1.B.2

Delete these specifications in their entirety.

4.1.E

Replace this specification and all subparagraphs with, "Verification of the MCPR operating limits shall be performed in accordance with the Core Operating Limits Report."

Page 31a, Specifications 3.1.B.2 (cont'd), 3.1.B.3, 4 and 5, and 4.1.E.3

Deleted the note associated with Specification 3.1.B.2.

3.1.B.3 and 4

Delete these specifications in their entirety.

3.1.B.5

Replace "limiting value for" with, "operating Limit."

Renumber this specification 3.1.B.1 and relocate to page 31.

4.1.E.3

Delete this specification in its entirety.

Remove page 31a from the specifications.

Page 35, Bases 3.1.B

Replace "Specification 3.1.B" with, "the Core Operating Limits Report (COLR)."

Replace "Specification 3.1.B" with, "the COLR."

Page 41, Table 3.1-1

Replace the Trip Level Setting formula for the APRM Flow Referenced Neutron Flux Scram with "(12)."

Delete the reference to notes (12) and (17) from the Trip Function for the APRM Flow Referenced Neutron Flux Scram.

Page 43, Notes of Table 3.1-1 (cont'd)

Replace note 12 with the following:

12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit established in the Core Operating Limits Report.

Page 43a, Notes of Table 3.1-1 (cont'd)

Delete note 17 in its entirety.

Relocate notes 14 through 16 to page 43 and remove page 43a from the Technical Specifications.

Pages 47a and 47b, Figures 3.1-1 and 3.1-2

Delete both figures and combine the pages into a single page 47a-b.

Page 72, Table 3.2-3

Replace the Trip Level Setting formulas for both Flow Biased APRM Upscale and Rod Block Monitor Control Rod Blocks with, "(8)."

Page 73, Notes for Table 3.2-3

Replace note 8 with the following:

8. The Flow Biased APRM Upscale and Rod Block Monitor trip level setpoint shall be less than or equal to the limit established in the Core Operating Limits Report.

Page 74, Notes for Table 3.2-3 (cont'd)

Delete notes 11 and 12 in their entirety.

Remove the headings from this page and insert "This Page Intentionally Blank."

Page 102, 3.3 and 4.3 Bases, §B.5

THIRD PARAGRAPH

Replace "i.e., MCPR limits as shown in Specification 3.1.B" with, "e.g., MCPR limit."

Page 123, Specification 3.5.H

Replace the second and third sentences with, "These values are provided in the Core Operating Limits Report."

In specifications 4.5.G.2 and 3 on this page, restore the Amendment 132 changes inadvertantly deleted by Amendment 134.

Page 124, Specification 3.5.I

Replace "of 14.4 KW/ft for GE8x8EB fuel and 13.4 KW/ft for the remainder of the fuel" with, "given in the Core Operating Limits Report."

Specification 4.5.I

Replace "checked" with "determined" to accurately reflect that the LHGR is a calculated value, not an instrument reading.

Page 130, Bases 3.5.H

SECOND PARAGRAPH

In the third sentence, replace "Figures 3.5-11 through 3.5-14" with, "the Core Operating Limits Report."

Delete the fourth and fifth sentences in their entirety.

Replace the sixth sentence through the word "during" with, "A multiplier is applied to these values during."

Page 135a through 135l, Figures 3.5-3 through 3.5-14

These figures have been deleted and the pages combined into a single page 135a-135l.

Page 245, Specification 5.2.A

Delete the second sentence through the end of this Specification. In their place insert, "The fuel types present in the core are listed in the Core Operating Limits Report."

Page 254-c, Specification 6.9.A.4

Insert a new Specification 6.9.A.4 on a new page 254-c. The text of this specification is given in Attachment I.

Page 254-c thru 254-f

Renumber this page "254-d thru 254-f" to support the change described above.

II. PURPOSE OF THE PROPOSED CHANGES

The purpose of the proposed Technical Specification changes is to remove cycle-specific parameter limits in accordance with the guidance provided by the NRC in Generic Letter 88-16 (Reference 1). Use of the Generic Letter 88-16 alternative consists of three separate actions to modify the Technical Specifications:

1) The addition of a definition of a formal report that includes the values of cycle-specific parameter limits that have been established using an NRC-approved methodology and consistent with all applicable limits of the safety analysis. At FitzPatrick, the report will be titled, "Core Operating Limits Report."

2) The addition of an administrative reporting requirement to submit the Core Operating Limit Report to the NRC for information.

3) The modification of individual Technical Specifications to note that cycle-specific parameters shall be maintained within the limits provided in the Core Operating Limits Report.

The proposed Technical Specification changes are responsive to industry and NRC efforts to improve Technical Specifications, reduce the administrative burden on the NRC and the New York Power Authority, and permit future reloads to be accomplished without license amendments. The proposed changes are consistent with those discussed previously between the NRC and General Electric Co. as described in Reference 2.

The following Technical Specification parameters have been identified as cycle-specific limits that can be relocated to the Core Operating Limits Report:

1) Operating Limit Minimum Critical Power Ratio (MCPR);

2) Flow Dependent MCPR Limits;

3) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR);

4) Linear Heat Generation Rate (LHGR);

5) Flow-biased Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) settings; and

6) Fuel design features.

In addition, discussions contained in the Technical Specification Bases associated with the above parameters which are cycle-specific are modified in accordance with the guidance of Generic Letter 88-16.

The Authority is implementing these Generic Letter 88-16 changes during the Reload 9/Cycle 10 refueling outage. The Authority will prepare a Core Operating Limits Report (COLR) to support the reloaded core. The Cycle 10 COLR will be provided to the NRC upon issuance, but no later than at the startup of Cycle 10 as required by the proposed Technical Specifications. The core will be operated for the remainder of the current operating cycle with the Cycle 9 specific limits contained in the Technical Specifications and will commence Cycle 10 with the Cycle 10 specific limits in a PORC and SRC reviewed COLR. At no time will the core be operated without the cycle-specific limits in either the Technical Specifications or the COLR.

As part of this Technical Specification amendment, an additional change is also proposed. The bases for Specification 2.1 on pages 15 and 16 state that the abnormal operational transients were analyzed at a power of 2535 MWt, corresponding to 104 percent of the licensed maximum power level of 2436 MWt. However, the NRC has approved the GE transient analysis methods designated GEMINI methods, which use the nominal (100%) power level in transient analyses. Consequently, the Bases to Specification 2.1 are modified to state that transient analyses are performed at 100 percent power (the maximum licensed power level) consistent with the NRC approval given in Reference 3. This method of transient analysis was approved for FitzPatrick Cycle 8 operation in Amendment 109 to the Technical Specifications (Reference 4).

III. IMPACT OF THE PROPOSED CHANGES

A. Generic Letter 88-16 Changes

The current method of controlling reactor physics parameters to assure conformance with 10 CFR 50.36 is to specify the values determined to be within specified acceptance criteria, usually the limits of the safety analysis, using an approved calculation methodology. The proposed Technical Specification changes maintain control of the values of cycle-specific parameters and assure conformance to 10 CFR 50.36 by specifying the approved calculation methodology and approved acceptance criteria. The Core Operating Limits Report documents the specific values of parameter limits that are determined using these methods and that meet the acceptance criteria. The Technical Specifications continue to require that operation will remain within limits, and that required remedial actions are taken if the limits are not met.

The Core Operating Limits Report for each cycle, and any necessary mid-cycle revisions, will be provided to the NRC for information. This report will be reviewed by both PORC and SRC to provide a similar level of quality assurance and document control for the Core Operating Limits Report as for the Technical Specifications. This will ensure that the proper operating limits are being enforced.

B. Technical Specification Bases 2.1 Change

This change updates the Bases to reflect the power level used in the FitzPatrick transient analyses. With the introduction of the approved GEMINI methods, transient analyses are performed at the 100 percent power level. Previously, analyses were performed at a power level in excess of 100 percent to account for uncertainties in power level measurement as required by Regulatory Guide 1.49. However, with GEMINI methods, power level measurement uncertainty is accounted for instead by increasing the MCPR calculated with the GEMINI methods instead of the power level as used previously. The NRC has generically approved this method of accounting for power level measurement uncertainty in Reference 3 and has approved its use at FitzPatrick in Reference 4.

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.**

   A. Generic Letter 88-16 Changes:

   The proposed amendment merely moves cycle-specific parameter limits from the Technical Specifications to the Core Operating Limits Report. NRC approved methodologies will continue to be used as the basis for establishing those limits. The establishment of these limits in accordance with NRC-approved methodology and the incorporation of these results into the Core Operating Limits Report will ensure that proper steps have been taken to establish the values of these limits. Furthermore, the submittal of the Core Operating Limits Report to the NRC will allow the staff to continue to trend the values of these limits.

   B. Technical Specification Bases 2.1 Change:

   The use of 100 percent power in the analysis of abnormal operational transients using GEMINI methods has been reviewed and approved previously by the NRC for both generic and FitzPatrick specific application (see References 3 and 4). Power level measurement uncertainties are accounted for adequately in the MCPR Operating Limit, and the level of confidence that the MCPR Safety Limit will not be violated as a result of a transient is not reduced.

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

   No safety-related equipment, function, or plant operation will be altered as a result of the proposed changes. The changes do not create any new accident mode. The level of document control and quality assurance applied to the preparation and use of the Core Operating Limits Report will be equivalent to that applied to Technical Specifications.

3. **involve a significant reduction in a margin of safety.**

   A. Generic Letter 88-16 Changes:

   The proposed changes are administrative in nature and do not impact the operation of the plant in a manner that will reduce the margin of safety. The proposed amendment still requires operation within the limits determined using NRC-approved methods, and that appropriate remedial actions be taken if the limits are violated.

B. Technical Specification Bases 2.1 Change:

The MCPR Operating Limit continues to be determined using an approved methodology that conservatively accounts for power level measurement uncertainties. The same criterion for acceptable operation is maintained; that is, 99.9 percent of all fuel rods will not enter boiling transition in the event of the limiting transient. Therefore, the margin of safety is not reduced.

V. IMPLEMENTATION OF THE PROPOSED CHANGE

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

VI. CONCLUSION

The change, as proposed, does not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, it:

a. will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

b. will not increase the possibility of an accident or malfunction of a different type from any previously evaluated in the Safety Analysis Report;

c. will not reduce the margin of safety as defined in the basis for any technical specification; and

d. involves no significant hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

2. GE letter, J. S. Charnley to M. W. Hodges (NRC), "Acceptance Implementation of Generic Letter 88-16," dated August 8, 1989.

3. NRC letter, G. C. Lainas to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, 'GE Generic Licensing Reload Report,' Supplement to Amendment 11," dated March 22, 1986.

4. NRC letter, H. I. Abelson to J. C. Brons (NYPA), "Amendment 109 to Technical Specifications," dated April 3, 1987.

5. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report.

6. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER), dated November 20, 1972, and Supplements.