ATTACHMENT B

License
and
Technical Specification Changes

Lic. Page 3

T.S. Pages 6
7
8
9
18
20
27
54
127
127A
127A-1
205A
205C

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B. Technical Specifications

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

C. Records

Boston Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

D. Equalizer Valve Restriction

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

E.

(Deleted)

F. Fire Protection

The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1 through 3.1.19 of the NRC's Fire Protection Safety Evaluation (SE), dated December 21, 1978 for the facility. These modifications will be completed in accordance with the schedule in Table 3.1.

In addition, the licensee shall submit the additional information identified in Table 3.2 of this SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

The licensee is required to implement the administrative controls identified in Section 6 of the SE. The administrative controls shall be in effect by December 31, 1978.

G. Security Plan

The licensee shall maintain in effect and fully implement all provisions of the Commission-approved physical security plan including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.790, referred to as Pilgrim Nuclear Power Station Physical Plan, dated November 7, 1977 with Revision 2 dated May 26, 1978 and Revision 3 dated January 8, 1979.

4. This license is subject to the following condition for the protection of the environment: Boston Edison shall continue, for a period of five

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Reactor Pressure > 800 psia and Core Flow >10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit. A MCPR of 1.07 is hereinafter referred to as the Safety Limit MCPR.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia and/or Core Flow ≤ 10%)

When the reactor pressure is ≤ 800 psia or core flow is less than or equal to 10% of rated, the steady state core thermal power shall not exceed 25% of design thermal power.

C. Power Transient

The safety limit shall be assumed to be exceeded when scram is known to have been accomplished by a means other than the expected scram signal unless analyses demonstrate that the fuel cladding integrity safety limits defined in Specifications 1.1A and 1.1B were not exceeded during the actual transient.

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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.

Specification:

A. Neutron Flux Scram

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

1. Nuetron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

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

S≤.65W + 55% 2 loop
S≤.65W + 51.7% 1 loop

Where:

S = Setting in percent of rated thermal power (1998 MWt)

W = Percent of drive flow to produce a rated core flow of 69 M lb/hr.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

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

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.65W + 55\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{2 Loop}}$$

$$S \leq (0.65W + 51.7\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{1 Loop}}$$

Where,

FRP = fraction of rated thermal power (1998 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8 and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. <u>APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)</u>

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. <u>IRM</u>

The IRM flux scram setting shall be ≤120/125 of scale.

B. <u>APRM Rod Block Trip Setting</u>

The APRM rod block trip setting shall be:

$$S_{RB} \leq 0.65W + 42\% \quad \underline{\text{2 Loop}}$$

$$S_{RB} \leq 0.65W + 38.7\% \quad \underline{\text{1 Loop}}$$

Where,

S_{RB} = Rod block setting in percent of rated thermal power (1998 MWt)

W = Percent of drive flow required to produce a rated core flow of 69M lb/hr.

In the event of operating with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S_{RB} \leq (0.65\ W + 42\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{2 Loop}}$$

$$S_{RB} \leq (0.65\ W + 38.7\%) \left[\frac{FRP}{MFLPD}\right] \quad \underline{\text{1 Loop}}$$

Where,

FRP = fraction of rated thermal power

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8 and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

C. Reactor low water level scram setting shall be $\geq$ 9 in. on level instruments.

D. Turbine stop valve closure scram setting shall be $\leq$ 10 percent valve closure.

E. Turbine control valve fast closure setting shall be $\geq$ 150 psig control oil pressure at acceleration relay.

F. Condenser low vacuum scram setting shall be $\geq$ 23 in. Hg. vacuum.

G. Main steam isolation scram setting shall be $\leq$ 10 percent valve closure.

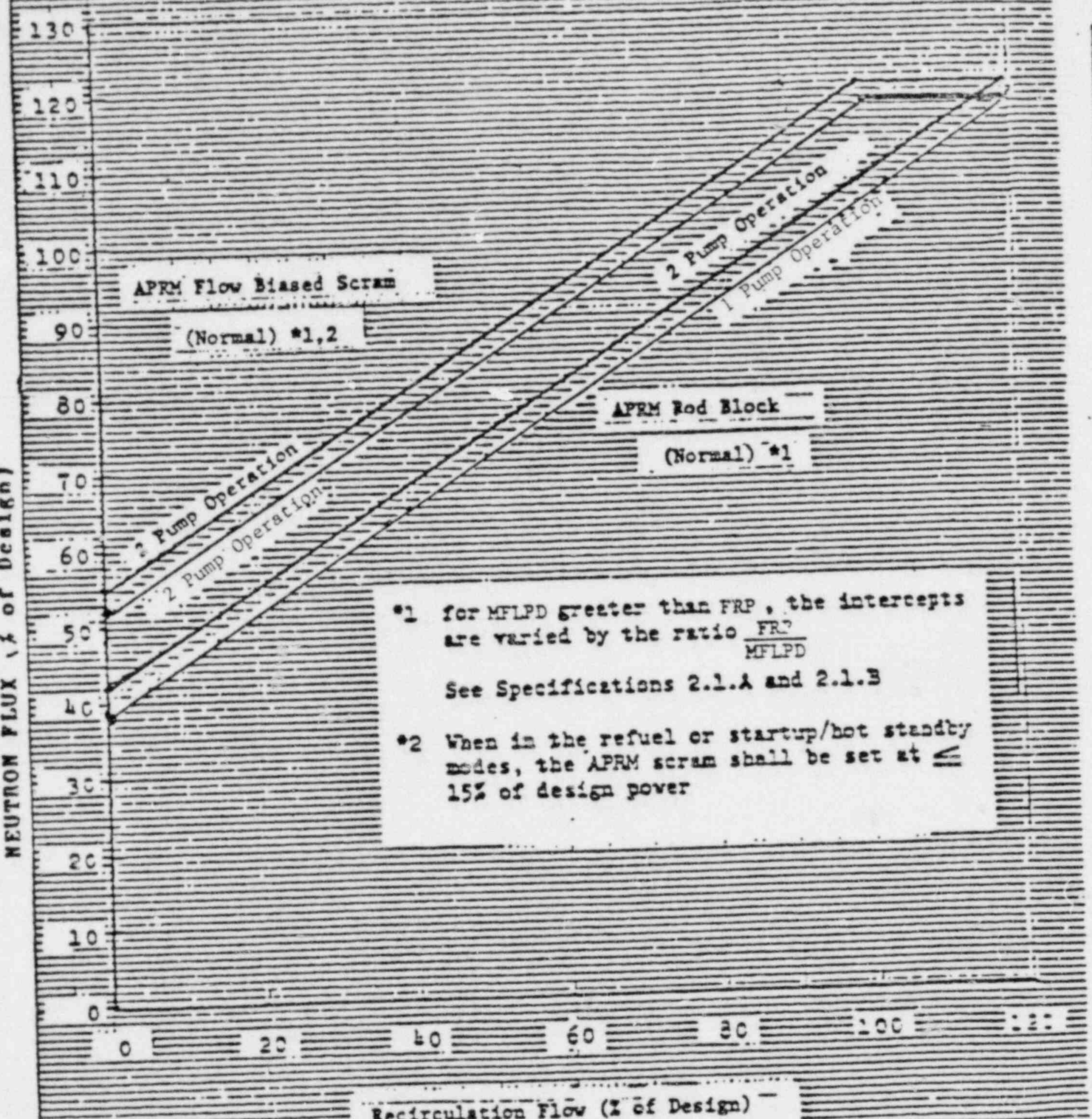

Figure 2.1.1

APRM Scram and Rod Block Trip Limiting Safety System Settings

2.1 BASES:

setting. 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 safety margin. Definition of single loop setpoints is given in Reference 2

C. Reactor Water Low Level Scram Trip Setting (LL1)

The set point for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results show that scram at this level adequately protects the fuel and the pressure barrier, because MCPR remains well above the safety limit MCPR in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 25 in. below the normal operating range and is thus adequate to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram Trip Setting

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

E. Turbine Control Valve Fast Closure Scram Trip Setting

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

F. Main Condenser Low Vacuum Scram Trip Setting

To protect the main condenser against overpressure, a loss of condenser vacuum initiates automatic closure of the turbine stop valves and turbine bypass valves. To anticipate the transient and automatic scram resulting from the closure of the turbine stop valves, low condenser vacuum initiates a scram. The low vacuum scram set point is selected to initiate a scram before the closure of the turbine stop valves is initiated.

2.1 BASES:

Transient and accident analyses demonstrate that these conditions result in adequate safety margins for the fuel.

References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

2. BECo. letter (J. E. Howard) to NRC (T. A. Ippolito) titled "Proposed Technical Specification Change Concerning Single Loop Operation", dated November 31, 1980.

Table 3.1.1 REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minm. Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action (1)
			Refuel (7)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	≤ 120/125 of full scale	X	X	(5)	A
3	Inoperative		X	X	(5)	A
	APRM					
2	High Flux	* (14) (15)	(17)	(17)	X	A or B
2	Inoperative		X	X(9)	X	A or B
2	Downscale	≥ 2.5 Indicated on Scale	(11)	(11)	X(12)	A or B
2	High Flux (15%)	≤ 15% of Design Power	X	X	(16)	A or B
2	High Reactor Pressure	≤ 1085 psig	X(10)	X	X	A
2	High Drywell Pressure	≤ 2.5 psig	X(8)	X(8)	X	A
2	Reactor Low Water Level	≥ 9 In. Indicated Level	X	X	X	A
2	High Water Level in Scram Discharge Tank	≤ 39 Gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	≥ 23 In. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	≤ 7X Normal Full Power Background	X	X	X	A or C
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3) (6)	X(3) (6)	X(6)	A or C
2	Turb. Cont. Valve Fast Closure	≥ 150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure	X(4)	X(4)	X(4)	A or D

*APRM high flux scram setpoint $\leq (.65W + 55)\left[\frac{FRP}{MFLPD}\right]$ Two recirc. pump operation

or $\leq (.65W + 51.7)\left[\frac{FRP}{\ }\right]$ One recirc. pump operation

PNPS

TABLE 3.2.C

INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum # of Operable Instrument Channels Per Trip Systems (1)	Instrument	Trip Level Setting
2	APRM Upscale (Flow Biased)	Two Loop $(0.65W + 42)\left[\frac{FRP}{MFLPD}\right]$ (2) One Loop $(0.65W + 38.7)\left[\frac{FRP}{MFLPD}\right]$
2	APRM Downscale	2.5 indicated on scale
1 (7)	Rod Block Monitor (Flow Biased)	Two Loop $(0.65W + 42)\left[\frac{FRP}{MFLPD}\right]$ (2) One Loop $(0.65W + 38.7)\left[\frac{FRP}{MFLPD}\right]$
1 (7)	Rod Block Monitor Downscale	5/125 of full scale
3	IRM Downscale (3)	5/125 of full scale
3	IRM Detector not in Startup Position	(8)
3	IRM Upscale	$\leq$108/125 of full scale
2 (5)	SRM Detector not in Startup Position	(4)
2 (5) (6)	SRM Upscale	$\leq 10^5$ counts/sec.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.D Safety and Relief Valves (Cont'd) pressure shall be below 104 psig within 24 hours.	
E. Jet Pumps 1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	E. Jet Pumps Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously: 1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed. 2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%. 3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from established jet pump P characteristics by more than 10%.
F. Jet Pump Flow Mismatch 1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than or equal to 80%. 2. When only one recirculation pump is in operation, a) The idle recirculation pump shall be electrically disarmed until such time as it can be restarted in accordance with 3.6.A.4 and 3.6.F.1 b) The recirculation controls shall be placed in the manual mode	F. Jet Pump Flow Mismatch Recirculation pump speeds shall be checked and logged at least once per day.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.F Jet Pump Flow Mismatch (Cont'd) c) Power level shall be limited to 50% d) Settings for the rod block monitor, APRM rod block trip and flow bias scram shall be modified for one loop operation in accordance with Specification 2.1 e) MAPLHGR limits shall be reduced as specified in 3.11A.	
G. Structural Integrity 1. The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1974	G. Structural Integrity 1. The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with AEC.

Edition (ASME Code, Section XI). In the interim until the nuclear system piping inspection evaluation level criteria of the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition, are completed, the applicable evaluation level provisions of the ASME Boiler and Pressure Vessel Code, Section XI, 1971 Summer Addenda shall be used in the Inservice Inspection of nuclear piping. Components of the primary system boundary whose in-service examination reveals the absence of flaw indications not in excess of the allowable indication standards of this code are acceptable for continued service. Plant operation with components which have in-service examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose in-service examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

   (i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications," of ASME Code, Section XI.

   (ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected components or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph a, above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive in-service inspection. An analysis and evaluation shall be submitted to the NRC following each in-service inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications," of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each in-service inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected components or require that the component be repaired or replaced.

c. Repair or replacement of components, including re-examinations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

3.11 REACTOR FUEL ASSEMBLY

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation with both recirculation pumps operating, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the applicable limiting value shown in Figures 3.11-1 through 3.11-5. The top curves are applicable for core flow greater than or equal to 90% of rated core flow. When core flow is less than 90% of rated core flow, the lower curves shall be limiting. For operation with one recirculation pump, values from these curves are to be multiplied by 0.65. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall 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, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 REACTOR FUEL ASSEMBLY

Applicability

The Surveillance Requirements apply to the parameters which the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. 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.

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

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

The peak cladding temperature (PCT) 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. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The limiting value for APLHGR is this LHGR of the highest powered rod divided by its local peaking factor.

The calculational procedure used to establish the APLHGR limit for each fuel type is based on a loss-of-coolant accident analysis. The emergency core cooling system (ECCS) evaluation models which are employed to determine the effects of the loss of coolant accident (LOCA) in accordance with 10CFR50 and Appendix K are discussed in Reference 1. The models are identified as LAMB, SCAT, SAFE, REFLOOD, and CHASTE. The LAMB Code calculates the short term blowdown response and core flow, which are input into the SCAT code to calculate blowdown heat transfer coefficients. The SAFE code is used to determine longer term system response and flows from the various ECC systems. Where appropriate, the output of SAFE is used in the REFLOOD code to calculate liquid levels. The results of these codes are used in the CHASTE code to calculate fuel clad temperatures and maximum average planar linear heat generation rates (MAPLHGR) for each fuel type.

The significant plant input parameters are given in Reference 2. MAPLHGR's for the present fuel types were calculated by the above procedure and are included in Reference 3. The curves in Figures 3.11-1 through 3.11-5 were generated by multiplying the values in Reference 3 by factors given in Reference 4. These multipliers were developed assuming no core spray heat transfer credit in the LOCA analysis.

General Electric is performing analyses for single loop operation of PNPS-1. Preliminary evaluation of these calculations performed according to the procudure outlined in NEDO-20566-2 indicates that a multiplier of 0.83 should be applied to the MAPLHGR limits for single loop operation of the PNPS-1. Further, GE has performed a large number of single loop analyses for similar plants; in no case has a multiplier of less than 0.70 been required. Because PNPS-1 does not have LPCI modification and because the limiting break is a suction line break, the single loop MAPLHGR multiplier is expected to be significantly better than for most other BWR's. Until the PNPS-1 calculations can be verified (as required by 10CFR.50 App. B) it is proposed that a multiplier of 0.65 be conservatively applied for single loop operation at the PNPS-1.