ATTACHMENT 2

SUMMARY OF PROPOSED

TECHNICAL SPECIFICATION/BASES CHANGES

FERMI 2 TECHNICAL SPECIFICATION/BASES CHANGES

Specification No./ Tables/Figure	Specification Title	Description of Change	Revised Page
Spec 1.8 (Definition) (add to Index)	CORE OPERATING LIMITS REPORT	Add definition and renumber the remaining definitions.	1.2 to 1.8, (i), (ii)
Spec 3/4.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE	Change reference to cycle-specific Figures 3.2.1-1 through 3.2.1-4 in LCO to the COLR. Change terminology "bundle type" to "fuel type" for consistency within the specification.	3/4 2-1
Figure 3.2.1-1 (delete in Index)	MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE (FUEL TYPE 8CR183)	Delete cycle-specific bundle MAPLHGR curve and relocate in the COLR. Exposure units changed from MWD/t to GWD/ST upon relocation to the COLR.	3/4 2-2 (xxi)
Figure 3.2.1-2 (delete in Index)	MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE (FUEL TYPE 8CR233)	Delete cycle-specific bundle MAPLHGR curve and relocate in the COLR. Exposure units changed from MWD/t to GWD/ST upon relocation to the COLR.	3/4 2-3 (xxi)
Figure 3.2.1-3 (delete in Index)	MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE (FUEL TYPE BC318D)	Delete cycle-specific bundle MAPLHGR curve and relocate in the COLR. Exposure units changed from MWD/t to GWD/ST upon relocation to the COLR.	3/4 2-4 (xxi)

FERMI 2 TECHNICAL SPECIFICATION/BASES CHANGES

Specification No./ Tables/Figure	Specification Title	Description of Change	Revised Page
Figure 3.2.1-4 (delete in Index)	MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE (FUEL TYPE BC318E)	Delete cycle-specific bundle MAPLHGR curve and relocate in the COLR. Exposure units changed from MWD/t to GWD/ST upon relocation to the COLR.	3/4 2-4a (xxi)
Spec 3/4.2.3	MINIMUM CRITICAL POWER RATIO	Delete the scram time equations and relocate in the COLR. Rephrase the ACTION statement to be non-cycle specific and consistent with the COLR. Change the reference to the MCPR figures in the ACTION statement to the COLR. Rephrase Surveillance Requirement 4.2.3.1 and 4.2.3.2 to be non-cycle specific and consistent with the COLR.	3/4 2-6 3/4 2-6a 3/4 2-7
Figure 3.2.3-1, 1A, and 1B (delete in Index)	EXPOSURE DEPENDENT MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS TAU AT RATED FLOW	Delete cycle-specific MCPR versus Tau curves at rated flow and relocate in the COLR. Figures in the COLR replace the term "operational mode" to "rod pattern" for clarity.	3/4 2-8 thru 3/4 2-8b (xxi)

FERMI 2 TECHNICAL SPECIFICATION/BASES CHANGES

Specification No./ Tables/Figure	Specification Title	Description of Change	Revised Page
Figure 3.2.3-2 (delete in Index)	FLOW CORRECTION (K_f) FACTOR	Delete the cycle-specific K_f curve and relocate in the COLR. Add equations for K_f in the COLR.	3/4 2-9 (xxi)
Spec 3/4.2.4	LINEAR HEAT GENERATION RATE	Delete the cycle-specific LHGR values and relocate the limiting LHGR values in the COLR. Rephrase LCO and Surveillance Requirements for clarity and to correct grammatical errors.	3/4 2-10
Spec 6.9.3 (add in Index)	CORE OPERATING LIMITS REPORT	Add adminstrative controls for report preparation and submittal in accordance with NRC Generic Letter 88-16 guidance.	6-21 (xx)
Bases 3/4.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE	Change the cycle-specific references to the ALPHGR figures which have been relocated in the COLR. Correct grammatical errors.	B 3/4 2-1

FERMI 2 TECHNICAL SPECIFICATION/BASES CHANGES

Specification No./ Tables/Figure	Specification Title	Description of Change	Revised Page
Bases Table B 3.2.1-1 (delete in Index)	SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS	Delete entire table. Table provides cycle-specific information and is no longer referenced in the Bases.	B 3/4 2-3 (xxv)
Bases 3/4.2.3	MINIMUM CRITICAL POWER RATIO	Delete the explanation of the cycle dependent MCPR curves. Change reference to figures to reference to the COLR. Change "MCPR" to "MCPR limit" for clarity. Change "thermal flow" to "core flow" to use correct terminology.	B 3/4 2-4 thru B 3/4 2-5
Bases 3/4.2.4	LINEAR HEAT GENERATION RATE	Expand definition of LHGR. Include reference to COLR.	B 3/4 2-5

ATTACHMENT 3

TECHNICAL SPECIFICATION

AND BASES CHANGES

INDEX

DEFINITIONS

SECTION

PAGE

DEFINITIONS

SECTION

DEFINITIONS (Continued)

ADMINISTRATIVE CONTROLS

LIST OF FIGURES

DELETED

MOVED TO COLR

LIST OF TABLES (Continued)

CORE ALTERATION

~~1.7~~ 1.8 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of SRMs, IRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

INSERT A

CRITICAL POWER RATIO

~~1.8~~ 1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of an NRC approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

~~1.9~~ 1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

Ē-AVERAGE DISINTEGRATION ENERGY

~~1.10~~ 1.11 Ē shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

~~1.11~~ 1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation set-point at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

~~1.12~~ 1.13 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

~~1.13~~ 1.14 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

Insert A

CORE OPERATING LIMITS REPORT

1.8 The CORE OPERATING LIMITS REPORT (COLR) is a plant specific document that provides selected core operating limits for the current reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.3. Plant operation within these core operating limits is addressed in individual specifications.

DEFINITIONS

FREQUENCY NOTATION

~~1.14~~ The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

1.15

IDENTIFIED LEAKAGE

~~1.15~~ IDENTIFIED LEAKAGE shall be:

1.16

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

~~1.16~~ 1.17 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

~~1.17~~ 1.18 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

~~1.18~~ 1.19 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

~~1.19~~ 1.20 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc., of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

~~1.20~~ 1.21 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

~~1.21~~ 1.22 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.22 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which
1.23 exists in the core.

OFF-GAS TREATMENT SYSTEM

1.23 An OFF-GAS TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting reactor coolant system offgases
1.24 from the reactor coolant and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

OFFSITE DOSE CALCULATION MANUAL

1.24 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses
1.25 due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the radiological environmental monitoring program.

OPERABLE - OPERABILITY

1.25 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s)
1.26 and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant
1.27 temperature as specified in Table 1.2.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation
1.28 and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault
1.29 in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

1.30 a. All primary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

6

2. Closed by at least one manual valve, blank flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirement of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The suppression chamber to reactor building vacuum breakers are in compliance with Specification 3.6.4.2.

THE PROCESS CONTROL PROGRAM

~~1.30~~ 1.31 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low-level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION, such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low-level radioactive waste disposal sites.

PURGE - PURGING

~~1.31~~ 1.32 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

~~1.32~~ 1.33 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.35 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.36 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.37 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed (except as noted in item g below).

e. The sealing mechanism associated with each secondary containment penetration, e.g., welds, bellows or O-rings, is OPERABLE.

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

g. Both railroad bay access doors are OPERABLE and closed except for ingress and egress or testing as specified by Specification 3.6.5.1.

SHUTDOWN MARGIN

1.38 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e., 68°F; and xenon free.

SITE BOUNDARY

1.39 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled, by the licensee.

DEFINITIONS

SOLIDIFICATION

~~1.39~~ 1.40 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the waste type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM (PCP).

SOURCE CHECK

~~1.40~~ 1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

~~1.41~~ 1.42 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.

THERMAL POWER

~~1.42~~ 1.43 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

~~1.43~~ 1.44 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

~~1.44~~ 1.45 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

~~1.45~~ 1.46 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

[redacted] A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

1.47

VENTING

[redacted] VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

1.48

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed:

a. The MAPLHGR limit which has been approved for the respective fuel and lattice type as a function of the average planar exposure (as determined by the NRC approved methodology described in GESTAR-II), or

b. When hand calculations are required, the most limiting lattice type MAPLHGR limit as a function of the average planar exposure shown in the ~~[illegible]~~ CORE OPERATING LIMITS REPORT for the applicable ~~bundle~~ fuel type.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the above limits, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits required by Specification 3.2.1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

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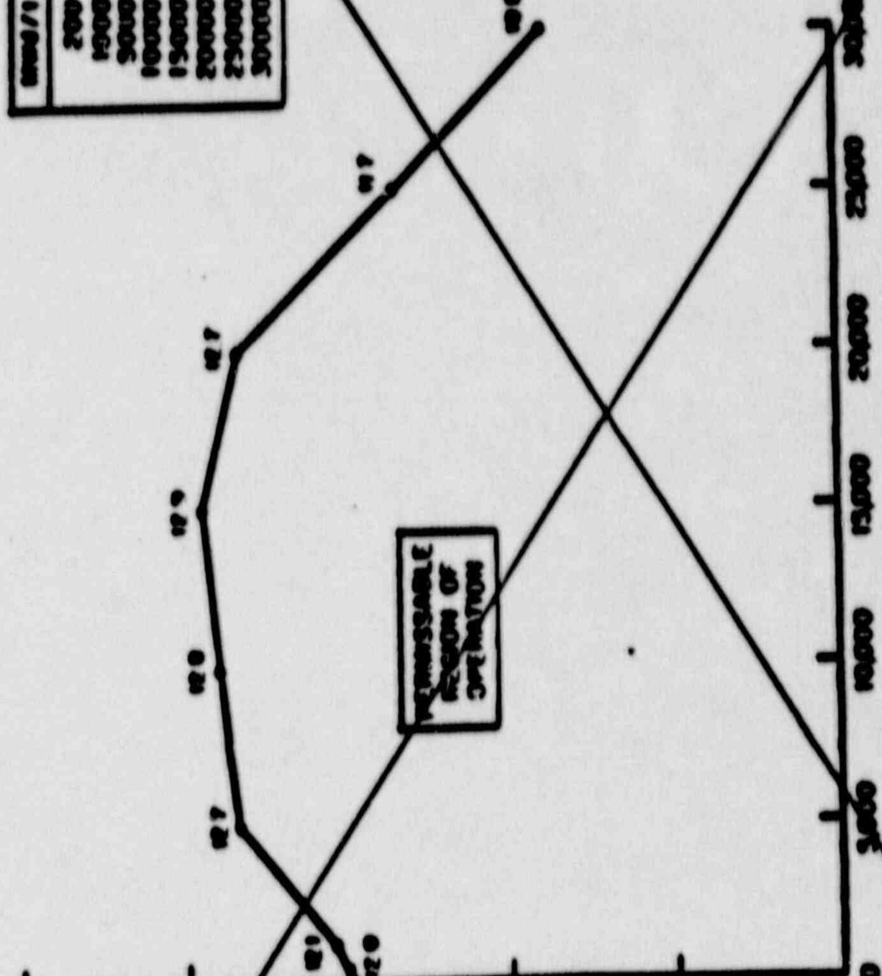

MWD/t	kW/ft
200	12.0
1000	12.1
5000	12.4
10000	12.8
15000	12.9
20000	12.7
25000	11.7
30000	10.8

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE INITIAL CORE FUEL TYPE 8CR183

FIGURE 3.2.1-1

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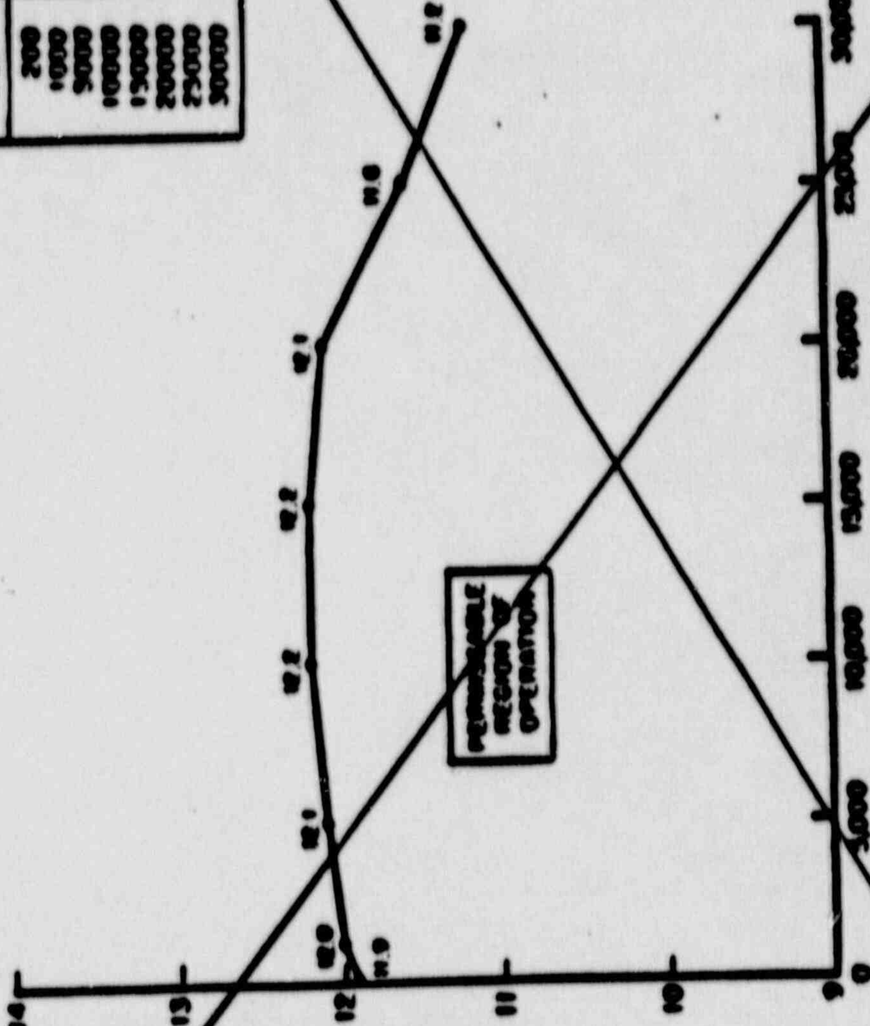
COLR

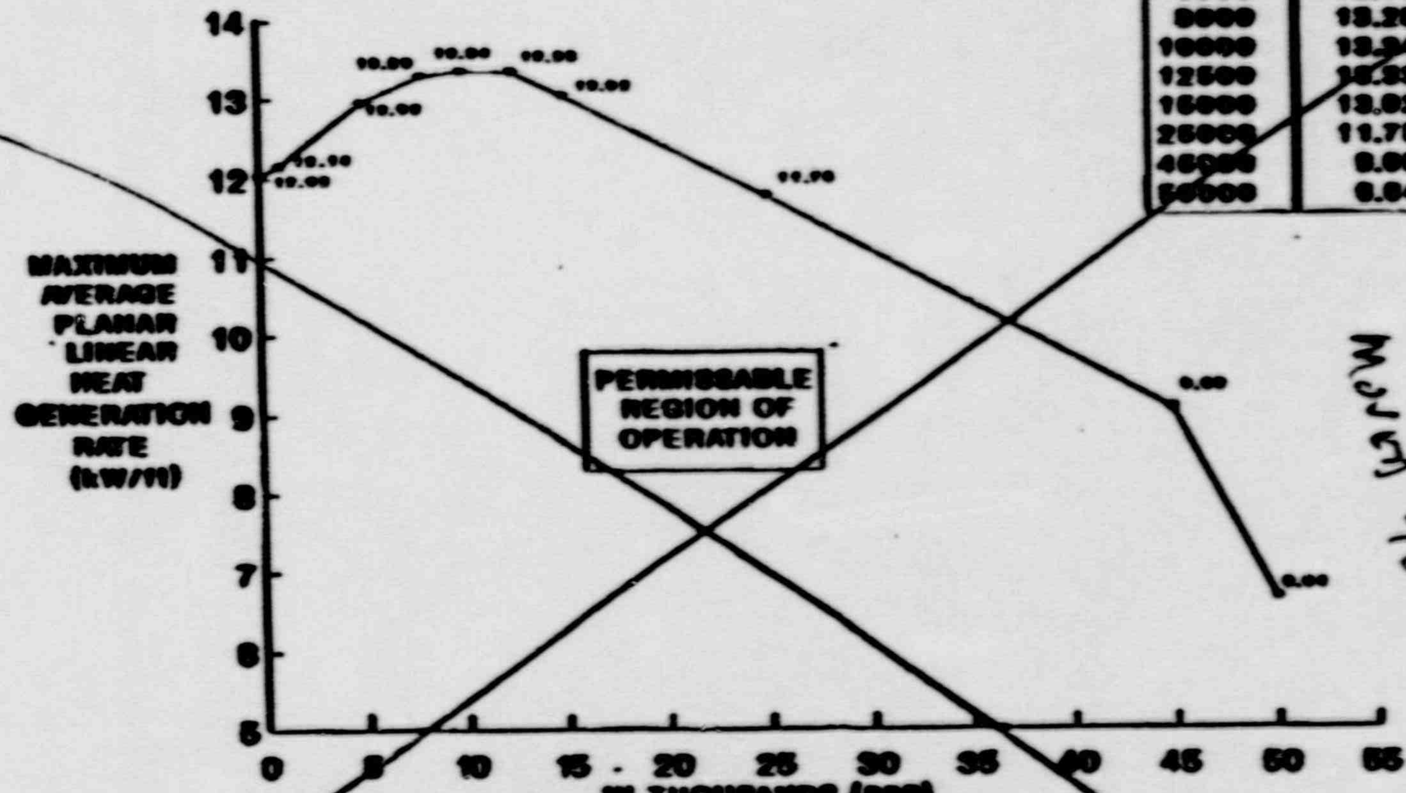
MWd/t	kW/ft
200	11.9
1000	12.0
5000	12.1
10000	12.2
15000	12.2
20000	12.1
25000	11.6
30000	11.2



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE INITIAL CORE FUEL TYPE

FIGURE 3.2.1-2

30



MWd/t	kW/ft
0	12.02
1000	12.14
5000	12.89
8000	13.20
10000	13.34
12500	13.33
15000	13.02
25000	11.75
45000	9.06
50000	6.84

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE RELOAD FUEL TYPE 8C318D

FIGURE 3.2.1-3

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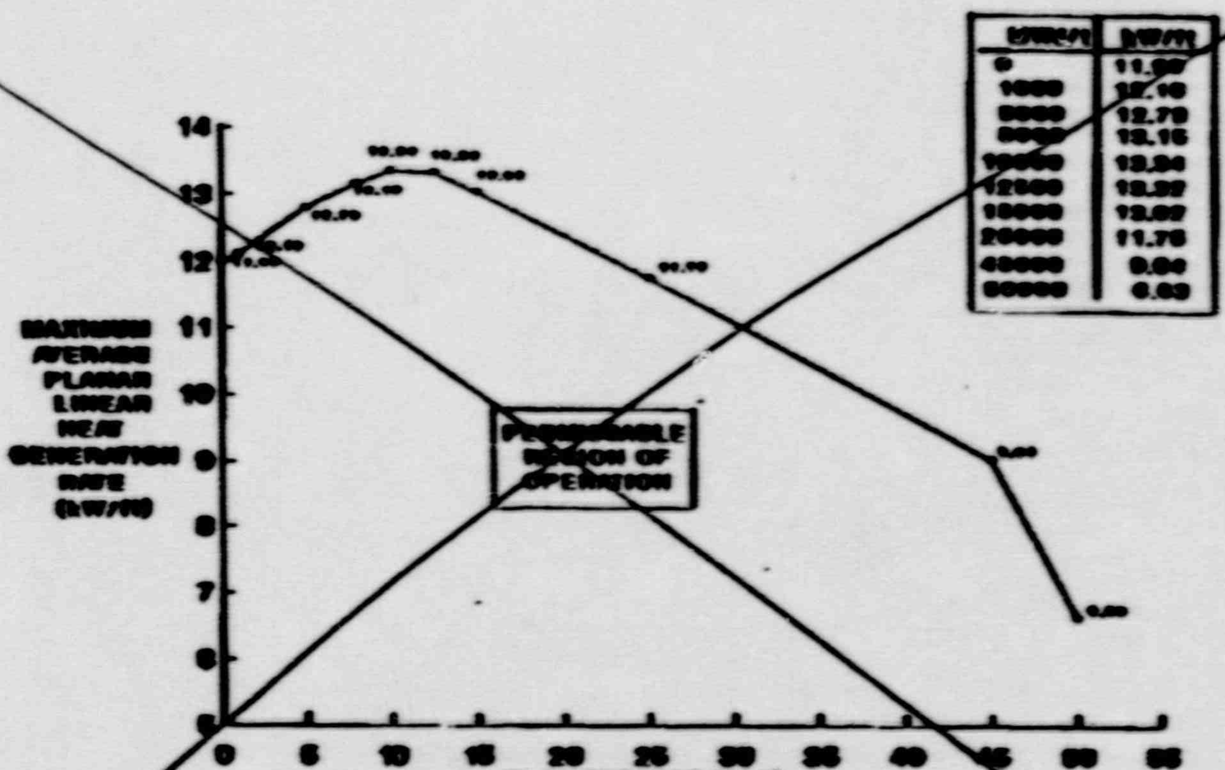

MWD/t	kW/ft
0	[illegible]
1000	[illegible]
5000	12.70
[illegible]	13.15
10000	[illegible]
12500	[illegible]
15000	[illegible]
20000	11.75
40000	[illegible]
50000	[illegible]

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE RELOAD FUEL TYPE [illegible]

FIGURE 3.2.1-4

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit ~~shown in Figures 3.2.3-1 thru 3.2.3-1B times the K_f shown in Figure 3.2.3-2, with:~~ specified in the CORE OPERATING LIMITS REPORT (COLR).

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

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where:

τ_A = 1.096 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.2.3.3,

$$\tau_B = 0.813 + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} 0.018,$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 36 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION

Replace with Insert B

a. Operating in the Control Cell Core (CCC) operating mode* and MCPR less than the applicable MCPR limit shown in Figures 3.2.3-1 thru 3.2.3-1B

*The CCC operating mode includes operation with only A2 rods, A1 shallow rods less than or equal to notch position 36, all peripheral rods inserted in the core, and rods inserted to position 46. Normal control rod operability checks, coupling checks, scram time testing, and friction testing of non-CCC control rods does not require the utilization of the more restrictive non-CCC operational mode MCPR limits. Any other operation is a non-CCC operating mode.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

(Curve A) times the applicable K_f curve shown in Figure 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

b. Operating in the non-CCC operating mode* and MCPR less than the applicable MCPR limit shown in Figures 3.2.3-1 thru 3.2.3-18 (Curve B) times the applicable K_f curve shown in Figure 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

c. Operating in either the CCC or non-CCC operating mode with either the main turbine bypass system inoperable per Specification 3.7.9 or the moisture separator reheater inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be equal to or greater than the MCPR limit as shown in Figures 3.2.3-1 thru 3.2.3-18 (Curve C) by the main turbine bypass or moisture separator reheater inoperable curve times the applicable K_f shown in Figure 3.2.3-2.

d. Operating in either the CCC or non-CCC operating mode with both the main turbine bypass system inoperable per Specification 3.7.9 and the moisture separator reheater inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be equal to or greater than the MCPR limit as shown in Figures 3.2.3-1 thru 3.2.3-18 by the main turbine bypass and moisture separator reheater inoperable curve times the applicable K_f shown in Figure 3.2.3-2.

INSERT B

*The CCC operating mode includes operation with only A2 rods, A1 shallow rods less than or equal to notch position 36, all peripheral rods inserted in the core, and rods inserted to position 46. Normal control rod operability checks, coupling checks, scram time testing, and friction testing of non-CCC control rods does not require the utilization of the more restrictive non-CCC operational mode MCPR limits. Any other operation is a non-CCC operating mode.

Insert B

ACTION

a. With MCPR less than the applicable MCPR limit in the COLR, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

b. With the main turbine bypass system inoperable and/or the moisture separator reheater inoperable per Specification 3.7.9, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be equal to or greater than the applicable MCPR limit in the COLR.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR, with:

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1 through 3.2.3-1B and 3.2.3-2:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 Prior to the use of Curve A and whenever Surveillance Requirement 4.2.3.1 is performed while using Curve A of Figures 3.2.3-1 through 3.2.3-1B, verify that all non-CCC control rods are fully withdrawn from the core. Non-CCC control rods are all control rods excluding A2 rods, A1 shallow rods inserted less than or equal to notch position 36, all peripheral rods, and rods inserted to position 46. Normal control rod operability checks, coupling checks, scram time testing, and friction testing of non-CCC control rods does not require the utilization of the more restrictive non-CCC operational mode MCPR limits.

INSERT C

Insert C

4.2.3.1 MCPR shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT (COLR):

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 Prior to the use of a MCPR limit which is based upon a specific control rod pattern and whenever Surveillance Requirement 4.2.3.1 is performed while using a MCPR limit based upon a specific control rod pattern, the required control rod pattern shall be verified.

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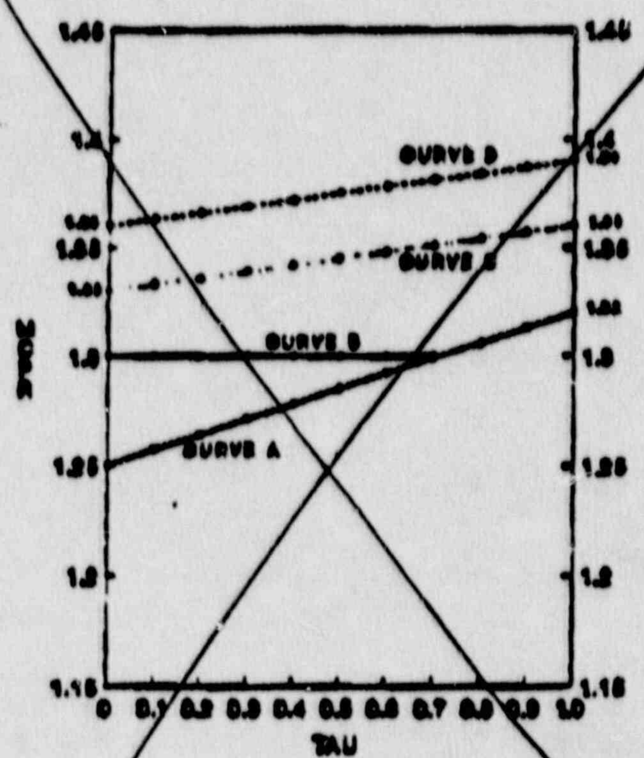

CURVE A - MCPR limit for [illegible] operational mode with both turbine bypass and moisture separator reheater in service.

CURVE B - MCPR limit for non-[illegible] operational mode with both turbine bypass and moisture separator reheater in service.

CURVE C - MCPR limit for both [illegible] or non-[illegible] operational modes with either turbine bypass or moisture separator reheater out of service.

CURVE D - MCPR limit for both [illegible] and non-[illegible] operational modes with both turbine bypass and moisture separator reheater out of service.

BOC TO 12,700 MWD/ST
MINIMUM CRITICAL POWER RATIO
(MCPR) VERSUS TAU AT RATED FLOW

FIGURE 3.2.3-1

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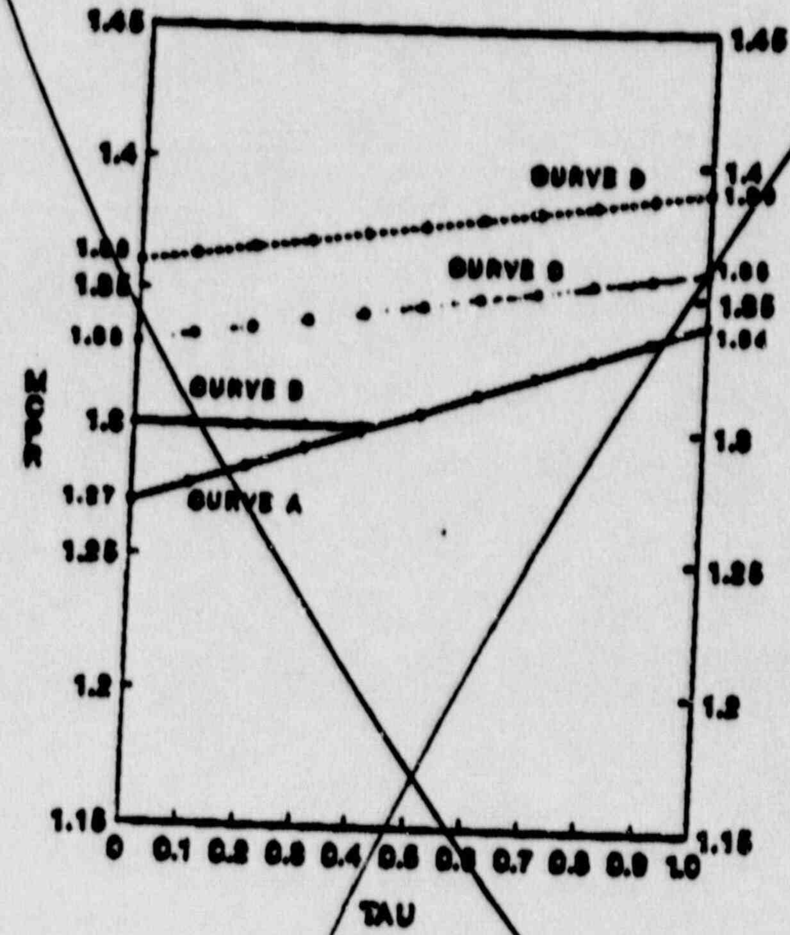

CURVE A - MCPR limit for OCC operational mode with both turbine bypass and moisture separator reheater in service.

CURVE B - MCPR limit for non-OCC operational mode with both turbine bypass and moisture separator reheater in service.

CURVE C - MCPR limit for both OCC or non-OCC operational modes with either turbine bypass or moisture separator reheater out of service.

CURVE D - MCPR limit for both OCC or non-OCC operational modes with both turbine bypass and moisture separator reheater out of service.

12,700 MWD/ST TO 13,700 MWD/ST
MINIMUM CRITICAL POWER RATIO
(MCPR) VERSUS TAU AT RATED FLOW

FIGURE 3.2.3-1A

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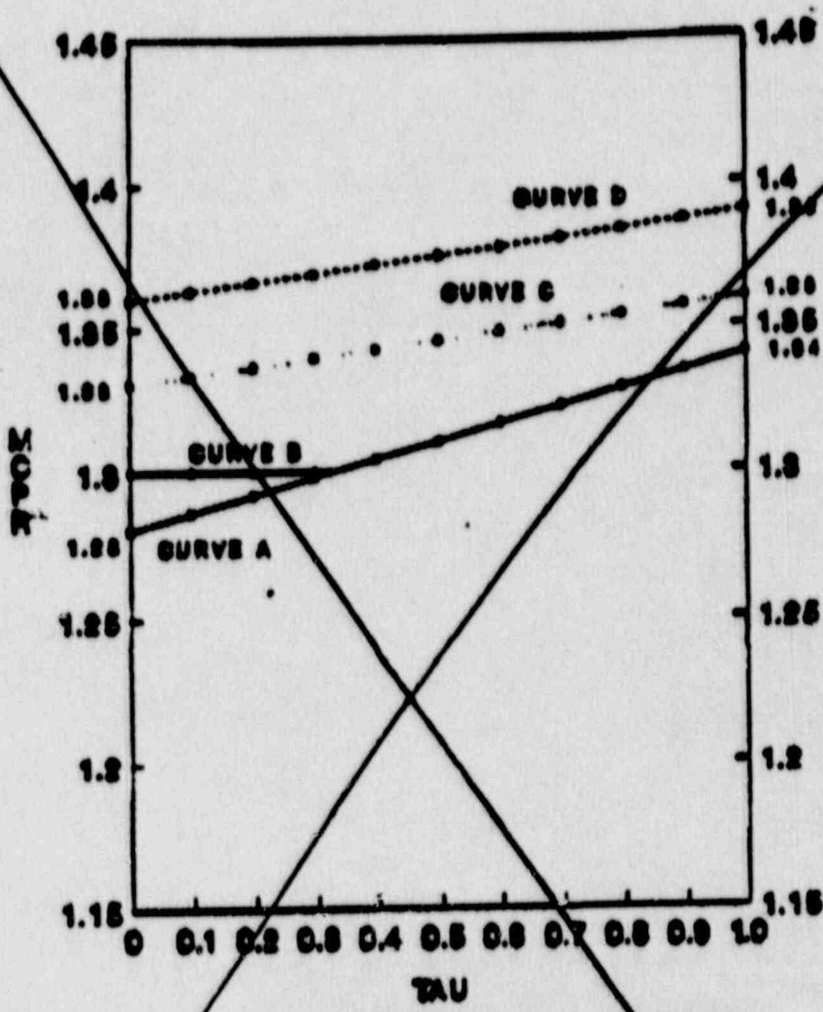

CURVE A - MCPR limit for EOC operational mode with both turbine bypass and moisture separator reheater in service.

CURVE B - MCPR limit for non-EOC operational mode with both turbine bypass and moisture separator reheater in service.

CURVE C - MCPR limit for both EOC and non-EOC operational modes with either turbine bypass or moisture separator reheater out of service.

CURVE D - MCPR limit for both EOC or non-EOC operational modes with both turbine bypass and moisture separator reheater out of service.

18,700 MWD/ST TO EOC
MINIMUM CRITICAL POWER RATIO
VERSUS TAU AT RATED FLOW

FIGURE 3.2.3-1B

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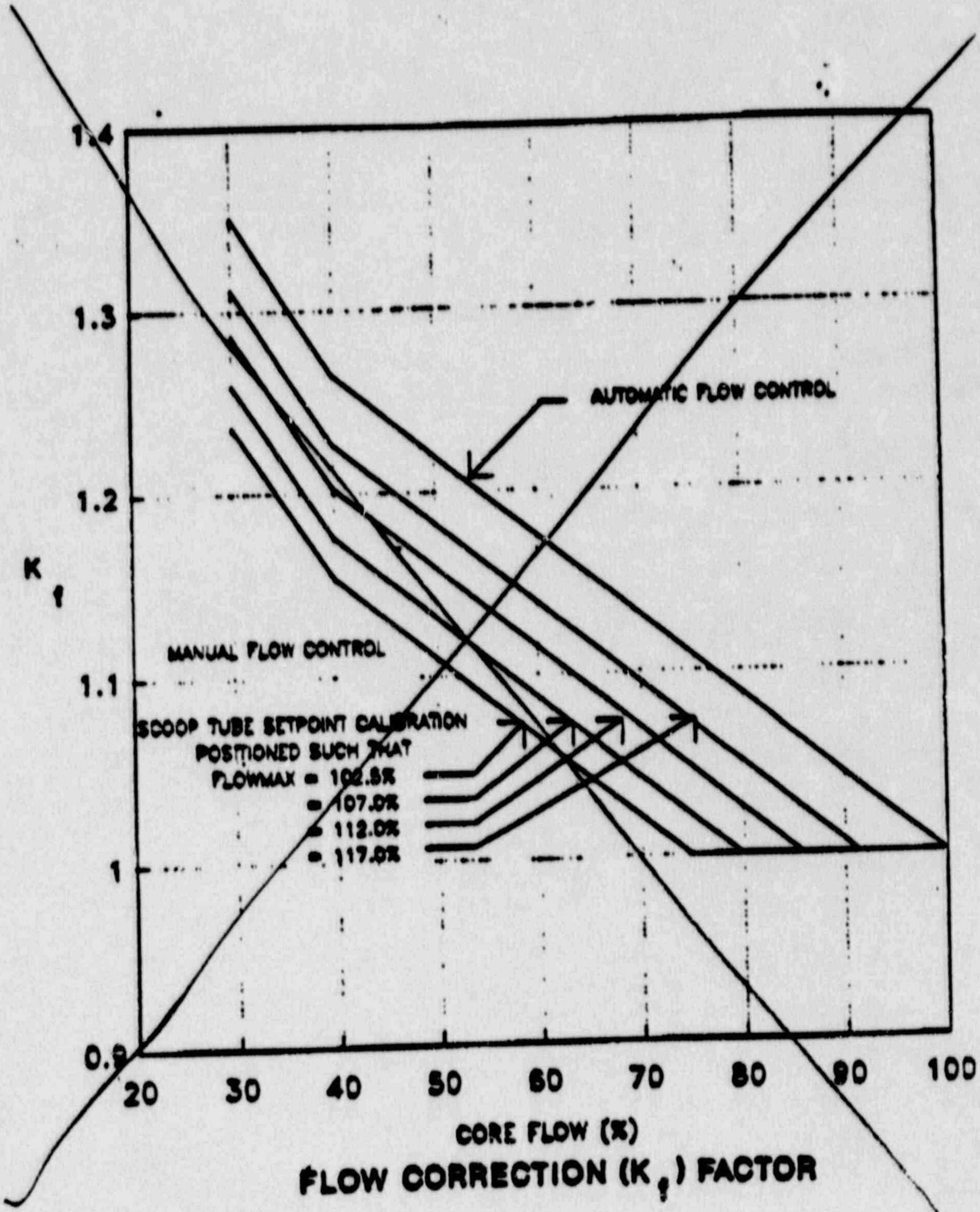

FLOW CORRECTION (K_f) FACTOR

FIGURE 3.2.3-2

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed [redacted] the LHGR limit specified in the CORE OPERATING LIMITS REPORT (COLR) for the applicable fuel type.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the applicable limit, initiate corrective action within 15 minutes and restore the LHGR to within the applicable limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 [redacted] LHGRs shall be determined to be equal to or less than the applicable limit:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN FOR LHGR.

d. The provisions of Specification 4.0.4 are not applicable.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

INSERT D

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

Insert D

CORE OPERATING LIMITS REPORT

6.9.3 Selected cycle specific 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 methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in GESTAR II (NEDE-24011-P-A). The core operating limits shall be determined so that all applicable limits (e.g., fuel 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 or supplement thereto, shall be submitted upon issuance to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector prior to use.

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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 dependent only 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 corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is [redacted] provided in the CORE OPERATING LIMITS REPORT (COLR).

The [redacted] COLR MAPLHGR value is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the ECCS MAPLHGR.

Fuel Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice, operating at the bounding power history, meet the fuel design limits specified in Reference 1. This bounding power history is used as the basis for the fuel design analysis MAPLHGR value.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the MAPLHGR values comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

Only the most limiting MAPLHGR values [redacted] are provided in the [redacted] COLR [redacted] for multiple lattice fuel. When hand calculations are required, these [redacted] COLR MAPLHGR [redacted] values for that fuel type are used for all lattices in that bundle.

For some fuel bundle designs MAPLHGR depends only on bundle type and burnup. Other fuel bundles have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle cross section at a particular axial node. Each particular combination of enriched uranium and gadolinia, for these fuel bundle types, is called a lattice type. These particular fuel bundle types have MAPLHGRs that vary by lattice (axially) as well as with fuel burnup.

Reference

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A ~~(latest approved revision)~~.

BASES TABLE B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core THERMAL POWER................. 3430 MWt* which corresponds to 105% of rated steam flow

Vessel Steam Output................ 14.86×10^6 lbm/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure......... 1055 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 4.1 ft^2

b. Small Breaks 0.1 ft^2

DELETE

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18
First Reload	8 x 8	14.4	1.4	1.18

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limiting MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transients 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 assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, 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 MCPR, the required minimum operating limiting MCPR of Specification 3.2.3 is obtained and presented in ~~Figures 3.2.3-1, 3.2.3-1A, 3.2.3-1B.~~ the CORE OPERATING LIMITS REPORT (COLR).

~~The MCPR curves illustrated in Figures 3.2.3-1 thru 3.2.3-1B were derived as described above for the following assumed operating conditions:~~

~~Curve A - MCPR limit with turbine bypass system, moisture separator reheater systems in service and CCC (Control Cell Core) operational mode (A2 rods, A1 shallows inserted less than or equal to notch position 36, all peripheral rods, and all rods inserted to position 46). The operating domain includes the 100% power/flow region and extended load line region with 100% power and reduced flow.~~

~~Curve B - MCPR limit with the turbine bypass system, moisture separator reheater systems in service and non-CCC operational mode (any non-CCC control rod inserted in the core). The operating domain includes the 100% power/flow region and the extended load line region with 100% power and reduced flow.~~

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~~Curve C - MCPR limit for either CCC or non-CCC operational modes with either the main turbine bypass system inoperative and the moisture separator reheater system available or the main turbine bypass system available and the moisture separator reheater system inoperable. The operating domain includes the 100% power/flow region and the extended load line region with 100% power with reduced flow.~~

~~Curve D - MCPR limit for either CCC or non-CCC operational modes with the main turbine bypass system inoperative and the moisture separator reheater system inoperable. The operating domain includes the~~

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

100% power/flow region and the extended load line region with 100% power and reduced flow.

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Curve A provides the MCPR limit assuming operation above 25 percent RATED THERMAL POWER with the turbine bypass system and moisture separator reheater in service. The curve was developed based upon the operating MCPR limits for a rod withdrawal error transient (UFSAR, Section 15.4.2) for operating within the CCC control rod patterns and a Main Turbine Trip with Turbine Bypass Failure transient (UFSAR, Section 15.2.3). CCC control rods are A2 rods, A1 shallow rods (inserted less than or equal to notch position 36), all peripheral rods, and all rods inserted to position 46. The analysis of the Main Turbine Trip with Turbine Bypass Failure takes credit for the steam flow to the moisture separator reheater.

Curve B provides the MCPR limit assuming operation above the 25 percent RATED THERMAL POWER with the turbine bypass system and moisture separator reheater system in service and non-CCC control rods inserted in the core. Non-CCC control rods are all rods excluding A2 rods, A1 shallow rods (inserted less than or equal to notch position 36), all peripheral rods, and all rods inserted to position 46. The curve was developed based upon the operating MCPR limits for a rod withdrawal error transient (UFSAR, Section 15.4.2) for any operating withdrawal sequence.

Curve C provides the MCPR limit assuming operation above the 25 percent RATED THERMAL POWER with the moisture separator reheater operable and turbine bypass system inoperable or the moisture separator reheater inoperable and the turbine bypass system operable. The curve was developed based upon the operating MCPR limits for several combinations of Feedwater Controller Failure.

Operation with main turbine bypass inoperable or with a moisture separator reheater inoperable results in a total reactor steam flow bypass capability of approximately 10 percent and 26 percent, respectively. The impact of operation with the moisture separator reheater inoperable but with bypass operable and utilization of Curve C is conservative because the 26 percent bypass capability is less limiting ith regard to the existing analysis used to establish Curve C which assumes only 10 percent bypass capability (with the main turbine bypass system inoperable). Therefore, the operation above 25-percent RATED THERMAL POWER with either the moisture separator reheater inoperable or main turbine bypass system inoperable is bounded by the existing Curve C.

Curve D provides the MCPR limit assuming operation above the 25 percent RATED THERMAL POWER with both the moisture separator reheater inoperable and the turbine bypass system inoperable. The curve was developed based upon the operating MCPR limits from the Feedwater Controller Failure.

There is no mode change restraint should the main turbine bypass or the moisture separator reheater be inoperable. However, should the main turbine

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3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

bypass system or the moisture separator reheater be inoperable as 25-percent RATED THERMAL POWER is exceeded, the MCPR check must be completed within one hour.

The evaluation of a given transient begins with the system initial parameters shown in UFSAR Table 15.0.1 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate transients are described in GESTAR II. The principal result of this evaluation is the reduction in MCPR caused by the transient.

(Reference 2)

in the CORE OPERATING LIMITS REPORT (COLR)

The purpose of the K_f factor ~~[illegible]~~ is to define operating limits at other ~~than rated~~ core flow conditions. At less than 100% of rated flow the ~~[illegible]~~ is the product of the ~~[illegible]~~ and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

than

in the COLR

The K_f factor values ~~[illegible]~~ were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow, although they are applicable for the extended operating region.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube setpoint and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

MCPR limit

MCPR limit at rated flow

POWER DISTRIBUTION LIMITS

BASES

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated ~~[illegible]~~ core flow.

The K_f factors ~~[illegible]~~ in the CORE OPERATING LIMITS REPORT (COLR) are conservative for the General Electric plant operation because the operating limit MCPRs ~~of Specification~~ in the COLR ~~[illegible]~~ are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

At THERMAL POWER levels less than or equal to 25 percent of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, a MCPR evaluation will be made at 25 percent of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25 percent of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3.4.2.4 LINEAR HEAT GENERATION RATE

~~This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.~~

INSERT E

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, ~~latest approved revision~~.

Insert E

The thermal expansion rates of UO_2 pellets and Zircalloy cladding are different in that, during heatup, the fuel pellet could come into contact with the cladding and create stress. If the stress exceeds the yield stress of the cladding material, the cladding will crack. The LHGR limit assures that at any exposure, 1% plastic strain on the clad is not exceeded. This limit is a function of fuel type and is presented in the COLR.

ATTACHMENT 4

Changes to Single Recirculation
Loop Operation Technical Specifications
(Reference 6)

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed:

a. The MAPLHGR limit which has been approved for the respective fuel and lattice type as a function of the average planar exposure (as determined by the NRC approved methodology described in GESTAR-II), or

b. When hand calculations are required, the most limiting lattice type MAPLHGR limit as a function of the average planar exposure shown in the ~~[illegible]~~ CORE OPERATING LIMITS REPORT (COLR) for the applicable ~~bundle~~ fuel type.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the above limits, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits required by Specification 3.2.1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

The MAPLHGR limit shall be reduced by a factor provided in the COLR during single loop operation.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation. ~~When total core flow is less than 45% of rated core flow, then THERMAL POWER must be less than or equal to the limit specified in Figure 3.4.1.1-1.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION: *Replace with Insert F*

a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

1. Monitor the APRM and LPRM** noise levels (Surveillance 4.4.1.1.3):
   a) Within 8 hours of entry into this condition and at least once per 24 hours thereafter while in this condition and,
   b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER in an hour by control rod movement.
2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

---

*See Special Test Exception 3.10.4.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored when operating with a nonsymmetric control rod pattern. Only the center of the core LPRM string detectors A and C and two other LPRM string detectors A and C need be monitored for operations with a symmetric control rod pattern.

Insert F

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

a) Place the individual recirculation pump flow controller for the operating recirculation pump in the Manual mode.

b) Reduce THERMAL POWER to less than or equal to 70% of RATED THERMAL POWER.

c) Limit the speed of the operating recirculation pump to less than or equal to 75% of rated pump speed.

d) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2.

e) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit per Specification 3.2.1.

f) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation* per Specifications 2.2.1, 3.2.2, and 3.3.6.

g) Perform Surveillance Requirement 4.4.1.1.4 if THERMAL POWER is less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated loop flow.

2. The provisions of Specification 3.0.4 are not applicable.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, while in OPERATIONAL CONDITION 1, immediately place the Reactor Mode Switch in the SHUTDOWN position.

c. With no reactor coolant system recirculation loops in operation, while in OPERATIONAL CONDITION 2, initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours.

---

* APRM gain adjustments may be made in lieu of adjusting the APRM and RBM Flow Biased Setpoints to comply with the single loop values for a period of up to 72 hours.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Neutron Flux-High setpoint, a time constant of 6 ± 3 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP. *Add Insert.*

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

INSERT PAGE B 2-7

For single recirculation loop operation, the reduced APRM setpoints are based on a ΔW value of 8%. The ΔW value corrects for the difference in indicated drive flow (in percentage of drive flow which produces rated core flow) between two loop and single loop operation of the same core flow. The decrease in setpoint is derived by multiplying the slope of the setpoint curve by 8%. The High Flow Clamped Flow Biased Neutron Flux-High setpoint is not applicable to single loop operation as core power levels which would require this limit are not achievable in a single loop configuration.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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 dependent only 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 corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is ~~[illegible]~~ provided in the CORE OPERATING LIMITS REPORT (COLR).
The ~~Technical Specification~~ COLR MAPLHGR value is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the ECCS MAPLHGR.

Fuel Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice, operating at the bounding power history, meet the fuel design limits specified in Reference 1. This bounding power history is used as the basis for the fuel design analysis MAPLHGR value.

LOCA Analysis: A LOCA analysis is performed in accordance with 10 CFR 50 Appendix K to demonstrate that the MAPLHGR values comply with the ECCS limits specified in 10 CFR 50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

Only the most limiting MAPLHGR values ~~[illegible]~~ are provided in the ~~[illegible]~~ COLR ~~[illegible]~~ for multiple lattice fuel. When hand calculations are required, these ~~[illegible]~~ COLR MAPLHGR ~~[illegible]~~ values for that fuel type are used for all lattices in that bundle.

For some fuel bundle designs MAPLHGR depends only on bundle type and burnup. Other fuel bundles have MAPLHGRs that vary axially depending upon the specific combination of enriched uranium and gadolinia that comprises a fuel bundle cross section at a particular axial node. Each particular combination of enriched uranium and gadolinia, for these fuel bundle types, is called a lattice type. These particular fuel bundle types have MAPLHGRs that vary by lattice (axially) as well as with fuel burnup.

Add Insert p. B 3/4 2-1

Reference

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A ~~(latest approved revision)~~

Insert p. B 3/4 2-1

For plant operation with a single recirculation loop, the above MAPLHGR limits are multiplied by a factor specified in the CORE OPERATING LIMITS REPORT (COLR). The COLR factor is derived from LOCA analysis initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA analysis.

BASES

3/4.4.1 RECIRCULATION SYSTEM

~~Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.~~

Insert G

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. Insert H

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. ~~Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.~~ Insert I

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 11 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that a potentially high thrust load (designated as load case C.3.3) on the SRV discharge lines is eliminated during subsequent actuations. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

Insert G:

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted at power levels up to 70% of RATED THERMAL POWER if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2. APRM scram and control rod block setpoints (or APRM gains) are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively. MAPLHGR limits are decreased by the factor given in Specification 3.2.1. A time period of 4 hours is allowed to make these adjustments following the establishment of single loop operation since the need for single loop operation often cannot be anticipated. MCPR operating limits adjustments in Specification 3.2.3 for different plant operating situations are applicable to both single and two recirculation loop operation.

To prevent potential control system oscillations from occurring in the recirculation flow control system, the operating mode of the recirculation flow control system must be restricted to the manual control mode for single-loop operation.

Additionally, surveillance on the pump speed of operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 30% THERMAL POWER or 50% rated recirculation loop flow is to prevent undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during a power or flow increase following extended operation in the single recirculation loop mode.

Insert H:

The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

Insert I:

Sudden equalization of a temperature difference >145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

Requirements are imposed to prohibit idle loop startup above the 80% rod line to minimize the potential for initiating core thermal-hydraulic instability.

3/4.4.10 CORE THERMAL HYDRAULIC STABILITY

BWR cores typically operate with the presence of global neutron flux noise in a stable mode which is due to random boiling and flow noise. As the power/flow conditions are changed, along with other system parameters (pressure, subcooling, power distribution, etc.) the thermal hydraulic / reactor kinetic feedback mechanism can be enhanced such that random perturbations may result in sustained limit cycle or divergent oscillations in power and flow.

Two major modes of oscillations have been observed in BWRs. The first mode is the fundamental or core-wide oscillation mode in which the entire core oscillates in phase in a given axial plane. The second mode involves regional oscillation in which one half of the core oscillates 180 degrees out of phase with the other half. Studies have indicated that adequate margin to the Safety Limit Minimum Critical Power Ratio (SLMCPR) may not exist during regional oscillations.

Regions A and B of Figure 3.4.10-1 represent the least stable conditions of the plant (high power/low flow). Region A and B are usually entered as the result of a plant transient (for example, recirculation pump trips) and therefore are generally not considered part of the normal operating domain. Since all stability events (including test experience) have occurred in either Region A or B, these regions are avoided to minimize the possibility of encountering oscillations and potentially challenging the SLMCPR. Therefore, intentional operation in Regions A or B is not allowed. It is recognized that during certain abnormal conditions within the plant, it may become necessary to enter Region A or B for the purpose of protecting equipment which, were it to fail, could impact plant safety or for the purpose of protecting a safety or fuel operating limit. In these cases, the appropriate actions for the region entered would be performed as required.

Most oscillations that have occurred during testing and operation have occurred at or above the 100% rod line with core flow near natural circulation. This behavior is consistent with analysis which predict reduced stability margin with increasing power or decreasing flow. As core flow is increased or power decreased, the probability of oscillations occurring will decrease. Region A of Figure 3.4.10-1 bounds the majority of the stability events and tests observed in GE BWRs. Since Region A represents the least stable region of the power/flow operating domain, the potential to rapidly encounter large magnitude core thermal hydraulic

oscillations is increased. During transients, the operator may not have sufficient time to manually insert control rods to mitigate the oscillations before they reach an unacceptable magnitude. Therefore, the prompt action of manually scramming the plant when Region A is entered is required to ensure protection of the SLMCPR.

Based on test and operating experience, the frequency of core thermal hydraulic oscillations is less in Region B than in Region A. Decay ratios are expected and predicted to be lower in this region since Region B covers a lower power and higher flow range than Region A. Also, the margin to the SLMCPR will typically be larger in Region B than in Region A. With more margin to SLMCPR and a lower probability of oscillations, exiting Region B by control rod insertion is justified. However, if oscillations are observed while exiting Region B, the reactor will be manually scrammed.

The potential for core thermal hydraulic oscillations to occur outside of Regions A and B is very small and therefore special requirements are not necessary outside of these regions.

ENCLOSURE 1

FERMI 2 CYCLE 2

CORE OPERATING LIMITS REPORT

REVISION 0