REVISED TECHNICAL SPECIFICATION PAGES FOR LASALLE UNITS 1 AND 2

LASALLE UNIT 1	LASALLE UNIT 2
B 2-2	B 2-2
Insert 2	Insert 2
5-4	5-4
Insert 10	Insert 10
6-25	6-25
Insert 9	Insert 9
B 3/4 1-4	B 3/4 1-4
B 3/4 2-6	B 3/4 2-6
Insert 6	Insert 6
Insert 7	Insert 7

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2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), GEXL correlation.

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A[a]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

a. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

b. General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Admendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

Insert #2 here

Insert #2

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the SPC-developed ANFB critical power correlation.

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-5. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524 (P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

3. ANFB Critical Power Correlation, ANF-1125 (P) (A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19 (P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

5.3 REACTOR CORE

Replace with Insert #10

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly consists of a matrix of Zircalloy clad fuel rods with an initial composition of slightly enriched uranium dioxide, UO_2. Fuel assemblies shall be limited to those fuel designs approved for use in BWR's.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide power (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

Insert #10

4.0 DESIGN FEATURES

4.1 Site Location [Text location of site location]

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain ~~[800]~~ 764 fuel assemblies. Each assembly shall consist of a matrix of ~~[Zircalloy or ZIRLO]~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material~~[, and water rods]~~. Limited substitutions of ~~zirconium alloy~~ or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

Zircalloy or ZIRLO

The bundles may contain water rods or water boxes.

4.2.2 Control Rod Assemblies

The reactor core shall contain [193] cruciform shaped control rod assemblies. The control material shall be [boron carbide, hafnium metal] as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum [k-infinity of [1.31] in the normal reactor core configuration at cold conditions] [average U-235 enrichment of [4.5] weight percent];

b. $k_{eff} \le 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];

(continued)

Semiannual Radioactive Effluent Release Report (Continued)

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.

6. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

Effects of analyzed equipment out of service are included

(1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

(2) The minimum Critical scram time Power Ratio (MCPR) ~~(including 20% scram time, tau (τ),~~ e dependent MCPR limits, and power and flow dependent MCPR limits~~)~~ for Technical Specification 3.2.3.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

(4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, ~~in the latest approved revision or supplement of the topical reports describing the methodology~~. For LaSalle County Station Unit 1, the topical reports are:

Insert #9

~~(1)~~ (18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

~~(2)~~ (19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

~~(3)~~ (20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

~~(4)~~ (21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

(22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," ~~(latest approved revision).~~ Revision 0, supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

Insert #9

1. ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

2. Letter, Ashok C. Thadani (NRC) to R. A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

3. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondance, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

4. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

5. HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A) Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuel Corporation, August 1986 and January 1991, respectively.

6. Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

7. Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.

8. Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

9. Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

10. Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.

11. Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

12. RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

13. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

14. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

15. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

16. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

17. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

3/4.1.3 CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3.65 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.~~4~~.9 (P)(A) of the FSAR and the techniques of the analysis are presented in ~~a topical report, Reference 1, and two supplements, References 2 and 3.~~ XN-NF-80-19, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Volume 1 and Supplements 1 and 2, March 1983.

3/4.2.4 LINEAR HEAT GENERATION RATE

GE fuel

The 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. ~~The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a 95% confidence that no more than one fuel rod exceeds the design LINEAR HEAT GENERATION RATE due to power spiking.~~

Insert #6 here

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDC-20566A, September 1986.

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Company Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Vol. III as supplemented by letter dated September 5, 1980, from R. H. Buchholz (GE) to P. S. Check (NRC).

3. "LaSalle County Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," General Electric Company Report NEDC-32258P, October 1993.

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

5. "Extended Operating Domain and Equipment Out-of-Service for LaSalle County Nuclear Station Units 1 and 2," NEDC-31455, November 1987.

6. "ARTS Improvement Program Analysis for LaSalle County Units 1 and 2," General Electric Company Report NEDC-31531P, December 1993.

Insert #7 here

The effects of fuel densification are discussed in the General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A. The GESTAR discusses the methods used to ensure LHGR remains below the design limit.

Insert #6

SPC Fuel

The Linear Heat Generation Rate (LHGR) is a measure of the heat generation rate per unit length of a fuel rod in a fuel assembly at any axial location. LHGR limits are specified to ensure that fuel integrity limits are not exceeded during normal operation or anticipated operational occurrences (AOOs). Operation above the LHGR limit followed by the occurrence of an AOO could potentially result in fuel damage and subsequent release of radioactive material. Sustained operation in excess of the LHGR limit could also result in exceeding the fuel design limits. The failure mechanism prevented by the LHGR limit that could cause fuel damage during AOOs is rupture of the fuel rod cladding caused by strain from the expansion of the fuel pellet. One percent plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur. Fuel design evaluations are performed to demonstrate that the mechanical design limits are not exceeded during continuous operation with LHGRs up to the limit defined in the CORE OPERATING LIMITS REPORT. The analysis also includes allowances for short term transient operation above the LHGR limit.

At reduced power and flow conditions, the LHGR limit may need to be reduced to ensure adherence to the fuel mechanical design bases during limiting transients. At reduced power and flow conditions, the LHGR limit is reduced (multiplied) using the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power. The $LHGRFAC_f$ multipliers are used to protect the core during slow flow runout transients. The $LHGRFAC_p$ multipliers are used to protect the core during plant transients other than core flow transients. The applicable $LHGRFAC_f$ and $LHGRFAC_p$ multipliers are specified in the CORE OPERATING LIMITS REPORT.

Insert #7

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19 (P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19 (P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.

4. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A) Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

6. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, Exxon Nuclear Company, February 1987.

7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1, and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

8. LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis, NEDC-32258P, General Electric Company, October 1993.

9. ARTS Improvement Program analysis for LaSalle County Station Units 1 and 2, NEDC-31531P, General Electric Company, December 1993.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

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e

Insert #2 here

e

a. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and Design Application," NEDO-10958-A.

Insert #2

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the SPC-developed ANFB critical power correlation.

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1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524 (P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

3. ANFB Critical Power Correlation. ANF-1125 (P) (A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19 (P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

6. "Application of the ANFB Critical Power Correlation to Coresident GE Fuel for LaSalle Unit 2 Cycle 8," EMF-96-021 (P), Revision 1, Siemens Power Corporation, February 1996; NRC SER letter dated September 26, 1996.

5.3 REACTOR CORE

Replace with Insert #10

FUEL ASSEMBLIES

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b. For a pressure of:

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2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

Insert #10

4.0 DESIGN FEATURES

4.1 Site Location [Text location of site location]

~~4.2 Reactor Core~~

4.2.1 Fuel Assemblies

764

The reactor shall contain ~~[800]~~ fuel assemblies. Each assembly shall consist of a matrix of ~~[~~Zircalloy ~~or ZIRLO]~~ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material~~[, and water rods]~~. Limited substitutions of ~~zirconium alloy~~ or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

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b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR];

(continued)

(1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

Effects of analyzed equipment out of service are included.

scram time

(2) The minimum Critical Power Ratio (MCPR) ~~(including 20% scram time, tau (τ),~~ dependent MCPR limits, and power and flow dependent MCPR limits~~)~~ for Technical Specification 3.2.3.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

(4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, ~~in the latest approved revision or supplement of the topical reports describing the methodology.~~ For LaSalle County Station Unit 2, the topical reports are:

Insert #9

~~(1)~~ (18) NEDE-24011-~~P-A~~, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

~~(2)~~ (19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

~~(3)~~ (20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

~~(4)~~ (21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS Limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the U.S. Nuclear Regulatory Commission Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Deleted.

(22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," ~~(latest approved revision)~~

Revision 0, supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

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1. ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

2. Letter, Ashok C. Thadani (NRC) to R. A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

3. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondance, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

4. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

5. HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A) Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuel Corporation, August 1986 and January 1991, respectively.

6. Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

7. Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.

8. Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

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14. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

15. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

16. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

17. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

3/4.1.3 CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3.65 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

(PXA)

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented ~~in a topical report, Reference 1, and two supplements, References 2 and 3.~~ XN-NF-80-19, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Volume 1 and Supplements 1 and 2, March 1983

3/4.2.4 LINEAR HEAT GENERATION RATE

GE fuel

The 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. ~~The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a 95% confidence that no more than one fuel rod exceeds the design LINEAR HEAT GENERATION RATE due to power spiking.~~

References:

Insert #6 here

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566A, September 1986.

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Co. Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Vol. III as supplemented by letter dated September 5, 1980, from R. H. Buchholz (GE) to P. S. Check (NRC).

3. "LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," General Electric Co. Report NEDC-32258P, October 1993.

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

5. "Extended Operating Domain and Equipment Out-of-Service for LaSalle County Nuclear Station Units 1 and 2," NEDC-31455, November 1987.

6. "ARTS Improvement Program Analysis for LaSalle County Station Units 1 and 2," General Electric Co. Report NEDC-31531P, December 1993.

Insert #7 here

The effects of fuel densification are discussed in the General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A. The GESTAR discusses the methods used to ensure LHGR remains below the design limit.

Insert #6

SPC Fuel

The Linear Heat Generation Rate (LHGR) is a measure of the heat generation rate per unit length of a fuel rod in a fuel assembly at any axial location. LHGR limits are specified to ensure that fuel integrity limits are not exceeded during normal operation or anticipated operational occurrences (AOOs). Operation above the LHGR limit followed by the occurrence of an AOO could potentially result in fuel damage and subsequent release of radioactive material. Sustained operation in excess of the LHGR limit could also result in exceeding the fuel design limits. The failure mechanism prevented by the LHGR limit that could cause fuel damage during AOOs is rupture of the fuel rod cladding caused by strain from the expansion of the fuel pellet. One percent plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur. Fuel design evaluations are performed to demonstrate that the mechanical design limits are not exceeded during continuous operation with LHGRs up to the limit defined in the CORE OPERATING LIMITS REPORT. The analysis also includes allowances for short term transient operation above the LHGR limit.

At reduced power and flow conditions, the LHGR limit may need to be reduced to ensure adherence to the fuel mechanical design bases during limiting transients. At reduced power and flow conditions, the LHGR limit is reduced (multiplied) using the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power. The $LHGRFAC_f$ multipliers are used to protect the core during slow flow runout transients. The $LHGRFAC_p$ multipliers are used to protect the core during plant transients other than core flow transients. The applicable $LHGRFAC_f$ and $LHGRFAC_p$ multipliers are specified in the CORE OPERATING LIMITS REPORT.

Insert #7

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19 (P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19 (P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.

4. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A) Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

6. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, Exxon Nuclear Company, February 1987.

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8. LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis, NEDC-32258P, General Electric Company, October 1993.

9. ARTS Improvement Program analysis for LaSalle County Station Units 1 and 2, NEDC-31531P, General Electric Company, December 1993.