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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

Delete

AVERAGE PLANAR EXPOSURE (APE)

The AVERAGE PLANAR EXPOSURE (APE) shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATE(s) for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL

A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of CHANNEL behavior during operation by observation. This determination shall include, where possible, comparison of the CHANNEL indication and/or status with other indications and/or status derived from independent instrument CHANNEL(s) measuring the same parameter.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

1.09

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

2.1 SAFETY LIMITS

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THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 for Unit 1 and 1.10* for Unit 2 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

* Applicable to Unit 2 for cycle 15 only. DELETE

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approach. Much of the data indicates that BWR fuel can survive for an extended period in an environment of transition boiling.

The Unit 1 MCPR Safety Limit is 1.07, based on General Electric methods for calculating the MCPR Safety Limit. The Unit 2 MCPR Safety Limit is 1.10*, based on Siemens Power Corporation (SPC) methods for calculating the MCPR Safety Limit.

DELETE

2.1.C Reactor Coolant System Pressure

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The reactor coolant system pressure Safety Limit of 1345 psig, as measured by the vessel steam space pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure Safety Limit was chosen as the lower of the pressure transients permitted by the applicable design codes, ASME Boiler and Pressure Vessel Code Section III for the pressure vessel, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over design pressure (120% x 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirculation suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology to assure that this Safety Limit pressure is not exceeded for any reload is documented by the specific fuel vendor. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provides similar margin of protection at the established pressure Safety Limit.

The normal operating pressure of the reactor coolant system is nominally 1000 psig. Both pressure relief and safety relief valves have been installed to keep the reactor vessel peak pressure below 1375 psig. However no credit is taken for relief valves during the postulated full closure of all MSIVs without a direct (valve position switch) scram. Credit, however, is taken for the neutron flux scram. The indirect flux scram and safety valve actuation provide adequate margin below the allowable peak vessel pressure of 1375 psig.

* Applicable to Unit 2 cycle 15 only. DELETE

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

delete

APPLICABILITY:

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and

2. Restore APLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

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

(15) 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, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

(16) COTRANSA 2: 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.

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

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

Delete

(19)* ComEd letter, "ComEd Response to NRC Staff Request for Additional Information (RAI) Regarding the Application of Siemens Power Corporation ANFB Critical Power Correlation to Coresident General Electric Fuel for LaSalle Unit 2 Cycle 8 and Quad Cities Unit 2 Cycle 15, NRC Docket No.'s 50-373/374 and 50-254/265", J.B. Hosmer to U.S. NRC, July 2, 1996, transmitting the topical report, Application of the ANFB Critical Power Correlation to Coresident GE Fuel for Quad Cities Unit 2 Cycle 15, EMF-96-051(P), Siemens Power Corporation - Nuclear Division, May 1996, and related information.

Insert A →

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

6.9.B Special Reports

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

Delete

*Applicable to Unit 2 for cycle 15 only.

INSERT A

QUAD CITIES Section 6.9.A.6.b Technical Specifications Insert

(19) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(20) ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

Attachment D

Marked Up Pages and Inserts for Dresden Technical Specifications

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

Delete

AVERAGE PLANAR EXPOSURE (APE)

The AVERAGE PLANAR EXPOSURE (APE) shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATE(s) for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL

A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of CHANNEL behavior during operation by observation. This determination shall include, where possible, comparison of the CHANNEL indication and/or status with other indications and/or status derived from independent instrument CHANNEL(s) measuring the same parameter.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

1.09

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

Delete

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of bundle average exposure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective action within 15 minutes, and

2. Restore APLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,

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

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

4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. STEADY STATE LINEAR HEAT GENERATION RATE

DELETE

The LINEAR HEAT GENERATION RATE (LHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With an LHGR exceeding the SLHGR limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the LHGR to within the SLHGR limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. STEADY STATE LINEAR HEAT GENERATION RATE

The SLHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for SLHGR.
4. The provisions of Specification 4.0.D are not applicable.

5.0 DESIGN FEATURES

5.3 REACTOR CORE

Fuel Assemblies

Delete

5.3.A The reactor core shall contain 724 fuel assemblies[1,2]. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. The assemblies may contain water rods or a water box. Limited substitutions of Zircaloy or ZIRLO or stainless steel filler rods for fuel rods, in accordance with NRC-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 fuel safety design bases[3]. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Delete

Control Rod Assemblies

5.3.B The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

Delete

1 ATRIUM-9B fuel with exception of lead test assemblies is only allowed in the reactor core in Operational Modes 3, 4 and 5, and with no more than one control rod withdrawn, for Unit 2 only.

2 Operation in all modes with ATRIUM-9B fuel is allowed for Dresden, Unit 3, Cycle 15, only.

3 The design bases applicable to ATRIUM-9B fuel are those which are applicable to Operational Modes 3, 4, and 5, for Unit 2 only.

b. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of topical reports:

(1) ANF-1125(P)(A), "Critical Power Correlation - ANFB."

(2) ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

(3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

(4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

(5) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel."

(6) ANF-913(P)(A), "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis."

(7) XN-NF-82-06(P)(A), Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 BWR Fuel, Supplement 1, Revision 2, Advanced Nuclear Fuels Corporation, May 1988.

(8) ANF-89-14(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, Revision 1 and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, October 1991.

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

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

(11) Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods", and associated Supplements on Neutronics Licensing Analyses (Supplement 1) and La Salle County Unit 2 Benchmarking (Supplement 2).

Insert B

INSERT B

DRESDEN Section 6.9.A.6.b Technical Specification Insert

(12) ANF-1125(P)(A), ANFB Critical Power Correlation Uncertainty For Limited Data Sets, Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

Attachment E

Marked Up Pages and Inserts for LaSalle Unit 1 Technical Specifications

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DEFINITIONS

SECTION

1.0 DEFINITIONS

Delete

1.0 DEFINITIONS

DELETED

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE DELETE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. Delete

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

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 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.

7 Replace

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.

Insert C

INSERT C

LASALLE UNIT 1 Bases Section 2.1.2 Technical Specifications Insert

6. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

7. ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

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) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

Delete

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, 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 specified in the CORE OPERATING LIMITS REPORT.

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.

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 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.

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 and BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1, Siemens Power Corporation, DATE TO BE DETERMINED.

References:

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.

Core Operating Limits Report (Continued)

(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.

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

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

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

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

Insert D

INSERT D

LASALLE UNIT 1 Section 6.6.A.6.b Technical Specifications Insert

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1, Siemens Power Corporation, (DATE TO BE DETERMINED).

(24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(25) ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

Attachment F

Marked up Pages and Inserts for LaSalle Unit 2 Technical Specifications

DELETED

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS PAGE

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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE DELETE

Delete

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

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 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.

7 Replace

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, 3-6. 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.

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.

Delete

Insert E

INSERT E

LASALLE UNIT 2 Bases Section 2.1.2 Technical Specifications Insert

6. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

7. ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

Delete

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, 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 specified in the CORE OPERATING LIMITS REPORT.

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.

3/4.2.4 LINEAR HEAT GENERATION RATE (Continued)

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.

References:

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.

INSERT

and BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1, Siemens Power Corporation, (DATE TO BE DETERMINED).

Core Operating Limits Report (Continued)

(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.

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

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

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

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

Insert F

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LASALLE UNIT 2 Section 6.6.A.6.b Technical Specifications Insert

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1, Siemens Power Corporation, (DATE TO BE DETERMINED).

(24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(25) ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation, (DATE TO BE DETERMINED).

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

G. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Adding References 1 and 7 to Technical Specification Section 6 and applying these methods at ComEd BWRs is evaluated for significant hazards consideration in this section. These documents have been submitted to the NRC under separate correspondence. References 1 and 7 are in NRC review, and require approval to be inserted into Section 6.

ComEd has evaluated the proposed Technical Specification amendment and determined it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazard consideration established in 10CFR50.92(c), operation of Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2, in accordance with the proposed amendments, will not represent a significant hazards consideration for the following reasons:

These changes do not:

1. **Involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits.

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The Reference 1 methodology to be added to the Technical Specifications is used as part of the LOCA analysis and does not introduce physical changes to the plant. The Reference 1 revised jet pump model changes the calculational behavior of the jet pump under reversed drive flow conditions. The revised jet pump model methodology makes the LOCA model behave more realistically and calculates small break LOCA PCTs that are comparable to the large break LOCA results. Therefore, this change only affects the methodology for analyzing the LOCA event and determining the protective APLHGR limits. The Technical Specification requirements for monitoring APLHGR are not affected by this change. The revised method will result in higher APLHGR limits, thus the SPC fuel will be allowed to operate at higher nodal powers. The approved methodology, however, still protects the fuel performance limits specified by 10CFR50.46. Therefore, the probability or consequences of an accident previously evaluated will not change.

ATTACHMENT G

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident are not increased by adding Reference 3 to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Reference 3 determines the additive constants and the associated uncertainty for application of the ANFB correlation to the coresident GE fuel. Therefore, it provides data that is used in the determination of the MCPR Safety Limit. This approved methodology for applying the ANFB critical power correlation to the GE fuel will protect the fuel from boiling transition. Operational MCPR limits will also be applied to ensure that the MCPR Safety Limit is protected during all modes of operation and anticipated operational occurrences. Because Reference 3 contains conservative methods and calculations and because the operability of plant systems designed to mitigate any consequences of accidents have not changed, the probability or consequences of an accident previously evaluated will not increase.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The probability or consequences of a previously evaluated accident is not increased by adding Reference 7 to Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. **Reference 7 documents the additive constant uncertainty for SPC ATRIUM-9B fuel design with an internal water channel. This methodology is used to** determine an input to the MCPR Safety Limit calculations, which ensures that more than 99.9% of the fuel rods avoid transition boiling during normal operation as well as anticipated operational occurrences. This change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. This methodology for determining the ATRIUM-9B additive constant uncertainty for the MCPR Safety Limit calculation will continue to support protecting the fuel from boiling transition. Operational MCPR limits will be applied to ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. Therefore, no individual precursors of an accident are affected and the operability of plant systems designed to mitigate the probability of consequences of an accident previously evaluated are not affected by these changes.

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

Changing the MCPR Safety Limit at Quad Cities Units 1 and 2 and Dresden Units 2 and 3 will not increase the probability of an accident previously evaluated. This change implements the MCPR Safety Limits resulting from the SPC ANFB critical power correlation methodology using a revised additive constant uncertainty from Reference 7. The MCPR Safety Limit of 1.09 that is proposed for Quad Cities Units 1 and 2 and Dresden Units 2 and 3 is anticipated to be conservative and acceptable for future cycles. Cycle specific MCPR Safety Limit calculations will be performed, consistent with SPC's approved methodology, to confirm the appropriateness of the MCPR Safety Limit. Additionally, operational MCPR limits will be applied that will ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. Changing the MCPR Safety Limit will not alter any physical systems or operating procedures. The MCPR Safety Limit is set to 1.09, which is the CPR value where less than 0.1% of the rods in the core are expected to experience boiling transition. This safety limit is expected to be applicable for future cycles of ATRIUM-9B at Dresden and Quad Cities. Therefore the probability or consequences of an accident will not increase.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The removal of footnotes from the Quad Cities and Dresden Technical Specifications does not involve any significant increase in the probability or consequences of an accident previously evaluated. The footnotes were added to clarify that cycle specific methods were used until the generic methodology was approved by the NRC. Since the NRC has approved SPC's generic methodology for application of the ANFB correlation to the coresident GE fuel (Reference 3) and SPC has addressed the concerns regarding the database used to calculate the ATRIUM-9B additive constant uncertainties (Reference 7), the footnotes are no longer necessary. The removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, removing these footnotes and "a" pages does not require any physical plant modifications, nor does it physically affect any plant components or entail changes in plant operation. Therefore, the probability or consequences of an accident previously evaluated is not expected to increase.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the Section 3 Technical Specification description of the APLHGR limits has no implications on accident analysis or plant operations. The purpose of the revision is to allow flexibility for the MAPLHGR limits and their exposure basis to be specified in the COLR and to establish consistency with approved methodologies currently utilized by Siemens Power Corporation, which calculates MAPLHGR limits based on bundle or planar average exposures. This revision also provides for consistency in the APLHGR limit Technical Specification wording between the ComEd BWRs. The revision to the 3.11.D SLHGR Technical Specification for Dresden also has no implications on accident analysis or plant operations. The purpose of this revision is to allow flexibility for the LHGR limits and their exposure basis to be specified in the COLR. This revision makes the Dresden LHGR definition consistent with NUREG 1433/1434 wording. The definition of the Average Planar Exposure is deleted, because the exposure basis of the APLHGR is being removed. Therefore, no plant equipment or processes are affected by this change. Thus, there is no alteration in the probability or consequences of an accident previously evaluated.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications to the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications to the plant configuration or allowable modes of operation. No new precursors of an accident are created and no new or different kinds of accidents are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology will be used to analyze the LOCA for LaSalle Units 1 and 2, and does not introduce any physical changes to the plant or the processes used to operate the plant. This change only affects the methods used to analyze the LOCA event and determine the MAPLHGR limits. Therefore, the possibility of a new or different kind of accident is not created.

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

Addition of the generic methodology for the application of the ANFB critical power correlation to GE fuel in Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications does not introduce any physical changes to the plant, the processes used to operate the plant, or allowable modes of operation. This change only involves adding an NRC approved methodology, which is used to determine the additive constants and additive constant uncertainty for GE fuel, to Section 6 of the Technical Specifications. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

Addition of the Reference 7 methodology to Section 6.9.A.6.b of the Quad Cities and Dresden Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications will not create the possibility of a new or different kind of accident from any accident previously evaluated. This methodology describes the calculation of an input to the MCPR Safety Limit - the ATRIUM-9B additive constant uncertainty. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Units 1 and 2 and Dresden Units 2 and 3)

Changing the MCPR Safety Limit will not create the possibility of a new accident from an accident previously evaluated. This change will not alter or add any new equipment or change modes of operation. The MCPR Safety Limit is established to ensure that 99.9% of the rods avoid boiling transition.

The MCPR Safety Limit is changing for Quad Cities Unit 1 due to the transition to SPC ATRIUM-9B fuel and SPC methodologies. The MCPR Safety Limit is changing for Quad Cities Unit 2 due to the Reference 7 methodology, which documents a 0.0195 ATRIUM-9B additive constant uncertainty and supports a 1.09 MCPR Safety Limit. This MCPR Safety Limit is lower than the current MCPR Safety Limit for Quad Cities Unit 2, 1.10, which is based on a higher interim conservative additive constant uncertainty of 0.029. The lower ATRIUM-9B additive constant uncertainty results in the lower MCPR Safety Limit for Quad Cities Unit 2. The new MCPR Safety Limit for Dresden Units 2 and 3, 1.09, is greater than the current value at Dresden Units 2 and 3 and is being increased now in anticipation of bounding future reloads of ATRIUM-9B. Therefore, no new accidents are created that are different from any accident previously evaluated.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The removal of the footnotes from the Quad Cities and Dresden Technical Specifications does not create a new or different kind of accident from any accident previously evaluated. The removal of the footnotes does not affect plant systems or operation. The footnotes were temporarily established to implement a conservative cycle specific MCPR Safety Limit until the SPC generic methodology was approved. With the approval of the generic Reference 3 methodology and the anticipated approval of the Reference 7 additive constant uncertainty methodology, these footnotes are no longer applicable. The removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications which is justified by the removal of the footnotes, also does not create a new or different kind of accident from any accident previously evaluated.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle 1 and 2)

The revision of the APLHGR and LHGR limit descriptions will not create the possibility of a new or different kind of accident from any accident previously evaluated. This revision will not alter any plant systems, equipment, or physical conditions of the site. This revision allows the flexibility of the APLHGR and the LHGR limits to be specified in the COLR and to maintain consistency with the calculated results of methodologies currently used to determine the APLHGR. The definition of the Average Planar Exposure is deleted, because it is being removed from LHGR and APLHGR Technical Specifications.

3. Involve a significant reduction in the margin of safety for the following reasons:

Addition of SPC Revised Jet Pump Methodology (LaSalle Units 1 and 2)

The revised jet pump model methodology, and the MAPLHGRs, resulting from the revised jet pump methodology, will continue to ensure fuel design criteria and 10CFR50.46 compliance. The results of LOCA analyses performed with this methodology must continue to comply with the requirements of 10CFR50.46. Therefore, there is no significant reduction in the margin of safety.

Addition of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel (Quad Cities Units 1 and 2 and LaSalle Units 1 and 2)

The margin of safety is not decreased by adding this reference to Section 6.9.A.6.b of the Quad Cities Technical Specifications and Bases Section 2.1.2 and Section 6.6.A.6.b of the LaSalle Technical Specifications. Siemens Power Corporation methodology for application of the ANFB Critical Power Correlation to coresident GE fuel is approved by the NRC and is the same methodology used in the cycle specific topical for coresident fuel (Reference 4 and 5). The MCPR Safety Limit will continue to ensure that greater than 99.9% of the rods in the core avoid boiling transition. Additionally, operating limits will be established to ensure the MCPR Safety Limit is not violated during all modes of operation.

Addition of SPC Topical for Revised ANFB Correlation Uncertainty (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. This Technical Specification amendment proposes to insert the topical report that describes SPC's calculation of the ATRIUM-9B additive constant uncertainty. The new ATRIUM-9B additive constant uncertainty calculation is conservative and is based on a larger database than previous calculations. Because a conservative method is used to calculate the ATRIUM-9B additive constant uncertainty, a decrease in the margin to safety will not occur due to adding this methodology to the Technical Specifications. In addition, operational limits will be established to ensure the MCPR Safety Limit is protected for all modes of operation. This revised methodology will only ensure that the appropriate level of fuel protection is being employed.

Change to Minimum Critical Power Ratio Safety Limit (Quad Cities Unit 1 and 2 and Dresden Units 2 and 3)

Changing the MCPR Safety Limit for Quad Cities and Dresden will not involve any reduction in margin of safety. The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR Safety Limit is not violated. The proposed Technical Specification amendment reflects the MCPR Safety Limit results from conservative evaluations by SPC using the

ANFB critical power correlation with the new 0.0195 ATRIUM-9B additive constant uncertainty documented in Reference 7 .

Because a conservative method is used to apply the ATRIUM-9B additive constant uncertainty in the MCPR Safety Limit calculation, a decrease in the margin to safety will not occur due to changing the MCPR Safety Limit. The revised MCPR Safety Limit will ensure the appropriate level of fuel protection. Additionally, operational limits will be established based on the proposed MCPR Safety Limit to ensure that the MCPR Safety Limit is not violated during all modes of operation including anticipated operation occurrences. This will ensure that the fuel design safety criterion of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as during an anticipated operational occurrence is met.

Removal of Footnotes Limiting Operation with ATRIUM-9B Fuel Reloads (Quad Cities Unit 2 and Dresden Unit 3)

The removal of the cycle specific footnotes in Quad Cities and Dresden Technical Specifications does not impose a change in the margin of safety. These footnotes were added due to concerns regarding the calculation of the additive constant uncertainty for the ATRIUM-9B fuel and the cycle specific application of the ANFB critical power correlation to coresident GE fuel in Quad Cities Unit 2 Cycle 15. Because the generic ANFB application to coresident GE fuel MCPR methodology (Reference 3) has received NRC approval and the topical report describing the increased database used to calculate the additive constant uncertainties for ATRIUM-9B (Reference 7) have been submitted to the NRC and both are proposed to be added to the Technical Specifications in this amendment, there is no reason for the footnotes to remain. Removal of the Unit 2 specific "a" pages, 2-1a and B2-3a, in the Quad Cities Technical Specifications is justified by the removal of the footnotes. Therefore, the removal of the "a" pages, 2-1a and B2-3a, also does not impose a change in the margin of safety.

Revision to Thermal Limit Descriptions (Quad Cities Units 1 and 2, Dresden Units 2 and 3, and LaSalle Units 1 and 2)

The revision to the APLHGR and LHGR limit descriptions will not involve a reduction in the margin of safety. The methodology used to calculate the APLHGR must comply with the guidelines of Appendix K of 10 CFR Part 50, and the APLHGR and LHGR will still be required to be maintained within the limits specified in the COLR. The surveillance requirements for these two thermal limits remain unchanged. Thus, there will be no reduction in the margin of safety.

ATTACHMENT G

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

This proposed amendment does not involve a significant relaxation of the criteria used to establish the safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in 10CFR50.92(c), the proposed change does not constitute a significant hazards consideration.

Attachment H

Environmental Assessment Applicability Review

ENVIRONMENTAL ASSESSMENT APPLICABILITY REVIEW

H. ENVIRONMENTAL ASSESSMENT APPLICABILITY REVIEW

ComEd has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). This conclusion has been determined because the changes requested do not pose significant hazards considerations and do not involve a significant increase in the amounts, and no significant changes in the types of any effluents that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

Attachment I

References

ATTACHMENT I

REFERENCES

I. REFERENCES

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REFERENCES

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12. "LaSalle County Nuclear Power Station Units 1 and 2 Application for Amendment Request to Facility Operating Licenses NPF-11 and NPF-18, Technical Specifications Changes for Siemens Power Corporation Fuel Transition Docket Numbers 050-373 and 050-374", R.E. Querio to US NRC, April 8, 1996.

13. "LaSalle County Nuclear Power Station Units 1 and 2 Supplement to Application for Amendment of Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specification Changes for Siemens Power Corporation Fuel Transition", W.T. Subalusky to U.S. NRC, October 14, 1996.

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