DEFINITIONS

SECTION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.8 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.A.6. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

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

Ē-AVERAGE DISINTEGRATION ENERGY

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

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to energization of the recirculation pump circuit

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME (Continued)

breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY Deleted

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

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

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

1.20 DELETED

LIMITING CONTROL ROD PATTERN

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

LINEAR HEAT GENERATION RATE

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

LOGIC SYSTEM FUNCTIONAL TEST

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

MAXIMUM FRACTION OF LIMITING POWER DENSITY

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

DELETED

MEMBERS(S) OF THE PUBLIC

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

MINIMUM CRITICAL POWER RATIO

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

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

BASES

2.0 The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

Insert #1 here

Insert #1

For certain conditions of pressure and flow, the ANFB correlation is not valid for all critical power calculations. The ANFB correlation is not valid for bundle mass velocities less than 0.10×10^6 lbs/hr-ft^2 (equivalent to a core flow of less than 10%) or pressures less than 590 psia. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr (approximately a mass velocity of 0.25×10^6 lbs/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full-scale ATLAS test data taken at pressures from 14.7 to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

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.

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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 (as supplemented) November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340, General Electric Company, June 1974.

3. ANFB Critical Power Correlation, ANF/EMF-1125 (P) (A), (as supplemented), 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, November 1990.

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

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4.1.2 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. 21% The results of this analysis show that the reactor is shutdown and peak power is limited to 1% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ~~ROD DENSITY~~ critical control rod configuration and the predicted ~~ROD DENSITY~~ critical control rod configuration shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

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

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ~~ROD DENSITY~~ critical control rod configuration and the predicted ~~ROD DENSITY~~ critical control rod configuration shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1. 1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta K or R + 0.28% delta K, as appropriate. The value of R in units of % delta K is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning-of-life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Replace with Reactivity Anomaly Insert

~~Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.~~

Reactivity Anomaly Insert

~~BASES (continued)~~

~~LCO~~

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the difference between the monitored core k_{eff} and the predicted core k_{eff} of 1% $\Delta k/k$ has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

Alternatively predicted control rod configuration can be compared with actual control rod configuration, and shown to be within 1% delta K/K.

~~APPLICABILITY~~

~~In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted, and, therefore, the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, reactivity anomaly is not required during these conditions.~~

~~ACTIONS~~

~~A.1~~

~~Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core~~

~~(continued)~~

3/4.2.3 MINIMUM CRITICAL POWER RATIO

SURVEILLANCE REQUIREMENTS

~~4.2.3~~ 4.2.3.1 MCPR, with:

a. τ_{ave} = 0.86 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit 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 MCPR.

Insert #3 here

Insert #3

4.2.3.2

The applicable MCPR limit shall be determined from the COLR based on:

a. Technical Specification Scram Speed (TSSS) MCPR limits, or

b. Nominal Scram Speed (NSS) MCPR limits if scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times identified in the COLR.

Within 72 hours of completion of each set of scram testing, the results will be compared against the nominal scram speed (NSS) insertion times specified in the COLR, to verify the applicability of the transient analyses. Prior to initial scram time testing for an operating cycle, the MCPR operating limits used shall be based on the Technical Specification Scram Speeds (TSSS).

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe (TIP) system shall be OPERABLE with:

a. Movable detectors, drives and readout equipment to map the core in the required measurement locations and

b. Indexing equipment to allow all required detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

*a. Recalibration of the LPRM detectors, and

*b. Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

ACTION:

a. With one or more TIP measurement locations inoperable, required measurements may be performed as described in 1 and 2 below, provided the reactor core is operating in an octant symmetric control rod pattern, and the total core TIP uncertainty for the present cycle has been measured to be less than 8.7 percent.

   1. TIP data for an inoperable measurement location may be replaced by data obtained from that string's redundant (symmetric) counterpart if the substitute TIP data was obtained from an operable measurement location.

   2. TIP data for an inoperable measurement location may be replaced by data obtained from a 3-dimensional BWR core simulator code normalized with available operating measurements, provided the total number of simulated channels (measurement locations) does not exceed:

      a) All channels of a single TIP machine, or

      b) A total of five channels if more than one TIP machine is involved.

b. Otherwise, with the TIP system inoperable, suspend use of the system for the above applicable monitoring or calibration functions.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the above applicable monitoring or calibration functions.

*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

a. With only one (1) reactor coolant system recirculation loop in operation, comply with Specification 3.4.1.5 and:

Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting condition for operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.

1. Within four (4) hours:

a) Place the recirculation flow control system in the Master Manual mode or lower, and

b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2, and

c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation by 0.01 per Specification 3.2.3, and,

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

e)

2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.

b. With no reactor coolant recirculation loops in operation:

1. Take the ACTION required by Specification 3.4.1.5, and

2. Be in at least HOT SHUTDOWN within the next six (6) hours.

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 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."

3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 50 to 125 minutes. A normal quantity of 4587 gallons net of solution having a 13.4% sodium pentaborate concentration is required to meet a shutdown requirement of 3%. There is an additional allowance of 25% in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel.

The temperature requirement on the sodium pentaborate solution is necessary to maintain the solubility of the solution as it was initially mixed to the appropriate concentration. Checking the volume of fluid and the temperature once each 24 hours assures that the solution is available for injection.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972

2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972

3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

GE Fuel

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor.

However, the for GE fuel ~~current~~ General Electric (GE) calculational models (SAFER/GESTR described in Reference ~~5~~ 8), which are consistent with the requirements of Appendix K to 10 CFR 50, have established that APLHGR values are not expected ~~to be~~ limited by LOCA/ECCS considerations. APLHGR limits are still required, ~~however~~, to assure that fuel rod mechanical integrity is maintained. They are specified for all ~~resident~~ GE fuel types in the ~~Core Operating Limit Report~~ CORE OPERATING LIMITS REPORT based on the fuel thermal-mechanical design analysis.

The purpose of the power- and flow-dependent MAPLHGR factors specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At less than 100% of rated flow or rated power, the required MAPLHGR is the minimum of either (a) the product of the rated MAPLHGR limit and the power-dependent MAPLHGR factor or (b) the product of the rated MAPLHGR limit and the flow-dependent MAPLHGR factor. The power- and flow-dependent MAPLHGR factors assure that the fuel remains within the fuel design basis during transients at off-rated conditions. Methodology for establishing these factors is described in Reference ~~6~~. 9

Insert #4 here

Insert #4

SPC Fuel

This specification assures that the peak cladding temperature of SPC fuel following a postulated design basis loss-of-coolant accident will not exceed the peak cladding temperature (PCT) and maximum oxidation limits specified in 10CFR50.46. The calculational procedure used to establish the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits is based on a loss-of-coolant accident analysis. The analysis is performed using calculational models which are consistent with the requirements of APPENDIX K to 10CFR50. The models are described in Reference 1.

The PCT following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod-to rod power distribution within the assembly.

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits for two-loop operation are specified in the CORE OPERATING LIMITS REPORT (COLR).

For single-loop operation, an APLHGR limit corresponding to the product of the two-loop limit and a reduction factor specified in the COLR can be conservatively used to ensure that the PCT for single-loop operation is bound by the PCT for two-loop operation.

3/4.2.2 DELETED

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in the CORE OPERATING LIMITS REPORT.

Analyses have been performed to determine the effects on CRITICAL POWER RATIO (CPR) during a transient assuming that certain equipment is out of service. A detailed description of the analyses is provided in Reference 5. The analyses performed assumed a single failure only and established the licensing bases to allow continuous plant operation with the analyzed equipment out of service. The following single equipment failures are included as part of the transient analyses input assumptions:

1) main turbine bypass system out of service,
2) recirculation pump trip system out of service,

Insert #5 here

Insert #5

The purpose of the power- and flow-dependent MCPR limits ($MCPR_p$ and $MCPR_f$ respectively) specified in the CORE OPERATING LIMITS REPORT (COLR) is to define operating limits dependent on core flow and core power. At a given power and flow operating condition, the required MCPR is the maximum of either the power-dependent MCPR limit or the flow-dependent MCPR limit. The required MCPR limit assures that the Safety Limit MCPR will not be violated.

The flow dependent MCPR limits ($MCPR_f$) are established to protect the core from inadvertent core flow increases. The core flow increase event used to establish the limits is a slow flow runout to maximum flow that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1, Item 2). A conservative flow control line is used to define several core power/flow state points at which the analyses are performed. $MCPR_f$ limits are established to support both the automatic and manual modes of operation. In the automatic mode, $MCPR_f$ limits are established to protect the operating limit MCPR. For the manual mode, the limits are set to protect against violation of the safety limit MCPR.

The power-dependent MCPR limits, ($MCPR_p$), are established to protect the core from plant transients other than core flow increases, including pressurization and the localized control rod withdrawal error events.

Analyses have been performed to determine the effects of assuming various equipment out-of-service scenarios on the (CPR) during transient events. Scenarios were performed to allow continuous plant operation with these systems out of service. Appropriate MCPR limits and/or penalties are included in the COLR for each of the equipment out-of-service scenarios identified in the COLR. In some cases, the reported limits or penalties are based on a cycle-independent analysis, while in other cases, analyses are performed on a cycle-specific basis.

References 2-6 describe the methodology and codes used to evaluate the potentially bounding non-LOCA transient events identified in Chapter 15 of the UFSAR.

MCPR limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits takes advantage of improved scram insertion rates, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.1.3.3. If the scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times, the appropriate NSS MCPR limits identified in the COLR are applied. If the scram insertion times do not meet the NSS insertion criteria, the TSSS MCPR limits are applied.

MINIMUM CRITICAL POWER RATIO (Continued)

3) safety/relief valve (S/RV) out of service, and
4) feedwater heater out of service (corresponding to a 100 degree F reduction in feedwater temperature).

For the main turbine bypass and recirculation pump trip systems, specific cycle-independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values are established to allow continuous plant operation with these systems out of service. A bounding end-of-cycle exposure condition was used to develop nuclear input to the transient analysis model. The bounding exposure condition assumes a more top-peaked axial power distribution than the nominal power shape, thus yielding a bounding scram response with reasonable conservatisms for the MCPR LCO values in future cycles. The MCPR LCO values shown in the CORE OPERATING LIMITS REPORT for the main turbine bypass and recirculation pump trip systems out of service are valid provided that these limits bound the cycle specific results.

The analysis for main turbine bypass and recirculation pump trip systems inoperable allows operation with either system inoperable, but not both at the same time.

For operation with the feedwater heater out of service, a cycle specific analysis will be performed. With reduced feedwater temperature, the Load Reject Without Bypass event will be less severe because of the reduced core steaming rate and lower initial void fraction. Consequently, no further analysis is needed for that event. However, the feedwater controller failure event becomes more severe with a feedwater heater out of service and could become the limiting transient for a specific cycle. Consequently, the cycle specific analysis for the feedwater controller failure event will be performed with a 100 degree F feedwater temperature reduction. The calculated change in CPR for that event will then be used in determing the cycle specific MCPR LCO value.

In the case of a single S/RV out of service, transient analysis results showed that there is no impact on the calculated MCPR LCO value. The change in CPR for this operating condition will be bounded by reload licensing calculations, and no further analyses are required. The analysis for a single S/RV out of service is valid in conjunction with dual and single recirculation loop operation.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-1 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate events are described in

MINIMUM CRITICAL POWER RATIO (Continued)

NEDE-24011-P-A-US (Reference 4). The outputs of these programs along with the initial MCPR form the input for further analyses of the thermally limiting bundle (Reference 4). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The need to adjust the MCPR operating limit as a function of scram time arises from the statistical approach used in the implementation of the ODYN computer code for analyzing rapid pressurization events. Generic statistical analyses were performed for plant groupings of similar design which considered the statistical variation in several parameters, i.e., initial power level, CRD scram insertion time, and model uncertainty. These analyses, which are described further in Reference 2, produced generic Statistical Adjustment Factors which have been applied to plant and cycle specific ODYN results to yield operating limits which provide a 95% probability with 95% confidence that the limiting pressurization event will not cause MCPR to fall below the fuel cladding integrity Safety Limit.

As a result of this 95/95 approach, the average 20% insertion scram time must be monitored to assure compliance with the assumed statistical distribution. If the mean value on a cycle cumulative, running average, basis were to exceed a 5% significance level compared to the distribution assumed in the ODYN statistical analyses, the MCPR limit must be increased linearly, as a function of the mean 20% scram time, to a more conservative value which reflects an NRC determined uncertainty penalty of 4.4%. This penalty is applied to the plant specific ODYN results, i.e. without statistical adjustment, for the limiting single failure pressurization event occurring at the limiting point in the cycle. It is not applied in full until the mean of all current cycle 20% scram times reaches the 0.86 seconds value of Specification 3.1.3.3. In practice, however, the requirements of 3.1.3.3 would most likely be reached, i.e., individual data set average > 0.86 secs, and the required actions taken well before the running average exceeds 0.86 secs.

The 5% significance level is defined in Reference 4 as:

$$\tau_B = \mu + 1.65 \left(N_1 / \sum_{i=1}^{n} N_i\right)^{1/2} \sigma$$

where μ = mean value for statistical scram time distribution to 20% inserted = 0.672

σ = standard deviation of above distribution = 0.016

N_1 = number of rods tested at BOC, i.e., all operable rods

$\sum_{i=1}^{n} N_i$ = total number of operable rods tested in the current cycle

MINIMUM CRITICAL POWER RATIO (Continued)

The value for τ_B used in Specification 3.2.3 is 0.687 seconds which is conservative for the following reason:

For simplicity in formulating and implementing the LCO, a conservative value for $\sum_{i=1}^{n} N_i$ of 598 was used. This represents one full core data set at BOC plus one full core data set following a 120 day outage plus twelve 10% of core, 19 rods, data sets. The 12 data sets are equivalent to 24 operating months of surveillance at the increased surveillance frequency of one set per 60 days required by the action statements of Specifications 3.1.3.2 and 3.1.3.4.

That is, a cycle length was assumed which is longer than any past or contemplated refueling interval and the number of rods tested was maximized in order to simplify and conservatively reduce the criteria for the scram time at which MCPR penalization is necessary.

The purpose of the power- and flow-dependent MCPR limits specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At a given power and flow operating condition, the required MCPR is the maximum of either the power-dependent MCPR limit or the flow-dependent MCPR limit. The required MCPR assures that the Safety Limit MCPR will not be violated. Methodology for establishing the power- and flow-dependent MCPR limits is described in Reference 6.

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

3/4.2.4 LINEAR HEAT GENERATION RATE

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, NEDO-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, 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 (as supplemented), 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 (as supplemented), Exxon Nuclear Company, January 1987.

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

5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 (as supplemented), 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 (as supplemented), Exxon Nuclear Company, February 1987.

7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1 (as supplemented), Siemens Power Corporation - Nuclear Division, 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.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

~~A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the Bases for Specification 3.2.3.~~

Insert #8

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.

Insert #8

Analyses were performed to support continued operation with one or both trip systems of the EOC-RPT inoperable. The analyses provide MINIMUM CRITICAL POWER RATIO (MCPR) values which must be used if the EOC-RPT system is inoperable. These MCPR limits are included in the COLR and ensure that adequate margin to the MCPR safety limit exists with the EOC-RPT function inoperable. Application of these limits are discussed further in the bases for Specification 3.2.3.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRMs and they can be retracted.

Deleted

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe (TIP) system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

The specification allows use of substituted TIP data from symmetric channels if the control rod pattern is symmetric since the TIP data is adjusted by the plant computer to remove machine dependent and power level dependent bias. The source of data for the substitution may also be a 3-dimensional BWR core simulator calculated data set which is normalized to available real data. Since uncertainty could be introduced by the simulation and normalization process, an evaluation of the specific control rod pattern and core operating state must be performed to ensure that adequate margin to core operating limits is maintained.

3/4.3.7.8 DELETED

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

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 Power Ratio (MCPR) ~~(including 20% scram time, tau (τ)~~ Scram time, 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).

Insert #9

1. ANFB Critical Power Correlation, ANF/EMF-1125(P)(A), (as supplemented).
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, ANF-524(P)(A) Revision 2 (as supplemented).
4. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 (as supplemented).
5. HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A) (as supplemented).
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.
7. Exxon Nuclear Methodology Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, 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, January 1987,
9. Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A).
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 (as supplemented).
11. Volume 1 - STAIF - A Computer Program for BWR Stability in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A).
12. RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 (as supplemented).
13. XCOBRA-T: A computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 (as supplemented).
14. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A).
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.

Insert #9 (continued)

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

17. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A).

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.

4.0 DESIGN FEATURES

Insert #10

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

(continued)

DEFINITIONS

SECTION

1.0 DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.3 INSTRUMENTATION

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.8 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.A.6. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to energization of the recirculation pump circuit breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

DELETED

FRACTION OF LIMITING POWER DENSITY

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

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

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

1.20 DELETED

LIMITING CONTROL ROD PATTERN

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

LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. LHGR is monitored by the ratio of LHGR to its fuel specific limit, as specified in the CORE OPERATING LIMITS REPORT.

LOGIC SYSTEM FUNCTIONAL TEST

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

Deleted

MAXIMUM FRACTION OF LIMITING POWER DENSITY

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

MEMBER(S) OF THE PUBLIC

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

MINIMUM CRITICAL POWER RATIO

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

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

2.1 SAFETY LIMITS

BASES

The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

Insert #1

Insert #1

For certain conditions of pressure and flow, the ANFB correlation is not valid for all critical power calculations. The ANFB correlation is not valid for bundle mass velocities less than 0.10×10^6 lbs/hr-ft^2 (equivalent to a core flow of less than 10%) or pressures less than 590 psia. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr (approximately a mass velocity of 0.25×10^6 lbs/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full-scale ATLAS test data taken at pressures from 14.7 to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

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.

Insert #2 here

e

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.

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 (as supplemented) November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340, General Electric Company, June 1974.

3. ANFB Critical Power Correlation, ANF/EMF-1125 (P) (A), (as supplemented),* 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, November 1990.

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

*Until ANF/EMF-1125 Supplement 1 Appendix C (ANFB Critical Power Correlation Application for Co-Resident Fuel) is approved by the NRC, cycle specific evaluations are submitted (e.g. EMF-96-021, Application of the ANFB Critical Power Correlation to Co-Resident Fuel for LaSalle Unit 2 Cycle 8).

THERMAL POWER, High Pressure and High Flow (Continued)

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.

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4.1.2 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to ~~1%~~ 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because

REACTIVITY CONTROL SYSTEM

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ~~ROD DENSITY~~ critical control rod configuration and the predicted ~~ROD DENSITY~~ critical control rod configuration shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

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

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ~~ROD DENSITY~~ critical control rod configuration and the predicted ~~ROD DENSITY~~ critical control rod configuration shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta K or R + 0.28% delta K, as appropriate. The value of R in units of % delta K is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning-of-life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

Replace with Reactivity Anomaly Insert

3/4.1.2 REACTIVITY ANOMALIES

~~Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.~~

Reactivity Anomaly Insert

~~BASES (continued)~~

~~LCO~~

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the difference between the monitored core k_{eff} and the predicted core k_{eff} of 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

Alternatively, predicted control rod configuration can be compared with actual control rod configuration, and shown to be within 1% delta K/K.

~~APPLICABILITY~~

~~In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and monitored core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted, and, therefore, the reactor is in the least reactive state, where monitoring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and monitored core reactivity at cold conditions; therefore, reactivity anomaly is not required during these conditions.~~

~~ACTIONS~~

~~A.1~~

~~Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core~~

~~(continued)~~

3/4.2.3 MINIMUM CRITICAL POWER RATIO

SURVEILLANCE REQUIREMENTS

~~4.2.3~~ 4.2.3.1 MCPR, with:

a. τ_{ave} = 0.86 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit 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 MCPR.

Insert #3 here

Insert #3

4.2.3.2

The applicable MCPR limit shall be determined from the COLR based on:

a. Technical Specification Scram Speed (TSSS) MCPR limits, or

b. Nominal Scram Speed (NSS) MCPR limits if scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times identified in the COLR.

Within 72 hours of completion of each set of scram testing, the results will be compared against the nominal scram speed (NSS) insertion times specified in the COLR, to verify the applicability of the transient analyses. Prior to initial scram time testing for an operating cycle, the MCPR operating limits used shall be based on the Technical Specification Scram Speeds (TSSS).

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe (TIP) system shall be OPERABLE with:

a. Movable detectors, drives and readout equipment to map the core in the required measurement locations and

b. Indexing equipment to allow all required detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

*a. Recalibration of the LPRM detectors, and
*b. Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

ACTION:

a. With one or more TIP measurement locations inoperable, required measurements may be performed as described in 1 and 2 below, provided the reactor core is operating in an octant symmetric control rod pattern, and the total core TIP uncertainty for the present cycle has been measured to be less than 8.7 percent.

   1. TIP data for an inoperable measurement location may be replaced by data obtained from that string's redundant (symmetric) counterpart if the substitute TIP data was obtained from an operable measurement location.

   2. TIP data for an inoperable measurement location may be replaced by data obtained from a 3-dimensional BWR core simulator code normalized with available operating measurements, provided the total number of simulated channels (measurement locations) does not exceed:

      a) All channels of a single TIP machine, or

      b) A total of five channels if more than one TIP machine is involved.

b. Otherwise, with the TIP system inoperable, suspend use of the system for the above applicable monitoring or calibration functions.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the above applicable monitoring or calibration functions.

*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

3/4.3.7.7 AND 3/4.3.7.8 INTENTIONALLY LEFT BLANK

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3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

a. With only one (1) reactor coolant system recirculation loop in operation, comply with Specification 3.4.1.5 and:

1. Within four (4) hours:

a) Place the recirculation flow control system in the Master Manual mode or lower, and

b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2, and

c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation by 0.01 per Specification 3.2.3, and,

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

e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.

2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.

b. With no reactor coolant recirculation loops in operation:

1. Take the ACTION required by Specification 3.4.1.5, and

2. Be in at least HOT SHUTDOWN within the next six (6) hours.

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 waar 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 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."

3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 50 to 125 minutes. A normal quantity of 4587 gallons net of solution having a 13.4% sodium pentaborate concentration is required to meet a shutdown requirement of 3%. There is an additional allowance of 25% in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel.

The temperature requirement on the sodium pentaborate solution is necessary to maintain the solubility of the solution as it was initially mixed to the appropriate concentration. Checking the volume of fluid and the temperature once each 24 hours assures that the solution is available for injection.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972

2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NEDO-10527, July 1972

3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

BASES

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

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

GE fuel

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. This specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady-state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor.

However, the current General Electric (GE) calculational models (SAFER/GESTR described in Reference ~~8~~ 8, which are consistent with the requirements of Appendix K to 10 CFR 50, have established that APLHGR values for GE fuel are not expected to be limited by LOCA/ECCS considerations. APLHGR limits are still required, ~~however~~, to assure that fuel rod mechanical integrity is maintained. They are specified for all ~~resident~~ GE fuel types in the ~~Core Operating Limit Report~~ CORE OPERATING LIMIT REPORT based on the fuel thermal-mechanical design analysis.

The purpose of the power- and flow-dependent MAPLHGR factors specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At less than 100% of rated flow or rated power, the required MAPLHGR is the minimum of either (a) the product of the rated MAPLHGR limit and the power-dependent MAPLHGR factor or (b) the product of the rated MAPLHGR limit and the flow-dependent MAPLHGR factor. The power- and flow-dependent MAPLHGR factors assure that the fuel remains within the fuel design basis during transients at off-rated conditions. Methodology for establishing these factors is described in Reference ~~6~~. 9

Insert #4 here

Insert #4

SPC Fuel

This specification assures that the peak cladding temperature of SPC fuel following a postulated design basis loss-of-coolant accident will not exceed the peak cladding temperature (PCT) and maximum oxidation limits specified in 10CFR50.46. The calculational procedure used to establish the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits is based on a loss-of-coolant accident analysis. The analysis is performed using calculational models which are consistent with the requirements of APPENDIX K to 10CFR50. The models are described in Reference 1.

The PCT following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod-to rod power distribution within the assembly.

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits for two-loop operation are specified in the CORE OPERATING LIMITS REPORT (COLR).

For single-loop operation, an APLHGR limit corresponding to the product of the two-loop limit and a reduction factor specified in the COLR can be conservatively used to ensure that the PCT for single-loop operation is bound by the PCT for two-loop operation.

3/4.2.2 DELETED

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in the CORE OPERATING LIMITS REPORT.

Analyses have been performed to determine the effects on CRITICAL POWER RATIO (CPR) during a transient assuming that certain equipment is out of service. A detailed description of the analyses is provided in Reference 5. The analyses performed assumed a single failure only and established the licensing bases to allow continuous plant operation with the analyzed equipment out of service. The following single equipment failures are included are part of the transient analyses input assumptions:

1. main turbine bypass system out of service,

2. recirculation pump trip system out of service,

$\boxed{\text{Insert \#5 here}}$

Insert #5

The purpose of the power- and flow-dependent MCPR limits ($MCPR_p$ and $MCPR_f$ respectively) specified in the CORE OPERATING LIMITS REPORT (COLR) is to define operating limits dependent on core flow and core power. At a given power and flow operating condition, the required MCPR is the maximum of either the power-dependent MCPR limit or the flow-dependent MCPR limit. The required MCPR limit assures that the Safety Limit MCPR will not be violated.

The flow dependent MCPR limits ($MCPR_f$) are established to protect the core from inadvertent core flow increases. The core flow increase event used to establish the limits is a slow flow runout to maximum flow that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1, Item 2). A conservative flow control line is used to define several core power/flow state points at which the analyses are performed. $MCPR_f$ limits are established to support both the automatic and manual modes of operation. In the automatic mode, $MCPR_f$ limits are established to protect the operating limit MCPR. For the manual mode, the limits are set to protect against violation of the safety limit MCPR.

The power-dependent MCPR limits, ($MCPR_p$), are established to protect the core from plant transients other than core flow increases, including pressurization and the localized control rod withdrawal error events.

Analyses have been performed to determine the effects of assuming various equipment out-of-service scenarios on the (CPR) during transient events. Scenarios were performed to allow continuous plant operation with these systems out of service. Appropriate MCPR limits and/or penalties are included in the COLR for each of the equipment out-of-service scenarios identified in the COLR. In some cases, the reported limits or penalties are based on a cycle-independent analysis, while in other cases, analyses are performed on a cycle-specific basis.

References 2-6 describe the methodology and codes used to evaluate the potentially bounding non-LOCA transient events identified in Chapter 15 of the UFSAR.

MCPR limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits takes advantage of improved scram insertion rates, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.1.3.3. If the scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times, the appropriate NSS MCPR limits identified in the COLR are applied. If the scram insertion times do not meet the NSS insertion criteria, the TSSS MCPR limits are applied.

MINIMUM CRITICAL POWER RATIO (Continued)

3. safety/relief valve (S/RV) out of service, and

4. feedwater heater out of service (corresponding to a 100 degree F reduction in feedwater temperature).

For the main turbine bypass and recirculation pump trip systems specific cycle-independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values are established to allow continuous plant operation with these systems out of service. A bounding end-of-cycle exposure condition was used to develop nuclear input to the transient analysis model. The bounding exposure condition assumes a more top-peaked axial power distribution than the nominal power shape, thus yielding a bounding scram response with reasonable conservatisms for the MCPR LCO values in future cycles. The MCPR LCO values shown in the CORE OPERATING LIMITS REPORT for the main turbine bypass and recirculation pump trip systems out of service are valid provided that these limits bound the cycle specific results.

The analysis for main turbine bypass and recirculation pump trip systems inoperable allows operation with either system inoperable, but not both at the same time.

For operation with the feedwater heater out of service, a cycle specific analysis will be performed. With reduced feedwater temperature, the Load Reject Without Bypass event will be less severe because of the reduced core steaming rate and lower initial void fraction. Consequently, no further analysis is needed for that event. However, the feedwater controller failure event becomes more severe with a feedwater heater out of service and could become the limiting transient for a specific cycle. Consequently, the cycle specific analysis for the feedwater controller failure event will be performed with a 100 degree F feedwater temperature reduction. The calculated change in CPR for that event will then be used in determining the cycle specific MCPR LCO value.

In the case of a single S/RV Out of service, transient analysis results showed that there is no impact on the calculated MCPR LCO value. The change in CPR for this operating condition will be bounded by reload licensing calculations and no further analyses are required. The analysis for a single S/RV out of service is valid in conjunction with dual and single recirculation loop operation.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-1 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate events are described

MINIMUM CRITICAL POWER RATIO (Continued)

in NEDE-24011-P-A-US (Reference 4). The outputs of these programs along with the initial MCPR form the input for further analyses of the thermally limiting bundle (Reference 4). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The need to adjust the MCPR operating limit as a function of scram time arises from the statistical approach used in the implementation of the ODYN computer code for analyzing rapid pressurization events. Generic statistical analyses were performed for plant groupings of similar design which considered the statistical variation in several parameters, i.e., initial power level, CRD scram insertion time, and model uncertainty. These analyses, which are described further in Reference 2, produced generic Statistical Adjustment Factors which have been applied to plant and cycle specific ODYN results to yield operating limits which provide a 95% probability with 95% confidence that the limiting pressurization event will not cause MCPR to fall below the fuel cladding integrity Safety Limit.

As a result of this 95/95 approach, the average 20% insertion scram time must be monitored to assure compliance with the assumed statistical distribution. If the mean value on a cycle cumulative, running average, basis were to exceed a 5% significance level compared to the distribution assumed in the ODYN statistical analyses, the MCPR limit must be increased linearly, as a function of the mean 20% scram time, to a more conservative value which reflects an NRC determined uncertainty penalty of 4.4%. This penalty is applied to the plant specific ODYN results, i.e. without statistical adjustment, for the limiting single failure pressurization event occurring at the limiting point in the cycle. It is not applied in full until the mean of all current cycle 20% scram times reaches the 0.86 seconds value of Specification 3.1.3.3. In practice, however, the requirements of 3.1.3.3 would most likely be reached, i.e., individual data set average > 0.86 secs, and the required actions taken well before the running average exceeds 0.86 secs.

The 5% significance level is defined in Reference 4 as:

$$\tau_B = \mu + 1.65 \left(N_1 / \sum_{i=1}^{n} N_i\right)^{1/2} \sigma$$

where μ = mean value for statistical scram time distribution to 20% inserted = .672

σ = standard deviation of above distribution = .016

N_1 = number of rods tested at BOC, i.e., all operable rods

$\sum_{i=1}^{n} N_i$ = total number of operable rods tested in the current cycle

MINIMUM CRITICAL POWER RATIO (Continued)

The value for τ_B used in Specification 3.2.3 is 0.687 seconds which is conservative for the following reason:

For simplicity in formulating and implementing the LCO, a conservative value for $\sum_{i=1}^{n} N_i$ of 598 was used. This represents one full core data set at BOC plus one full core data set following a 120 day outage plus twelve 10% of core, 19 rods, data sets. The 12 data sets are equivalent to 24 operating months of surveillance at the increased surveillance frequency of one set per 60 days required by the action statements of Specifications 3.1.3.2 and 3.1.3.4.

That is, a cycle length was assumed which is longer than any past or contemplated refueling interval and the number of rods tested was maximized in order to simplify and conservatively reduce the criteria for the scram time at which MCPR penalization is necessary.

The purpose of the power- and flow-dependent MCPR limits specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At a given power and flow operating condition, the required MCPR is the maximum of either the power-dependent MCPR limit or the flow-dependent MCPR limit. The required MCPR assures that the Safety Limit MCPR will not be violated. Methodology for establishing the power- and flow-dependent MCPR limits is described in Reference 6.

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

3/4.2.4 LINEAR HEAT GENERATION RATE

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, 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, 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 (as supplemented), 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 (as supplemented), Exxon Nuclear Company, January 1987.

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

5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 (as supplemented), 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 (as supplemented), Exxon Nuclear Company, February 1987.

7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1 (as supplemented), Siemens Power Corporation - Nuclear Division, 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.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (continued)

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)", Parts 1 and 2, December 1988, and RE-025 Revision 1, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", April 1991. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains ECCS initiation capability.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

~~A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.~~

Insert #8

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Insert #8

Analyses were performed to support continued operation with one or both trip systems of the EOC-RPT inoperable. The analyses provide MINIMUM CRITICAL POWER RATIO (MCPR) values which must be used if the EOC-RPT system is inoperable. These MCPR limits are included in the COLR and ensure that adequate margin to the MCPR safety limit exists with the EOC-RPT function inoperable. Application of these limits are discussed further in the bases for Specification 3.2.3.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe (TIP) system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

The specification allows use of substituted TIP data from symmetric channels if the control rod pattern is symmetric since the TIP data is adjusted by the plant computer to remove machine dependent and power level dependent bias. The source of data for the substitution may also be a 3-dimensional BWR core simulator calculated data set which is normalized to available real data. Since uncertainty could be introduced by the simulation and normalization process, an evaluation of the specific control rod pattern and core operating state must be performed to ensure that adequate margin to core operating limits is maintained.

3/4.3.7.8 DELETED

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 DELETED

Core Operating Limits Report (Continued)

Effects of analyzed equipment out of service are included.

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

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

Insert #9

1. ANFB Critical Power Correlation, ANF/EMF-1125(P)(A), (as supplemented).

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, ANF-524(P)(A) Revision 2 (as supplemented).

4. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 (as supplemented).

5. HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A) (as supplemented).

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.

7. Exxon Nuclear Methodology Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, 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, January 1987.

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

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 (as supplemented).

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

12. RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 (as supplemented).

13. XCOBRA-T: A computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 (as supplemented).

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

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).

17. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A).

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 cruicform 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.

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.

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

(continued)

Attachment C

Evaluation of Significant Hazards Evaluation

C. Evaluation of Significant Hazards Considerations

References:

1. ANF-89-014(P)(A) Rev. 1 Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9X9-IX and 9X9-9X BWR Reload Fuel.

2. EMF-94-217(P) Rev. 1, Siemens Boiling Water Reactor Licensing Methodology Summary.

The fuel supplier for LaSalle is being changed from General Electric (GE) to Siemens Power Corporation (SPC). As a result, certain items in the Technical Specifications are being revised. These changes can be classified in three categories: (a) fuel thermal limits, (b) miscellaneous, and (c) minor changes not related to the SPC transition. Each is discussed below.

a. Fuel thermal limits

The fuel thermal limits in the Technical Specifications are LHGR, APLHGR, and MCPR. Each fuel vendor provides LHGR and APLHGR limits for their fuel. As required by the Technical Specification Surveillance Requirements (SR's), each fuel type will continue to be monitored via its vendor supplied LHGR and APLHGR limits. As such, the change to the Technical Specifications Bases for LHGR and APLHGR will be the addition of background information related to the SPC LHGR and APLHGR. The Limiting Conditions for Operation (LCO), Action Statements and SR's are unaffected since they refer to the Core Operating Limits Report for the fuel type dependent limits.

The CPR is calculated using a Nuclear Regulatory Commission (NRC) approved CPR correlation. The GE correlation (GEXL) is being replaced by the SPC correlation, ANFB. The co-resident GE fuel will be monitored by the ANFB correlation supplemented with bundle geometry dependent factors to ensure the calculated CPR data is conservative with respect to that which would have been calculated by the GEXL correlation. This mixed core treatment of CPR is being documented in EMF-1125(P) Supplement 1 Appendix C, "ANFB Critical Power Correlation Application for Co-Resident Fuel," November 1995. In light of the requested schedule for the approval of these Technical Specification changes, a LaSalle Unit 2 Cycle 8 specific document, EMF-96-021, "Application of the ANFB Critical Power Correlation to Co-resident GE Fuel for LaSalle Unit 2 Cycle 8, has been submitted to the NRC for interim approval. The Technical Specifications and Bases related to the GE methods for determining the operating limit for MCPR are replaced by the SPC methods.

b. Miscellaneous change

The Reactivity Anomaly surveillance is being upgraded to be consistent with SPC methods and NUREG-1434.

c. Minor Changes Not related to SPC Transition

The Traversing In-core Probe (TIP) uncertainty limits (Specification 3.3.7.7) is being re-located from the Technical Specifications as a line item improvement from the Improved Technical Specifications (NUREG-1434). The same is true for the fuel description in Specification 5. A typographical error is being corrected in the Bases (page B 2-9) related to the power level at which the IRM system terminates the low power control rod withdrawal error event.

ComEd has evaluated the proposed Technical Specification amendment and determined that 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 LaSalle Units 1 and 2 in accordance with the proposed amendment(s) 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 will be established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed Technical Specifications amendment reflects previously approved SPC methodology used to analyze normal operations, including anticipated operational occurrences (AOOs), and to determine the potential consequences of accidents.

Licensing Methods and Models

The proposed amendment is to support operation with NRC approved fuel and licensing methods supplied from Siemens Power Corporation. In accordance with FSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and methods as were analyzed by GE for GE fuel. The analysis methods and models are NRC approved (Note the mixed core treatment of CPR is being addressed under separate correspondence). These approved methods and models are used to determine the fuel thermal limits. Traversing In-core Probe (TIP) uncertainty are assumptions in the approved Siemens core monitoring methodologies. The SPC core monitoring code enables the site to monitor k_{eff} as well as rod density to perform the reactivity anomaly surveillance. This is consistent with GE methodology. Therefore, the change in licensing analysis methods and models does not significantly increase the probability of an accident or the consequences of an accident previously identified. The support systems for minimizing the consequences of transients and accidents are not affected by the proposed amendment.

New Fuel Design

The use of ATRIUM 9B fuel at LaSalle does not involve a significant increase in the probability or consequences of any accident previously evaluated in the FSAR. The ATRIUM-9B fuel is generically approved for use as a reload BWR fuel type (See Reference 1). Limiting postulated occurrences and normal operation have been analyzed using NRC-approved methods for the ATRIUM 9B fuel design to ensure that safety limits are protected and that acceptable transient and accident performance is maintained.

The reload fuel has no adverse impact on the performance of in-core neutron flux instrumentation or CRD response. The ATRIUM-9B fuel design will not adversely affect performance of neutron instrumentation nor will it adversely affect the movement of control blades. The exterior dimensions of the ATRIUM-9B fuel assembly are essentially identical to the GE9B; the ATRIUM-9B fuel assembly for LaSalle uses a standard fuel channel and normal control cell positioning (i.e., no offset). Thus, no adverse interactions with the adjacent control blade and nuclear instrumentation are anticipated. Additionally, given the above mentioned overall envelope similarities, no problems are anticipated with other station equipment such as the fuel storage racks, the new fuel inspection stand and the spent fuel pool fuel preparation machine.

The ATRIUM 9B design is neutronically compatible with the existing fuel types and core components in the LaSalle core. SPC tests have demonstrated that the ATRIUM-9B fuel design is hydraulically compatible with the GE9 fuel. The bundle pressure drop characteristics of the

ATRIUM 9B bundle are similar to those of the GE9 fuel design, hence core thermal-hydraulic stability characteristics are not adversely affected by the ATRIUM 9B design.

An evaluation of the Emergency Procedures is being performed to ensure that the use of the ATRIUM-9B fuel at LaSalle does not alter any assumptions previously made in evaluating the radiological consequences of an accident at LaSalle Station.

Methods approved by the NRC are being used in the evaluation of fuel performance during normal and abnormal operating conditions. The ComEd and SPC methods to be used for the cycle specific transient analyses have been previously NRC approved. The exception is the mixed core treatment of CPR, which is being addressed under separate correspondence.

The description of the fuel is expanded to be consistent with NUREG-1434. The description of the fuel materials, lead test assembly use, and stating that designs must have been analyzed with NRC Staff approved codes does not change existing methods; it only describes them.

Review of the above concludes that the probability of occurrence and the consequences of an accident previously evaluated in the safety analysis report have not been significantly increased.

ComEd has evaluated the proposed License amendment and determined that 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 LaSalle Units 1 and 2 in accordance with the proposed amendment(s) will not represent a significant hazards consideration for the following reasons:

These changes do not:

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 of the plant configuration, including changes in allowable modes of operation.

Licensing Methods and Models

The proposed Technical Specification amendment reflects previously approved SPC methodology used to analyze normal operations, including AOOs, and to determine the potential consequences of accidents. As stated above, the proposed changes do not permit modes of reactor operation which differ from those currently permitted.

New Fuel Design

The basic design concept of a 9x9 fuel pin array with an internal water box has been used in various lead assembly programs and in reload quantities in Europe since 1986. WNP-2 has loaded reload quantities since 1991. Approximately 650 water box assemblies have been irradiated in the United States through 1995, with a substantially higher number being irradiated overseas. The NRC has reviewed and approved the ATRIUM-9B fuel design (See Reference 1). The similarities in fuel design and operation indicate there would be no expectation of introducing new or different types of accidents than have been considered for the existing fuel. Therefore, the use of ATRIUM-9B fuel at LaSalle does not create the possibility of a new or different kind of accident from any accident previously evaluated.

ComEd has evaluated the proposed License amendment and determined that 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 LaSalle Units 1 and 2 in accordance with the proposed amendment(s) will not represent a significant hazards consideration for the following reasons:

These changes do not:

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

The existing margin to safety is provided by the existing acceptance criteria (e.g., 10CFR50.46 limits). The proposed Technical Specification amendment reflects previously approved SPC methodology used to demonstrate that the existing acceptance criteria are satisfied. The revised methodology has been previously reviewed and approved by the USNRC for application to reload cores of GE BWRs. References for the Licensing Topical Reports which document this methodology, and include the Safety Evaluation Reports prepared by the USNRC, are added to the Reference section of the Technical Specifications as part of this amendment.

Licensing Methods and Models

The proposed amendment does not involve changes to the existing operability criteria. NRC approved methods and established limits (implemented in the COLR) ensure acceptable margin is maintained. The ComEd and SPC reload methodologies for the ATRIUM-9B reload design are consistent with the Technical Specification Bases. The Limiting Conditions for Operation are taken into consideration while performing the cycle specific and generic reload safety analyses. NRC approved methods are listed in Specification 6 of the Technical Specifications.

Analyses performed with NRC-approved methodology have demonstrated that fuel design and licensing criteria will be met during normal and abnormal operating conditions. Therefore, there is not a significant reduction in the margin of safety.

New Fuel Design

The exterior dimensions of the ATRIUM-9B fuel assembly are essentially identical to the GE9B; the ATRIUM-9B fuel assembly for LaSalle uses a standard fuel channel and normal control cell positioning; i.e. no offset. Thus, no adverse interactions with the adjacent control blade and nuclear instrumentation are anticipated. The change does not adversely impact equipment important to safety and, therefore does not reduce the margin of safety .

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. This proposed amendment most closely fits the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

This proposed amendment does not involve a significant relaxation of the criteria used to establish 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 the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

Attachment D

Environmental Assessment Applicability Review

D. 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 10 CFR 51.21. 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 or 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 E

Boiling Water Reactor Licensing Methodology Summary,
EMF-94-217