SECTION PAGE

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.7.

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than or equal to twelve inches above the top of the active irradiated fuel.* (a)

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.7.

a The top of active irradiated fuel is defined to be 360 inches above vessel zero.

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
1. Intermediate Range Monitor:	
a. Neutron Flux - High	≤120/125 divisions of full scale
b. Inoperative	NA
2. Average Power Range Monitor:	
a. Setdown Neutron Flux - High	≤15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High	
1) Dual Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 62%, with a maximum of
b) High Flow Maximum	≤120% of RATED THERMAL POWER
2) Single Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 58.5%, with a maximum of
b) High Flow Maximum	≤116.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High	≤120% of RATED THERMAL POWER
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤1060 psig
4. Reactor Vessel Water Level - Low	≥144 inches above top of active fuel (b)
5. Main Steam Line Isolation Valve - Closure	≤10% closed
6. Main Steam Line Radiation - High	≤3[b] x normal full power background (without hydrogen addition) (c)

a W shall be the recirculation loop flow expressed as a percentage of the recirculation loop flow which produces a rated core flow of 98 million lbs/hr.

b With Unit 2 operating above 20% RATED THERMAL POWER and hydrogen being injected into the primary coolant, this Unit 2 setting may be increased to "≤3 x full power background (with hydrogen addition)."

c

b The top of active fuel is defined to be 360 inches above vessel zero

BASES

2.1 SAFETY LIMITS

The Specifications in Section 2.1 establish operating parameters to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). These parameters are based on the Safety Limits requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

is calculated to

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 limit is set such that no calculated fuel damage would occur as a result of an AOO. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit for the MINIMUM CRITICAL POWER RATIO (MCPR) that represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical boundaries which separate radioactive materials from the environs. The integrity of the fuel cladding 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 protection system safety settings. While fission product migration from cladding perforations 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 integrity Safety Limit is defined with 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. Therefore, the fuel cladding integrity Safety Limit is established such that no calculated fuel damage shall result from an abnormal operational transient. This is accomplished by selecting a MCPR fuel cladding integrity Safety Limit which assures that during normal operation and AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Exceeding a Safety Limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

2.1.A THERMAL POWER, Low Pressure or Low Flow

This fuel cladding integrity Safety Limit is established by establishing a limiting condition on core THERMAL POWER developed in the following method. At pressures below 800 psia (~785 psig), the core elevation pressure drop (0% power, 0% flow) is greater than 4.56 psi. At low powers and flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lb/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. At 25% of RATED THERMAL POWER, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

2.1.B THERMAL POWER, High Pressure and High Flow

This includes consideration of

This fuel cladding integrity Safety Limit is set such that no (mechanistic) 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 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 ratio (CPR) 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.

such that, with at the MCPR Safety Limit,

The margin between a MCPR of 1.0 (onset of transition boiling) and the Safety Limit, is derived from a detailed statistical analysis which considers the uncertainties in monitoring the core operating state, including uncertainty in the critical power correlation. Because the transition boiling correlation is based on a significant quantity of practical test data, there is a very high confidence that operation of a fuel assembly at the condition where MCPR is equal to the fuel cladding integrity Safety Limit would not produce transition boiling. In addition, during single recirculation loop operation, the MCPR Safety Limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

fuel cladding failure due to overheating/dryout.

However, if transition boiling were to occur, cladding perforation would not necessarily be expected. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative

BASES

2.1.D Reactor Vessel Water Level

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds of the core height. The Safety Limit has been established at 12 inches above the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action. The top of active fuel is ~360 inches above vessel zero.

defined to be

irradiated

2.2 LIMITING SAFETY SYSTEM SETTINGS

The Specifications in Section 2.2 establish operational settings for the reactor protection system instrumentation which initiates the automatic protective action at a level such that the Safety Limits will not be exceeded. These settings are based on the Limiting Safety System Settings requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. "

2.2.A Reactor Protection System Instrumentation Setpoints

The Reactor Protection System (RPS) instrumentation setpoints specified in the table are the values at which the reactor scrams are set for each parameter. The scram settings 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 assist in mitigating the consequences of accidents. Conservatism incorporated into the transient analysis is documented by each approved fuel vendor. The bases for individual scram settings are discussed in the following paragraphs.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of eight chambers, four in each of the reactor protection system logic CHANNELs. The IRM is a 5 decade, 10 range, instrument which covers the range of power level between that covered by the SRM and the APRM. The IRM scram setting at 120 of 125 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal events has been analyzed. This analysis included starting the event at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM CHANNEL closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power is limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity Safety Limit. Based on the above analysis, the IRM provides protection against local

7.7

decrease as power is increased to 100% in comparison to the level outside the shroud, to a maximum of seven inches, due to the pressure drop across the steam dryer. Therefore, at 100% power, an indicated water level of +8 inches water level may be as low as +1 inches inside the shroud which corresponds to 144 inches above the top of active fuel and 504 inches above vessel zero.

(The top of active fuel is defined to be 360 inches above vessel zero.)

5. Main Steam Line Isolation Valve - Closure

Automatic isolation of the main steam lines is provided to give protection against rapid reactor depressurization and cooldown of the vessel. When the main steam line isolation valves begin to close, a scram signal provides for reactor shutdown so that high power operation at low reactor pressures does not occur. With the scram setting at 10% valve closure (from full open), there is no appreciable increase in neutron flux during normal or inadvertent isolation valve closure, thus providing protection for the fuel cladding integrity Safety Limit. Operation of the reactor at pressures lower than the MSIV closure setting requires the reactor mode switch to be in the Startup/Hot Standby position, where protection of the fuel cladding integrity Safety Limit is provided by the IRM and APRM high neutron flux scram signals. Thus, the combination of main steam line low pressure isolation and the isolation valve closure scram with the mode switch in the Run position assures the availability of the neutron flux scram protection over the entire range of applicability of fuel cladding integrity Safety Limit.

6. Main Steam Line Radiation - High

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. When high radiation is detected, a scram is initiated to mitigate the failure of fuel cladding. The scram setting is high enough above background radiation levels to prevent spurious scrams yet low enough to promptly detect gross failures in the fuel cladding. This setting is determined based on normal full power background (NFPB) radiation levels without hydrogen addition. With the injection of hydrogen into the feedwater for mitigation of intergranular stress corrosion cracking, the full power background levels may be significantly increased. The setting is increased based on the new background levels to allow for the injection of hydrogen. This trip function provides an anticipatory scram to limit offsite dose consequences, but is not assumed to occur in the analysis of any design basis event.

INSERT A

Current fuel designs incorporate slight variations in the length of the active fuel and, thus the actual top of active fuel, when compared to the original fuel designs. Safety Limits, water level instrument setpoints and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiations associated with these events.

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which torminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

← Insert A

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the loss-of-coolant accident (LOCA) analysis for Dresden Units 2 & 3. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

BASES

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

Alternatively, monitored K_{eff} can be compared with the predicted K_{eff} as calculated by the 3-D core simulator code.

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk/k. Deviations in core reactivity greater than 1% Δk/k are not expected and require thorough evaluation. A 1% Δk/k reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by control

rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

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.

3/4.3.D Control Rod Maximum Scram Insertion Times;

3/4.3.E Control Rod Average Scram Insertion Times; and

3/4.3.F Four Control Rod Group Scram Insertion Times

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve

INSERT B

Transient analyses are performed for both Technical Specification Scram Speed (TSSS) and Nominal Scram Speed (NSS) insertion times. These analyses result in the establishment of the fuel cycle dependent TSSS MCPR operating limits and NSS MCPR operating limits which are presented in the COLR. Results of the control rod scram timing tests performed during the current fuel cycle are used to determine the operating limit for MCPR. Following the completion of each set of scram time testing, the results will be compared with the assumptions used in the transient analysis to verify the applicability of the MCPR operating limits. Prior to the initial scram time testing for an operating cycle, the MCPR operating limits will be based on the TSSS insertion times.

solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

Insert B

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

INSERT C

The overpressure protection system must accommodate the peak transient pressure during the most severe licensing basis pressurization transient. This includes but is not limited to the licensing basis ASME Section III compliance event which is the closure of all MSIVs with no credit for solenoid relief valve function or direct scram from MSIV position. For the purpose of the ASME Section III analysis, the SRV (combination safety/relief valve) is assumed to operate in the Safety Mode, only. The ASME Section III analysis demonstrates that the combined capacity of the SVs and SRV is capable of maintaining the reactor pressure below the ASME code limit. The licensing basis pressurization transients are evaluated for each reload to assure compliance with the ASME code limit of 110% of vessel design pressure. This LCO ensures that the acceptance limit of 1375 psig is met during the most severe licensing basis pressurization transient.

reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if operation can continue. The evaluation must verify the reactor coolant system integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

The 72 hour completion time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor scram on high neutron flux. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

Insert C

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

INSERT D

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the Peak Cladding Temperature (PCT) and maximum oxidation limits specified in 10 CFR 50.46. The calculational procedure used to establish the Average Planar Linear Heat Generation Rate (APLHGR) operating limits is based on a loss-of-coolant accident analysis. The analysis is performed using calculational models which are consistent with the requirements of 10 CFR 50.46.

The PCT following a postulated loss-of-coolant accident is primarily a function of the initial condition's 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 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for two-loop and single-loop operation are specified in the Core Operating Limits Report (COLR).

INSERT E

The APRM scram settings must be adjusted to ensure that the LHGR transient limit (TLHGR) is not violated for any power distribution. This is accomplished using FDLRC. The APRM scram setting is decreased in accordance with the formula in Specification 3.11.B, when FDLRC is greater than 1.0.

The adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

INSERT D

~~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 LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.~~

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification 6.9.

10 CFR 50.46

The daily requirement for calculating APLHGR 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 to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B TRANSIENT LINEAR HEAT GENERATION RATE

The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that ≥1% plastic strain does not occur; and, the fuel does not experience centerline melt during anticipated operational occurrences beginning at any power level and terminating at 120% of RATED THERMAL POWER. ~~The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.~~

INSERT E

The daily requirement for calculating FDLRC when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when

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MCPR Operating 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 3.3.E. The measured scram insertion times are compared with the nominal scram insertion times and the Technical Specification Scram Speeds. The appropriate operating limit is applied, as specified in the COLR.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the Safety Limit MCPR in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying the appropriate flow dependent MCPR limits presented in the COLR. The MCPR Operating Limit for a given power/flow state is the greatest value of MCPR as given by the rated conditions MCPR limit or the flow dependent MCPR limit. For automatic flow control, in addition to protecting the Safety Limit MCPR during the flow run-up event, protection is provided to prevent exceeding the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow.

there have not been significant power or control rod changes. The requirement to calculate FDLRC within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating FDLRC after initially determining FDLRC is greater than 1.0 exists to ensure that FDLRC will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

The FUEL DESIGN LIMIT RATIO FOR CENTERLINE MELT (FDLRC) is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)};$$

where LHGR is the LINEAR HEAT GENERATION RATE, and TLHGR is the TRANSIENT LINEAR HEAT GENERATION RATE. The TLHGR is specified in the CORE OPERATING LIMITS REPORT.

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C 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 the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change 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 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in Specification 6.9.

The purpose of the reduced flow MCPR curves specified in the CORE OPERATING LIMITS REPORT are to define MCPR operating limits at other than rated core flow conditions. The reduced flow MCPR curves assure that the Safety Limit MCPR will not be violated.

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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 has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for

5.0 DESIGN FEATURES

5.3 REACTOR CORE

The assemblies may contain water rods or a water box.

Fuel Assemblies

5.3.A The reactor core shall contain 724 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material ~~and water rods~~. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Control Rod Assemblies

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

5.0 DESIGN FEATURES

5.4 REACTOR COOLANT SYSTEM

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Design Pressure and Temperature

5.4.A The reactor coolant system is designed and shall be maintained:

1. In accordance with the code requirements specified in Section 5 of the UFSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

2. For a pressure and temperature of:

   a. 1175 psig at 565°F on the suction side of the recirculation pump.

   b. 1450 psig at 575°F from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   c. 1325 psig at 580°F from the discharge shutoff valve to the jet pumps.

Volume

5.4.B The total water and steam volume of the reactor vessel and recirculation system is approximately 14,626 cubic feet at 68°F.

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3. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

4. Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to April 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

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:

(1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

(2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

(3) The Steady State Linear Heat Generation Rate (SLHGR) for Specification 3.11.D.

(4) The Minimum Critical Power Operating Limit (including 20% scram insertion time) for Specification 3.11.C. This includes rated and off-rated flow conditions.

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(7) Qualification of Exxon Nuclear Fuel for Extended Burnup: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A).

(8) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A).

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

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

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

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

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

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

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

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

(6) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology."

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

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

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c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.B Special Reports

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

SECTION PAGE

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.7.

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than or equal to twelve inches above the top of active irradiated fuel[a].

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.7.

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a The top of active irradiated fuel is defined to be 360 inches above vessel zero.

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
1. Intermediate Range Monitor:	
a. Neutron Flux - High	≤120/125 divisions of full scale
b. Inoperative	NA
2. Average Power Range Monitor:	
a. Setdown Neutron Flux - High	≤15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High	
1) Dual Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 62%, with a maximum of
b) High Flow Maximum	≤120% of RATED THERMAL POWER
2) Single Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 58.5%, with a maximum of
b) High Flow Maximum	≤116.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High	≤120% of RATED THERMAL POWER
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤1060 psig
4. Reactor Vessel Water Level - Low	≥144 inches above top of active fuel[b]
5. Main Steam Line Isolation Valve - Closure	≤10% closed

a W shall be the recirculation loop flow expressed as a percentage of the recirculation loop flow which produces a rated core flow of 98 million lbs/hr.

b The top of active fuel is defined to be 360 inches above vesel zero.

TABLE 2.2.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
6. Main Steam Line Radiation - High	≤3[c] x normal full power background (without hydrogen addition)
7. Drywell Pressure - High	≤2 psig
8. Scram Discharge Volume Water Level - High	≤40.4 gallons (Unit 2) ≤41 gallons (Unit 3)
9. Turbine Stop Valve - Closure	≤10% closed
10. Turbine EHC Control Oil Pressure - Low	≥900 psig
11. Turbine Control Valve Fast Closure	≥460 psig EHC fluid pressure
12. Turbine Condenser Vacuum - Low	≥21 inches Hg vacuum
13. Reactor Mode Switch Shutdown Position	NA
14. Manual Scram	NA

c With Unit 2 operating above 20% RATED THERMAL POWER and hydrogen being injected into the primary coolant, this Unit 2 setting may be increased to "≤3 x full power background (with hydrogen addition)."

2.1 SAFETY LIMITS

The Specifications in Section 2.1 establish operating parameters to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). These parameters are based on the Safety Limits requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

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 limit is set such that no fuel damage is calculated to occur as a result of an AOO. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit for the MINIMUM CRITICAL POWER RATIO (MCPR) that represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical boundaries which separate radioactive materials from the environs. The integrity of the fuel cladding 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 protection system safety settings. While fission product migration from cladding perforations 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 integrity Safety Limit is defined with 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. Therefore, the fuel cladding integrity Safety Limit is established such that no calculated fuel damage shall result from an abnormal operational transient. This is accomplished by selecting a MCPR fuel cladding integrity Safety Limit which assures that during normal operation and AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Exceeding a Safety Limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

2.1.A THERMAL POWER, Low Pressure or Low Flow

This fuel cladding integrity Safety Limit is established by establishing a limiting condition on core THERMAL POWER developed in the following method. At pressures below 800 psia (~785 psig), the core elevation pressure drop (0% power, 0% flow) is greater than 4.56 psi. At low powers and flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lb/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. At 25% of RATED THERMAL POWER, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

2.1.B THERMAL POWER, High Pressure and High Flow

This fuel cladding integrity Safety Limit is set such that no (mechanistic) 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 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 ratio (CPR) 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 such that, with the limiting fuel assembly at the MCPR Safety Limit, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. This includes consideration of the power distribution within the core and all uncertainties.

The margin between a MCPR of 1.0 (onset of transition boiling) and the Safety Limit, is derived from a detailed statistical analysis which considers the uncertainties in monitoring the core operating state, including uncertainty in the critical power correlation. Because the transition boiling correlation is based on a significant quantity of practical test data, there is a very high confidence that operation of a fuel assembly at the condition where MCPR is equal to the fuel cladding integrity Safety Limit would not produce fuel cladding failure due to overheating/dryout. In addition, during single recirculation loop operation, the MCPR Safety Limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

However, if transition boiling were to occur, cladding perforation would not necessarily be expected. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative

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

2.1.C Reactor Coolant System Pressure

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

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

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

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

2.1.D Reactor Vessel Water Level

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds of the core height. The Safety Limit has been established at 12 inches above the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action. The top of active irradiated fuel is defined to be 360 inches above vessel zero.

2.2 LIMITING SAFETY SYSTEM SETTINGS

The Specifications in Section 2.2 establish operational settings for the reactor protection system instrumentation which initiates the automatic protective action at a level such that the Safety Limits will not be exceeded. These settings are based on the Limiting Safety System Settings requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. "

2.2.A Reactor Protection System Instrumentation Setpoints

The Reactor Protection System (RPS) instrumentation setpoints specified in the table are the values at which the reactor scrams are set for each parameter. The scram settings 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 assist in mitigating the consequences of accidents. Conservatism incorporated into the transient analysis is documented by each approved fuel vendor. The bases for individual scram settings are discussed in the following paragraphs.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of eight chambers, four in each of the reactor protection system logic CHANNELs. The IRM is a 5 decade, 10 range, instrument which covers the range of power level between that covered by the SRM and the APRM. The IRM scram setting at 120 of 125 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal events has been analyzed. This analysis included starting the event at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM CHANNEL closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power is limited to 7.7% of rated power, thus maintaining MCPR above the fuel cladding integrity Safety Limit. Based on the above analysis, the IRM provides protection against local

decrease as power is increased to 100% in comparison to the level outside the shroud, to a maximum of seven inches, due to the pressure drop across the steam dryer. Therefore, at 100% power, an indicated water level of +8 inches water level may be as low as +1 inches inside the shroud which corresponds to 144 inches above the top of active fuel and 504 inches above vessel zero. The top of active fuel is defined to be 360 inches above vessel zero.

5. Main Steam Line Isolation Valve - Closure

Automatic isolation of the main steam lines is provided to give protection against rapid reactor depressurization and cooldown of the vessel. When the main steam line isolation valves begin to close, a scram signal provides for reactor shutdown so that high power operation at low reactor pressures does not occur. With the scram setting at 10% valve closure (from full open), there is no appreciable increase in neutron flux during normal or inadvertent isolation valve closure, thus providing protection for the fuel cladding integrity Safety Limit. Operation of the reactor at pressures lower than the MSIV closure setting requires the reactor mode switch to be in the Startup/Hot Standby position, where protection of the fuel cladding integrity Safety Limit is provided by the IRM and APRM high neutron flux scram signals. Thus, the combination of main steam line low pressure isolation and the isolation valve closure scram with the mode switch in the Run position assures the availability of the neutron flux scram protection over the entire range of applicability of fuel cladding integrity Safety Limit.

6. Main Steam Line Radiation - High

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. When high radiation is detected, a scram is initiated to mitigate the failure of fuel cladding. The scram setting is high enough above background radiation levels to prevent spurious scrams yet low enough to promptly detect gross failures in the fuel cladding. This setting is determined based on normal full power background (NFPB) radiation levels without hydrogen addition. With the injection of hydrogen into the feedwater for mitigation of intergranular stress corrosion cracking, the full power background levels may be significantly increased. The setting is increased based on the new background levels to allow for the injection of hydrogen. This trip function provides an anticipatory scram to limit offsite dose consequences, but is not assumed to occur in the analysis of any design basis event.

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which terminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

Current fuel designs incorporate slight variations in the length of the active fuel and, thus the actual top of active fuel, when compared to the original fuel designs. Safety Limits, water level instrument setpoints and associated LCOs refer to the top of active fuel. In these cases, the top of active fuel is defined as 360 inches above vessel zero. Licensing analyses, both accident and transient, utilize this definition for the automatic initiations associated with these events.

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Alternatively, monitored K_{eff} can be compared with the predicted K_{eff} as calculated by the 3-D core simulator code. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

BASES

Control rods that are inoperable due to exceeding allowed scram times, but are movable by control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

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.

3/4.3.D Control Rod Maximum Scram Insertion Times;

3/4.3.E Control Rod Average Scram Insertion Times; and

3/4.3.F Four Control Rod Group Scram Insertion Times

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve

solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Transient analyses are performed for both Technical Specification Scram Speed (TSSS) and Nominal Scram Speed (NSS) insertion times. These analyses result in the establishment of the fuel cycle dependent TSSS MCPR operating limits and NSS MCPR operating limits which are presented in the COLR. Results of the control rod scram timing tests performed during the current fuel cycle are used to determine the operating limit for MCPR. Following the completion of each set of scram time testing, the results will be compared with the assumptions used in the transient analysis to verify the applicability of the MCPR operating limits. Prior to the initial scram time testing for an operating cycle, the MCPR operating limits will be based on the TSSS insertion times. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed asan indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if operation can continue. The evaluation must verify the reactor coolant system integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

The 72 hour completion time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the peak transient pressure during the most severe licensing basis pressurization transient. This includes but is not limited to the licensing basis ASME Section III compliance event which is closure of all MSIVs with no credit for solenoid relief valve function or direct scram from MSIV position. For the purpose of the ASME Section III analysis, the SRV (combination safety/relief valve) is assumed to operate in the Safety Mode, only. The ASME Section III analysis demonstrates that the combined capacity of the SVs and SRV is capable of maintaining the reactor pressure below the ASME code limit. The licensing basis pressurization transients are evaluated for each reload to assure compliance with the ASME code limit of 110% of vessel design pressure. This LCO ensures that the acceptance limit of 1375 psig is met during the most severe licensing basis pressurization transient.

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which

closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the Peak Cladding Temperature (PCT) and maximum oxidation limits specified in 10 CFR 50.46. The calculational procedure used to establish the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) operating limits is based on a loss-of-coolant accident analysis. The analysis is performed using calculational models which are consistent with the requirements of 10 CFR 50.46.

The PCT following a postulated loss-of-coolant accident is primarily a function of the initial condition's 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 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for two-loop and single-loop operation are specified in the Core Operating Limits Report (COLR).

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of 10 CFR 50.46. The approved calculational models are listed in Specification 6.9.

The daily requirement for calculating APLHGR 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 to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B TRANSIENT LINEAR HEAT GENERATION RATE

The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that ≥1% plastic strain does not occur; and, the fuel does not experience centerline melt during anticipated operational occurrences beginning at any power level and terminating at 120% of RATED THERMAL POWER.

The APRM scram settings must be adjusted to ensure that the LHGR transient limit (TLHGR) is not violated for any power distribution. This is accomplished by using FDLRC. The APRM scram setting is decreased in accordance with the formula in Specification 3.11.B, when FDLRC is greater than 1.0.

The adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. This provides the same degree of protection as reducing the trip setting by 1/FDLRC by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

The daily requirement for calculating FDLRC 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 to calculate FDLRC within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating FDLRC after initially determining FDLRC is greater than 1.0 exists to ensure that FDLRC will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

The FUEL DESIGN LIMIT RATIO FOR CENTERLINE MELT (FDLRC) is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)};$$

where LHGR is the LINEAR HEAT GENERATION RATE, and TLHGR is the TRANSIENT LINEAR HEAT GENERATION RATE. The TLHGR is specified in the CORE OPERATING LIMITS REPORT.

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C 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 are analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated are change 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 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in Specification 6.9.

MCPR Operating 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 3.3.E. The measured scram inserted times are compared with the nominal scram insertion times and the Technical Specification Scram Speeds. The appropriate operating limit is applied, as specified in the COLR.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the Safety Limit MCPR in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying the appropriate flow dependent MCPR limits presented in the COLR. The MCPR Operating Limit for a given power/flow state is the greatest value of MCPR as given by the rated conditions MCPR limit of the flow dependent MCPR limit. For automatic flow control, in addition to protecting the Safety Limit MCPR during the flow run-up event, protection is provided to prevent exceeding the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow.

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 has considerable margin. Thus, the demonstration of MCPR below this power level is 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 after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

3/4.11.D STEADY STATE LINEAR HEAT GENERATION RATE

This specification assures that the maximum LINEAR HEAT GENERATION RATE in any fuel rod is less than the design STEADY STATE LINEAR HEAT GENERATION RATE even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distributions shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating SLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that SLHGR

BASES

will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

5.3 REACTOR CORE

Fuel Assemblies

5.3.A The reactor core shall contain 724 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. The assemblies may contain water rods or a water box. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Control Rod Assemblies

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

5.4 [INTENTIONALLY LEFT BLANK]

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3. Annual Radiological Environmental Operating Report

   The Annual Radiological Environmental Operating Report covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

4. Radioactive Effluent Release Report

   The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to April 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

   Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

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:

      (1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

      (2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

      (3) The Steady State Linear Heat Generation Rate (SLHGR) for Specification 3.11.D.

      (4) The Minimum Critical Power Operating Limit (including scram insertion times) for Specification 3.11.C. This includes rated and off-rated flow conditions.

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

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

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

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

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

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

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

(7) Qualification of Exxon Nuclear Fuel for Extended Burnup: Extended Burnup Qualification of ENC 9x9 BWR Fuel, XN-NF-82-06(P)(A).

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

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

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

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

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.B Special Reports

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

6.10 [Intentionally Left Blank]

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92 (c), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

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

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

3. Involve a significant reduction in a margin of safety.

The standard reload fuel type to be used at Dresden Station is being changed to the ATRIUM-9B, an NRC approved, Siemens Power Corporation (SPC) manufactured reload BWR fuel assembly. As a result, certain items in the Technical Specifications are being revised. These changes can be classified into two categories: (a) ATRIUM-9B related, and (b) minor changes not related to the introduction of the ATRIUM-9B fuel type. Each is discussed below.

a. <u>ATRIUM-9B</u>

The fuel description in Specification 5.3.A is being modified to reflect the water box in the ATRIUM-9B fuel design and the list of references in Specification 6.9.A is modified primarily to include the use of the latest NRC approved revision to the Siemens LOCA analysis methodology which will be used to evaluate the ATRIUM-9B and other co-resident fuel types at Dresden. The listing of other Siemens methodologies has been enhanced.

b. <u>Miscellaneous changes</u>

A footnote is added to the reactor vessel water level references in the Safety Limits and Limiting Safety System Settings sections (and the associated bases are changed) to provide a consistent use of the reactor vessel reference elevation known as top of active fuel. The Bases for Specification 3/4.2 is also enhanced to provide a clearer

description of the use of the top of active fuel as a reactor vessel water level reference elevation.

A typographical error is being corrected in Bases 2.2.A.1, which refers to the power level at which the IRM system terminates the low power Control Rod Withdrawal Error (RWE) event.

The Reactivity Anomaly surveillance is modified to better describe the current use of the k_{eff} method (SPC methodology) for monitoring core reactivity while maintaining the control rod density method as an option.

The scram insertion timing terminology in Bases 3/4.3.E are modified to clarify the use of this data under Siemens' methods.

The Bases discussion of pressurization transients for the ASME over-pressurization event is modified to reflect the fact that Siemens' methodology determines the most limiting pressurization transient each fuel cycle.

The Power Distribution Limit Bases are enhanced by providing additional detail on the application of Siemens' licensing methodology.

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 10 CFR 50.92 (c), operation of Dresden Units 2 and 3 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 NRC 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 UFSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and methods. The latest NRC approved revision to the Siemens LOCA analysis methodology (Reference: ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model) will be used to evaluate the ATRIUM-9B and other co-resident fuel types. The other licensing analysis methods and models are also NRC approved. These approved methods and models are used to determine the fuel thermal limits (e.g., APLHGR, TLHGR, MCPR and LHGR). The SPC core monitoring code enables the site to monitor k_{eff} as well as control rod density to perform the reactivity anomaly surveillance. 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 reload quantities of ATRIUM 9B fuel at Dresden 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 (Reference: ANF-89-014(P)(A) Revision 1 Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9X9-IX and 9X9-9X BWR

Reload Fuel). 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 relative to the current Dresden fuel type, the Siemens manufactured 9x9-2. The exterior dimensions of the ATRIUM-9B fuel have been evaluated by ComEd; the ATRIUM-9B fuel design provides adequate clearances relative to the co-resident 9x9-2 fuel. Thus, no increased interactions with the adjacent control blade or nuclear instrumentation are created. 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 storage pool fuel preparation machine. Therefore, the probability of adverse interactions between the ATRIUM-9B fuel and components in the core and fuel handling equipment is not significantly increased.

The ATRIUM-9B design is neutronically compatible with the existing fuel types and core components in the Dresden core. SPC tests have demonstrated that the ATRIUM-9B fuel design is hydraulically compatible with the co-resident 9x9-2 fuel. The bundle pressure drop characteristics of the ATRIUM 9B bundle are similar to those of the 9x9-2 fuel design, hence core thermal-hydraulic stability characteristics are not adversely affected by the ATRIUM-9B design. Cycle stability calculations are performed by SPC. Therefore, the probability of thermal hydraulic instability is not significantly increased.

Evaluations of the Dresden Emergency Procedures and UFSAR Chapter 15 AOOs are being performed to ensure that the use of the ATRIUM-9B fuel at Dresden does not alter any assumptions previously made in evaluating the radiological consequences of an accident at Dresden Units 2 and 3. Therefore, the radiological consequences of accidents are not significantly increased.

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 proposed methodologies are administrative in nature and do not significantly affect any accident precursors or accident results; as such, the proposed change to the listing of the SPC methodologies for Dresden does not significantly increase the probability or consequences of any previously evaluated accidents.

The description of the fuel is modified to include the water box design of the NRC approved ATRIUM-9B fuel type.

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 10 CFR 50.92 (c), operation of Dresden Units 2 and 3 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. In accordance with FSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and method as have been previously performed. As stated above, the proposed changes do not permit modes of reactor operation which differ from those currently permitted; therefore, the possibility of a new or different kind of accident is not created. Plant support equipment is not affected by the proposed changes; therefore, no new failure modes are created.

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. Eight lead ATRIUM-9B assemblies were loaded into Dresden 2 during Cycle 15. 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 (Reference: ANF-89-014(P)(A) Revision 1 Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9X9-IX and 9X9-9X BWR Reload Fuel). The similarities in fuel design and operation between the ATRIUM-9B and the 9x9-2, and the previous Boiling Water Reactor experience with Siemens fuel, indicate there would be no new or different types of accidents for Dresden than have been considered for the existing fuel. Therefore, the use of ATRIUM-9B fuel at Dresden 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 10 CFR 50.92 (c), operation of Dresden Units 2 and 3 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., 10 CFR 50.46 limits). The proposed Technical Specification amendment reflects previously approved SPC methodology used to demonstrate that the existing acceptance criteria are satisfied. The revised LOCA methodology has been previously reviewed and approved by the USNRC for application to reload cores of 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. USNRC approved methods are listed in Specification 6.9.A of the Technical Specifications.

Analyses performed with USNRC-approved methodology have demonstrated that fuel design and licensing criteria will be met during normal and abnormal operating conditions. The same margins of safety will continue to be utilized by SPC (e.g., limits on peak cladding temperature, cladding oxidation, plastic strain). Therefore, there is not a significant reduction in the margin of safety.

New Fuel Design

The exterior dimensions of the ATRIUM-9B fuel assembly result in equivalent clearances relative to the co-resident 9x9-2 fuel. Thus, no increased interactions with the adjacent control blade and nuclear instrumentation are created. The change does not adversely impact equipment important to safety; therefore the margin of safety is not significantly reduced.

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 STATEMENT APPLICABILITY REVIEW

Commonwealth Edison 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 a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts and no significant changes in the types of any effluents that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT E

IDENTIFICATION AND DESCRIPTION OF ANY IRREVERSIBLE CONSEQUENCES

Commonwealth Edison has evaluated the proposed amendment and determined that the proposed changes do not involve a significant increase in the amounts, or types, of any effluents or radiation that may be released offsite. The revised LOCA analysis demonstrates that all consequences of the postulated event are within the required acceptance criteria and limits. Therefore, no irreversible consequences will result because of the requested changes.