B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with Reference 1, Appendix 1C, Criteria 27, 29, and 30. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SHUTDOWN MARGIN requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of Limiting Condition for Operation (LCO) 3.1.6. When the unit is in the shutdown and refueling MODEs, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE SAFETY ANALYSIS

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Reference 1, Section 3.4) establishes a SDM that ensures specified acceptable fuel design limits (SAFDLs) are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limit is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that SAFDLs are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio [DNBR], fuel centerline temperature limit AOOs, and an acceptable energy deposition for the CEA ejection accident [Reference 1, Chapter 14]); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a main steam line break (MSLB) or an Excess Load event (with failure of an MSIV to close), as described in the accident analysis (Reference 1, Chapter 14). The increased steam flow causes an increased energy removal from the affected steam generator, and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient (MTC), this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of the event decreases. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line outside containment, initiated at the end of core life. Following the MSLB or Excess Load event, a post-trip return to power may occur; however, no fuel damage occurs as a result of the post-trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1. The limiting Excess Load event with respect to potential return-to-power after reactor trip is the opening of all steam dump and bypass valves at full power with failure of an MSIV to close.

In addition to the limiting MSLB transient, the SDM requirement for MODEs 3 and 4 must also protect against an uncontrolled CEA withdrawal from a hot zero power or low power condition, and a CEA ejection.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of CEAs from hot zero power or low power conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time-dependent redistribution of core power.

The uncontrolled CEA withdrawal transient is terminated by the Variable High Power Trip. In all cases, power level, RCS pressure, linear heat rate (LHR), and the DNBR do not exceed allowable limits.

SHUTDOWN MARGIN satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The MSLB (or the Excess Load event) and the boron dilution accidents (Reference 1, Chapter 14) are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents (or the Excess Load event), if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed the acceptance criteria given in Reference 1, Chapter 14. For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable. Because both initial RCS level and the dilution flow rate also significantly impact the boron dilution event in MODE 5 with pressurizer level < 90 inches from the bottom of the pressurizer, the LCO also includes limits for these parameters during these conditions.

SHUTDOWN MARGIN is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEA) in MODEs 1 and 2 and through the soluble boron concentration in all other MODEs.

APPLICABILITY

In MODEs 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODEs 1 and 2, SDM is ensured by complying with LCOs 3.1.5 and 3.1.6. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1.

ACTIONS

A.1, A.2, and A.3

With non-borated water sources of > 88 gpm available, while the unit is in MODE 5 with the pressurizer level < 90 inches, the consequences of a boron dilution event may exceed the analysis results. Therefore, action must be initiated immediately to reduce the potential for such an event. To accomplish this, Required Action A.1 requires immediate suspension of positive reactivity additions. However, since Required Action A.1 only reduces the potential for the event and does not eliminate it, immediate action must also be initiated to increase the SDM to compensate for the non-borated water sources (Required Action A.2). Finally, Required Action A.3 requires periodic verification, once per 12 hours, that the SDM increase is maintained sufficient to compensate for the additional sources of non-borated water. Required Action A.1 is modified by a Note indicating that the suspension of positive reactivity additions is not required if SDM has been sufficiently increased to compensate for the additional sources of non-borated water. The immediate Completion Time reflects the urgency of the corrective actions. The periodic Completion Time of 12 hours is considered reasonable, based on other administrative controls available and operating experience.

B.1 and B.2

With the RCS level at or below the bottom of the hot leg nozzles, while the unit is in MODE 5 with the pressurizer level < 90 inches, the consequences of a boron dilution event may exceed the analysis results. Therefore, action must be initiated immediately to reduce the potential for such an event. To accomplish this, Required Action B.1 requires immediate suspension of operations involving positive reactivity additions that could result in loss of the required SDM. Suspending positive reactivity additions that could result in failure to meet the minimum SDM limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have boron concentration greater than that required in the RCS for the minimum SDM. This may result in an overall reduction in RCS boron concentration, but provides an acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of the required SDM. However, since Required Action B.1 only reduces the potential for the event and does not eliminate it, immediate action must also be initiated to increase the RCS level to above the bottom of the hot leg nozzles (Required Action B.2). The immediate Completion Time reflects the urgency of the corrective actions.

C.1

If the SDM requirements are not met for reasons other than addressed in Condition A or B, boration must be initiated promptly. A Completion Time of immediately is required to meet the assumptions of the safety analysis. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the refueling water tank. The operator should borate with the best source available for the plant conditions. However, as a minimum, the boration flow rate shall be ≥ 40 gpm and the boron concentration shall be ≥ 2300 ppm boric acid solution or equivalent.

Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 40 gpm from the boric acid storage tank, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 15 minutes. If an inverse boron worth of 100 ppm/% $\Delta k/k$ is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 40 gpm and 100 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SHUTDOWN MARGIN is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CEA positions;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient.

Using the isothermal temperature coefficient accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.1.2 and SR 3.1.1.3

These Surveillance Requirements (SRs) periodically verify the significant assumptions of a boron dilution event are maintained. A non-borated water source of ≤ 88 gpm allows for only one charging pump to be capable of injection during these conditions since each charging pump is capable of an injection rate of 46 gpm. Each SR is modified by a Note indicating that it is only required when the unit is in MODE 5 with the pressurizer level < 90 inches. Since the applicable conditions for the SR may be attained while already in MODE 5, each SR is provided with a Frequency of once within 1 hour after achieving MODE 5 with pressurizer level < 90 inches. This provides a short period of time to verify compliance after the conditions are attained. Additionally, each SR must be completed once each 12 hours after the initial verification. The Frequency of 12 hours is considered reasonable, in view of other administrative controls available and operating experience.

REFERENCES

1. Updated Final Safety Analysis Report (UFSAR)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to Reference 1, Appendix 1C, Criteria 27, 29, and 30, reactivity shall be controllable, such that, subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and AOOs. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CEA worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1) in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the RCS versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as CEA height, temperature, pressure, and power) provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in the fuel remaining from the previous cycle provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at hot full power, the excess positive reactivity is compensated by burnable absorbers (if any), CEAs, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RATED THERMAL POWER (RTP). Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Most accident evaluations (Reference 1, Section 14.1) are, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CEA withdrawal accidents or CEA ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning-of-cycle (BOC) do not agree, the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the CEAs in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

The reactivity balance satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of $\pm$ 1% $\Delta k/k$ has been established, based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should, therefore, be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state

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RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached.

These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODE 1 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur. This Specification does not apply in MODEs 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1) ensure that fuel movements are performed within the bounds of the safety analysis. A SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, or CEA replacement, or shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and to complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, the boron letdown curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, they must be defined.

The required Completion Time of 7 days is adequate for preparing whatever operating restrictions or SRs may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable, including CEA position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The SR is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC and every 31 days after 60 effective full power days (EFPD). The SR is modified by two Notes. The Note in the SR column indicates that the normalization of predicted core reactivity to the measured value may take place within the first 60 EFPD after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD following the initial 60 EFPD, after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g., quadrant power tilt ratio, etc.) for prompt indication of an anomaly. The Frequency Note, "only required after 60 EFPD after each fuel loading," is added to the Frequency column to allow this.

REFERENCES

1. UFSAR

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

The MTC relates a change in core reactivity to a change in reactor coolant temperature. A positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature. The reactor is designed to operate with a negative MTC over a large range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

Moderator temperature coefficient values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Reload cores are designed so that the MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons (burnable poison) to yield an MTC at the BOC within the range analyzed in the plant accident analysis. The end-of-cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Reference 1, Section 14.2.2); and

b. The MTC must be such that inherently stable power operations result during normal operation and during accidents, such as overheating and overcooling events.

Reference 1, Section 14.1.2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. Moderator temperature coefficient is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst-case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding.

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the CEA withdrawal and CEA ejection transients from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to a positive MTC is a CEA ejection accident from full power (Reference 1, Section 14.13).

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction combined with the large negative MTC may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power is produced with all CEAs inserted, except the most reactive one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

Moderator temperature coefficient values are bounded in reload safety evaluations assuming steady state conditions at BOC, peak RCS boron, and EOC. A 2/3 core burnup MTC measurement is conducted and the measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The MTC satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

Limiting Condition for Operation 3.1.3 requires the MTC to be within specified limits of the Core Operating Limits Report (COLR), with the maximum positive limit specified in Figure 3.1.3-1, to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

Moderator temperature coefficient is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement. The surveillance checks at BOC and 2/3 core burnup provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup accidents, such as the uncontrolled CEA or group withdrawal, will not violate the assumptions of the accident analysis. In MODEs 3, 4, 5, and 6, this LCO is not applicable, since no DBAs using the MTC as an analysis assumption are initiated from these MODEs. However, the variation of the MTC, with temperature in MODEs 3, 4, and 5 for DBAs initiated in MODEs 1 and 2, is accounted for in the accident analysis. The variation of the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC and 2/3 core burnup measurements are used for normalization.

ACTIONS

A.1

Moderator temperature coefficient is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 and SR 3.1.3.2

The SRs for measurement of the MTC at the beginning and 2/3 core burnup of each fuel cycle provide for confirmation of the limiting MTC values. The MTC varies with boron concentration during fuel cycle operation. The MTC becomes more negative as the RCS boron concentration is reduced. The requirement for measurement prior to entering MODE 1 after each fuel loading satisfies the confirmatory check on the most positive (least negative) MTC value. The requirement for measurement, within 7 EFPD of initially reaching an equilibrium condition with THERMAL POWER ≥ 90% RTP, and within 7 EFPD of reaching 2/3 core burnup, satisfies the confirmatory check of the most negative MTC value. The 2/3 core burnup measurement is performed at any THERMAL POWER, so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. Moderator temperature coefficient values may be extrapolated and compensated to permit direct comparison to the specified MTC limits.

Surveillance Requirement 3.1.3.2 is modified by a Note, which indicates that if the extrapolated MTC is more negative than the EOC COLR limit, the SR may be repeated, and that shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. An engineering evaluation is

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performed if the extrapolated value of MTC exceeds the Specification limits.

REFERENCES

1. UFSAR

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Element Assembly (CEA) Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating CEAs is an initial assumption in all safety analyses that assume CEA insertion upon reactor trip.

The applicable criteria for these reactivity and power distribution design requirements are found in Reference 1, Appendix 1C, Criteria 6, 27, 29, and 30, and Reference 2.

Mechanical or electrical failures may cause a CEA to become inoperable or to become misaligned from its group. Control element assembly inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CEA worth for reactor shutdown. Therefore, CEA alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on CEA alignment and OPERABILITY have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Control element assemblies are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately 3/4-inch) at a time.

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEA groups do not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the Plant Computer CEA Position Indication System and the Reed Switch Position Indication System.

The Plant Computer CEA Position Indication System counts the commands sent to the CEA gripper coils from the CEDM Control System that moves the CEAs. There is a one step counter for each CEA. Individual CEAs in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. Plant Computer CEA Position Indication System is considered highly precise (± 1 step or ± 3/4-inch). If a CEA does not move one step for each command signal, the step counter will still count the command and incorrectly reflect the position of the CEA.

The Reed Switch Position Indication System provides a highly accurate indication of actual CEA position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of reed switches spaced along a tube with a center-to-center distance of 1.5 inches, which is two steps. To increase the reliability of the system, there are redundant reed switches at each position.

APPLICABLE SAFETY ANALYSES

Control element assembly misalignment accidents are analyzed in the safety analysis (Reference 1, Sections 14.2, 14.11, and 14.13). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdraws, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDM Control System, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper. A dropped CEA could be caused by an electrical failure in the CEA coil power programmers.

The acceptance criteria for addressing CEA inoperability/ misalignment are that:

a. There shall be no violations of:

   1. SAFDLs, or

   2. RCS pressure boundary integrity; and

b. The core must remain subcritical after accidents or transients.

Two types of misalignment are distinguished in the safety analysis (Reference 1, Appendix 1C). The first type of misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. The second type of misalignment occurs when one CEA drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return toward the original power, due to positive reactivity feedback from the negative MTC. Increased peaking during the power increase may result in excessive local LHRs (Reference 1, Section 14.14).

None of the above CEA misoperations will result in an automatic reactor trip. In the case of the full-length CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

The results of the CEA misoperation analysis show that, during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or RCS pressure occur.

Control element assembly alignment satisfies 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The limits on shutdown and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the CEA banks maintain the correct power distribution and CEA alignment.

The requirement is to maintain the CEA alignment to within 7.5 inches between any CEA and its group.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CEA OPERABILITY and alignment are applicable in MODEs 1 and 2 because these are the only MODEs in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODEs 3, 4, 5, and 6, the alignment limits do not apply because the CEAs are bottomed, and the reactor is shut down and not producing fission power. In the shutdown MODEs, the OPERABILITY of the shutdown and regulating CEAs has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1 for SDM in MODEs 3, 4, and 5, and LCO 3.9.1 for boron concentration requirements during refueling.

ACTIONS

A.1 and B.1

A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary.

If one or more regulating or shutdown CEAs are misaligned by > 7.5 inches and ≤ 15 inches but trippable, or one CEA is misaligned by > 15 inches but trippable, continued operation in MODEs 1 and 2 may continue, provided CEA alignment is restored within 1 hour for CEAs misaligned ≤ 15 inches and within the time specified in the COLR for CEAs misaligned > 15 inches. (The maximum time provided in the COLR is 2 hours.)

Regulating and shutdown CEA alignment is restored by either aligning the misaligned CEA(s) to within 7.5 inches of its group or aligning the misaligned CEAs group to within 7.5 inches of the misaligned CEA.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Restoring CEA alignment ensures acceptable power distributions are maintained. For small misalignments (≤ 15 inches) of the CEAs, there is:

a. A small effect on the time-dependent, long-term power distributions relative to those used in generating LCOs and limiting safety system settings setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected CEA worth used in the accident analysis.

With a large CEA misalignment (> 15 inches), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time-dependent, long-term power distributions relative to those used in generating LCOs and limiting safety system settings setpoints.

The effect on the available SDM and the ejected CEA worth used in the accident analysis remains small.

Therefore, this condition is limited to a single CEA misalignment, while still allowing time for recovery.

In both cases, the allowed time period is sufficient to:

a. Identify cause of a misaligned CEA;

b. Take appropriate corrective action to realign the CEAs; and

c. Minimize the effects of xenon redistribution.

If a CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCOs 3.1.5 and 3.1.6 does not ensure that adequate SDM exists. Condition F must be entered.

C.1 and C.2

If any CEA is not restored to within its alignment limits within the Completion Time provided in Required Action A.1 or B.1, an additional 2 hours is allowed to restore CEA alignment, provided THERMAL POWER is reduced ≤ 70% RTP. Prompt action must be taken to reduce THERMAL POWER, and the reduction must be completed within 1 hour. Reducing THERMAL POWER ensures acceptable power distributions are maintained during the additional time provided to restore alignment. The Completion Times are acceptable based on the reasons provided in the Bases for Required Actions A.1 and B.1.

D.1, D.2.1, and D.2.2

The CEA motion inhibit permits CEA motion within the requirements of LCO 3.1.6, and prevents regulating CEAs from being misaligned from other CEAs in the group.

Performing SR 3.1.4.1 within 1 hour and every 4 hours thereafter is considered acceptable, in view of other information continuously available to the operator in the Control Room.

With the CEA motion inhibit inoperable, a Completion Time of 6 hours is allowed for restoring the CEA motion inhibit to OPERABLE status, or fully withdrawing the CEAs in groups 3 and 4, and withdrawing all CEAs in group 5 to < 5% insertion.

Withdrawal of the CEAs to the positions required in Required Action D.2.2 provides additional assurance that core perturbations in local burnup, peaking factors, and SDM will not be more adverse than the Conditions assumed in the safety analyses and LCO setpoint determination (Reference 1, Chapter 14).

The 6-hour Completion Time takes into account Required Action D.1, the protection afforded by the CEA deviation circuits, and other information continuously available to the operator in the Control Room, so that during actual CEA motion, deviations can be detected.

Required Action D.2.2 is modified by a Note indicating that performing this Required Action is not required when in conflict with Required Actions A.1, B.1, C.2, or E.1.

E.1

When the CEA deviation circuit is inoperable, performing SR 3.1.4.1 within 1 hour and every 4 hours thereafter ensures improper CEA alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the Control Room, so that during CEA movement, deviations can be detected, and the protection provided by the CEA inhibit and deviation circuit is not required.

F.1

If any Required Action and associated Completion Time of Condition C, Condition D, or Condition E is not met, one or more regulating or shutdown CEAs are untrippable, two or more CEAs are misaligned by > 15 inches, the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside the MODE of applicability. Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > 15 inches, or one or more CEAs untrippable. This is because these cases could result in a loss of SDM and power distribution and a loss of safety function, respectively.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification is required that individual CEA positions are within 7.5 inches (indicated reed switch positions) of all other CEAs in the group within 1 hour of any CEA movement of > 7.5 inches. The CEA position verification after each movement of > 7.5 inches ensure that the CEAs in that group are properly aligned at the time when CEA misalignments are most likely to have occurred. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.4.2

Demonstrating the CEA motion inhibit OPERABLE verifies that the CEA motion inhibit is functional, even if it is not regularly operated. The verification shall ensure that the motion inhibit circuit maintains the CEA group overlap and sequencing requirements of LCO 3.1.6, and prevents any regulating CEA from being misaligned from all other CEAs in its group by > 7.5 inches (indicated position). The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.4.3

Demonstrating the CEA deviation circuit is OPERABLE verifies the circuit is functional. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.4.4

Verifying each CEA is trippable would require that each CEA be tripped. In MODEs 1 and 2, tripping each CEA would result in radial or axial power tilts or oscillations. Therefore, individual CEAs are exercised to provide increased confidence that all CEAs continue to be trippable, even if they are not regularly tripped. A movement of 7.5 inches is adequate to demonstrate motion without exceeding the alignment limit when only one CEA is being moved. For the purposes of performing the CEA operability test, if the CEA has an inoperable position indicator channel, the alternate indication system (pulse counter or voltage dividing network) will be used to monitor position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. Between required performances of SR 3.1.4.4, if a CEA(s)is discovered to be immovable, but remains trippable and aligned, the CEA is considered to be OPERABLE. At any time, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of the CEA(s) must be made, and appropriate action taken.

SR 3.1.4.5

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position over the entire length of the CEA's travel. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specification tests at least once per refueling interval with applicable extensions. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.4.6

Verification of CEA drop times determined that the maximum CEA drop time permitted is consistent with the assumed drop time used in that safety analysis (Reference 1, Chapter 14). Control element assembly drop time is measured from the time when electrical power is interrupted to the CEDM until the CEA reaches its 90% insertion position, from a fully withdrawn position, with $T_{ave} \geq 515°F$ and all reactor coolant pumps operating. Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures that reactor internals and CEDM will not interfere with CEA motion or drop time, and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality, based on the need to perform this SR under the conditions that apply during a unit outage and because of the potential for an unplanned unit transient if the SR were performed with the reactor at power.

REFERENCES

1. UFSAR

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown CEAs are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected CEA worth, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are in Reference 1, Appendix 1C, Criteria 6, 27, 28, 29, and 30, and Reference 2. Limits on shutdown CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected CEA worth, and SDM limits are preserved.

The shutdown CEAs are arranged into groups that are radially symmetric. Therefore, movement of the shutdown CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The design calculations are performed with the assumption that the shutdown CEAs are withdrawn prior to the regulating CEAs. The shutdown CEAs can be fully withdrawn without the core going critical. The shutdown CEAs are controlled manually by the Control Room operator. During normal unit operation, the shutdown CEAs are fully withdrawn. The shutdown CEAs must be completely withdrawn from the core prior to withdrawing any regulating CEAs during an approach to criticality. The shutdown CEAs are left in this position until the reactor is shut down. They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown CEAs are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a CEA ejection accident are limited to acceptable limits.

Control element assemblies are considered fully withdrawn at 129 inches.

On a reactor trip, all CEAs (shutdown and regulating), except the most reactive CEA, are assumed to insert into the core. The shutdown and regulating CEAs shall be at or above their insertion limits and available to insert the required amount of negative reactivity on a reactor trip signal. The regulating CEAs may be partially inserted in the core as allowed by LCO 3.1.6. The shutdown CEA insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1) following a reactor trip from full power. The combination of regulating CEAs and shutdown CEAs (less the most reactive CEA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Reference 1, Sections 3.2 and 3.4). The shutdown CEA insertion limit also limits the reactivity worth of an ejected shutdown CEA.

The acceptance criteria for addressing shutdown CEA, as well as regulating CEA insertion limits and inoperability or misalignment, are that:

a. There be no violation of:

   1. SAFDLs, or

   2. RCS pressure boundary damage; and

b. The core remains subcritical after accident transients.

As such, the shutdown CEA insertion limits affect safety analyses involving core reactivity, ejected CEA worth, and SDM (Reference 1, Section 14.1.2).

BASES

The shutdown CEA insertion limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The shutdown CEAs must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

APPLICABILITY

The shutdown CEAs must be within their insertion limits, with the reactor in MODEs 1 and 2. The Applicability in MODE 2 begins anytime any regulating CEA is not fully inserted. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 3, 4, 5, or 6, the shutdown CEAs are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODEs 3, 4, and 5. Limiting Condition for Operation 3.9.1 ensures adequate SDM in MODE 6.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the shutdown CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1

When one shutdown CEA is withdrawn ≥ 121.5 inches and < 129 inches, the accumulated times the shutdown CEAs have been withdrawn within this range must be verified. The Completion Time for this action is once within 4 hours and 24 hours thereafter. Operation is allowed for 7 consecutive days and a total of 14 days per 365 days. The peaking factors may not be outside required limits when one shutdown CEA is misaligned; therefore, continued operation is allowed. Since the power distribution limits are being maintained via the LCOs of Technical Specification Section 3.2, any out-of-limit peaking factor conditions will require entry into the Actions of the appropriate Section 3.2 LCO(s). The limits on consecutive days and total days in this condition reflect that the core may be approaching the acceptable limits placed on operation with flux patterns outside those assumed in the long-term burnup assumptions. Therefore, operation in this condition cannot continue and the CEA is required to be restored per Action B. The accumulated times are required to be verified once within 4 hours to determine which accumulated time limit is more limiting. The periodic Completion Time of 24 hours after the initial completion within 4 hours is adequate to ensure that the accumulated time limits are not exceeded.

B.1

Prior to entering this condition, the shutdown CEAs were fully withdrawn or all but one shutdown CEA was withdrawn $\geq$ 129 inches. If one shutdown CEA is withdrawn $\geq$ 121.5 inches and $<$ 129 inches for $>$ 7 days per occurrence or $>$ 14 days per 365 days, or one shutdown CEA withdrawn $<$ 121.5 inches, or two or more shutdown CEAs withdrawn $<$ 129 inches, the out-of-limit CEAs must be restored to within limits within 2 hours. The Completion Time of 2 hours reflects that the power distribution limits may be outside required limits and that the core may be approaching the acceptable limits placed on operation within flux patterns outside those assumed in the long-term burnup assumptions.

The CEA(s) must be restored to within limits within 2 hours. The 2-hour total Completion Time allows the operator adequate time to adjust the CEA(s) in an orderly manner.

C.1

When Required Action A.1 or B.1 cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown CEAs are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown CEAs will be available to shut down the

BASES

reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown CEAs are withdrawn before the regulating CEAs are withdrawn during a unit startup.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants"

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Regulating Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating CEAs are initial assumptions in all safety analyses that assume CEA insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are Reference 1, Appendix 1C, Criteria 27, 29, 30, and 31, and Reference 2.

Limits on regulating CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected CEA worth, reactivity insertion rate, and SDM limits are preserved.

The regulating CEA groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between CEA worth and CEA position (integral CEA worth). The regulating CEA groups are withdrawn and operate in a predetermined sequence. The group sequence and overlap limits are specified in the COLR. Regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches.

The regulating CEAs are used for precise reactivity control of the reactor. The positions of the regulating CEAs are manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain SAFDLs, including limits that preserve the criteria specified in Reference 2. Together, LCOs 3.1.6, 3.2.4, and LCO 3.2.5 provide limits on control component operation and on monitored process variables to ensure the core operates within the LHR (LCO 3.2.1); and Total Integrated Radial Peaking Factor (F_r^T) (LCO 3.2.3) limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling

System analysis. Operation within the F_r^T limit given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR and F_r^T limits, certain reactivity limits are preserved by regulating CEA insertion limits. The regulating CEA insertion limits also restrict the ejected CEA worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODEs 1 and 2.

The regulating CEA insertion and alignment limits are process variables that together characterize and control the three-dimensional power distribution of the reactor core. Additionally, the regulating bank insertion limits control the reactivity that could be added in the event of a CEA ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protective System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and AOOs (Condition II). The acceptance criteria for the regulating CEA insertion, ASI, F_r^T, LHR, and AZIMUTHAL POWER TILT (T_q) LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F (Reference 2);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the energy input to the fuel must not exceed accepted limits (Reference 1, Section 14.3); and

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, Reference 1, Appendix 1C, Criterion 29.

Regulating CEA position, ASI, F_r^T, LHR, and T_q are process variables that together characterize and control the three-dimensional power distribution of the reactor core.

Fuel cladding damage does not normally occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result if an accident or AOO occurs with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown CEA insertion limits, so that the allowable inserted worth of the CEAs is such that sufficient reactivity is available to shut down the reactor to hot zero power. SHUTDOWN MARGIN assumes the maximum worth CEA remains fully withdrawn upon trip (Reference 1, Section 3.4).

The most limiting SDM requirements for MODEs 1 and 2 conditions at BOC are determined by the requirements of several transients, e.g., Loss of Flow, Seized Rotor, Boron Dilution, etc. However, the most limiting SDM requirements for MODEs 1 and 2 at EOC come from the SLB and Excess Load transients. The requirements of the SLB and Excess Load events at EOC for both the full power and no load conditions are significantly larger than those of any other event at that time in cycle.

To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle a are sufficient since the differences between available SDMs and the limiting SDM requirements are the smallest at these times in a cycle. The measurement of CEA bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCOs 3.1.5 and 3.1.6 provides assurance that the available SDM at any time in a cycle will exceed the limiting SDM requirements at that time in a cycle.

Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected CEA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected CEA worths.

The regulating and shutdown CEA insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected CEA worth, and power distribution peaking factors are preserved (Reference 1, Section 3.4).

The regulating CEA insertion limits satisfy 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

The limits on regulating CEAs sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected CEA worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating CEA motion.

The power-dependent insertion limit (PDIL) alarm circuit is required to be OPERABLE for notification that the CEAs are outside the required insertion limits. The PDIL alarm circuit required to be OPERABLE receives its signal from the reed switch position indication system. When the PDIL alarm circuit is inoperable, the verification of CEA positions is increased to ensure improper CEA alignment is identified before unacceptable flux distribution occurs.

APPLICABILITY

The regulating CEA sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODEs 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected CEA worth, SDM, and reactivity rate insertion assumptions. Applicability in MODEs 3, 4, and 5 is not required, since neither the power distribution nor ejected CEA worth assumptions would be exceeded in these MODEs. SHUTDOWN MARGIN is preserved in MODEs 3, 4, and 5 by adjustments to the soluble boron concentration.

This LCO has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.4. This SR verifies the freedom of the CEAs to move, and requires the regulating CEAs to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1 and A.2

Operation beyond the transient insertion limit may result in a loss of SDM and excessive peaking factors. The transient insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the CEAs in response to changing plant conditions. When the regulating groups are inserted beyond the transient insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual CEA insertion limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1 and B.2

If the CEAs are inserted between the long-term steady state insertion limits and the transient insertion limits for intervals > 4 hours per 24 hour period, and the short-term steady state insertions are exceeded, peaking factors can develop that are of immediate concern (Reference 1, Chapter 14).

Verifying the short-term steady state insertion limits are not exceeded ensures that the peaking factors that do develop are within those allowed for continued operation. Fifteen minutes provides adequate time for the operator to verify if the short-term steady state insertion limits are exceeded.

Experience has shown that rapid power increases in areas of the core, in which the flux has been depressed, can result in fuel damage, as the LHR in those areas rapidly increases. Restricting the rate of THERMAL POWER increases to $\leq 5\%$ RTP per hour, following CEA insertion beyond the long-term steady-state insertion limits, ensures the power transients experienced by the fuel will not result in fuel failure.

C.1

With the regulating CEAs inserted between the long-term steady state insertion limit and the transient insertion limit, and with the core approaching the 5 EFPD per 30 EFPD or 14 EFPD per 365 EFPD limits, the CEAs must be returned to within the long-term steady state insertion limits, or the core must be placed in a condition in which the abnormal fuel burnup cannot continue. A Completion Time of 2 hours is allotted to return the CEAs to within the long-term steady state insertion limits.

The required Completion Time of 2 hours from initial discovery of a regulating CEA group outside the limits until its restoration to within the long-term steady state limits, shown on the figures in the COLR, allows sufficient time for borated water to enter the RCS from the chemical addition and makeup systems, and to cause the regulating CEAs to withdraw to the acceptable region. It is reasonable to continue operation for 2 hours after it is discovered that the 5-day or 14-day EFPD limit has been exceeded. This Completion Time is based on limiting the potential xenon redistribution, the low probability of an accident, and the steps required to complete the action.

D.1

When the PDIL alarm circuit is inoperable, performing SR 3.1.6.1 within 1 hour and once per 4 hours thereafter

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ensures improper CEA alignments are identified before unacceptable flux distributions occur.

E.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating CEA group position is sufficient to detect CEA positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.6.2

Verification of the accumulated time of CEA group insertion between the long-term steady state insertion limits and the transient insertion limits ensures the cumulative time limits are not exceeded. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.6.3

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR
2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" 10 CFR 50.46

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Special Test Exception (STE)-SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The primary purpose of the SDM STE is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are constructed to determine the CEA worth.

Reference 1, Appendix B, Section XI requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and AOOs must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the Nuclear Regulatory Commission, for the purpose of conducting tests and experiments, are specified in Reference 1, 10 CFR 50.59.

The key objectives of a test program (Reference 2) are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Reference 3, Section 13.4).

PHYSICS TESTS' procedures are written and approved in accordance with an established process. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are independently reviewed prior to continued power escalation and long- term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because adequate limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 2 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in the UFSAR Reference 3, Section 13.4. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the LHR remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

a. LCO 3.1.1; and

b. LCO 3.1.6.

Therefore, this LCO places limits on the minimum amount of CEA worth required to be available for reactivity control when CEA worth measurements are performed.

The individual LCOs cited above govern SDM CEA group height, insertion, and alignment. Additionally, the LCOs governing RCS flow, reactor inlet temperature, and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the LOCA are specified in Reference 2, 10 CFR 50.46. The criteria for the loss of forced reactor coolant flow accident are specified in Reference 3, Chapter 14. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

Surveillance tests are conducted as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

Requiring that shutdown reactivity equivalent to at least the highest estimated CEA worth (of those CEAs actually withdrawn) be available for trip insertion from the OPERABLE CEA provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident, a stuck CEA. When LCO 3.1.1 is suspended, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth CEA was stuck out and calculational uncertainties or the estimated highest CEA worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because SAFDLs are still met. The risk of experiencing a stuck CEA and subsequent criticality is reduced during this PHYSICS TESTS exception by the Surveillance Requirements; and by ensuring that shutdown reactivity is available, equivalent to the reactivity worth of the estimated highest worth withdrawn CEA (Reference 3, Chapter 3).

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total integrated radial peaking factor, T_q and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shut down of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with STE LCOs is optional and, therefore, no criteria of 10 CFR

50.36(c)(2)(ii) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. The STE is required to permit the periodic verification of the actual versus predicted worth of the regulating and shutdown CEAs. The SDM requirements of LCO 3.1.1, the shutdown CEA insertion limits of LCO 3.1.5, and the regulating CEA insertion limits of LCO 3.1.6 may be suspended.

APPLICABILITY

This LCO is applicable in MODEs 2 and 3. Although CEA worth testing is conducted in MODE 2, sufficient negative reactivity is inserted during the performance of these tests to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the STE allows limited operation to 6 consecutive hours in MODE 3, as indicated by the Note, without having to borate to meet the SDM requirements of LCO 3.1.1.

ACTIONS

A.1

With any CEA not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all CEAs inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth CEA, restoration of the minimum SDM requirements must be accomplished by increasing the RCS boron concentration. The boration flow rate shall be $\geq$ 40 gpm and the boron concentration shall be $\geq$ 2300 ppm boric acid solution or equivalent. The required Completion Time of immediately is required to meet the assumptions of the safety analysis. It is assumed that boration will be continued until the SDM requirements are met.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification of the position of each partially or fully withdrawn full-length or part-length CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.7.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The Frequency ensures that the CEAs are OPERABLE prior to reducing SDM to less than the limits of LCO 3.1.1.

The SR is modified by a Note that allows the SR to not be performed during initial power escalation following a refueling outage if SR 3.1.4.6 has been met during that refueling outage. This allows the CEA drop time test, which also proves the CEAs are trippable, to be credited for this SR.

REFERENCES

1. 10 CFR Part 50
2. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978
3. UFSAR

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Special Test Exceptions (STE)-MODEs 1 and 2

BASES

BACKGROUND

The primary purpose of these MODEs 1 and 2 STEs is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.

Reference 1, Appendix B, Section XI requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and AOOs must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the Nuclear Regulatory Commission, for the purpose of conducting tests and experiments, are specified in Reference 1, 10 CFR 50.59.

The key objectives of a test program (Reference 2) are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Reference 3, Section 13.4).

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long-term power operation.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during a PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 3, Section 13.4 defines the requirements for initial testing of the facility, including PHYSICS TESTS. Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCO must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the LHR remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended: LCO 3.1.3; LCO 3.1.4; LCO 3.1.5; LCO 3.1.6; LCO 3.2.3; and LCO 3.2.4.

The safety analysis (Reference 3, Section 13.4) places limits on allowable THERMAL POWER during PHYSICS TESTS and requires the LHR and the DNB parameter to be maintained within limits.

The individual LCOs governing CEA group height, insertion and alignment, ASI, F_r^T, and T_q preserve the LHR limits. Additionally, the LCOs governing RCS flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the LOCA are specified in Reference 1, 10 CFR 50.46. The criteria for the loss of forced reactor coolant flow accident are specified in Reference 1, 10 CFR 50.46. Operation within the LHR limit preserves the LOCA criteria; operation within the DNB parameter limits preserves the loss of flow criteria.

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these SRs allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are F_r^T, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shut down of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with STE LCOs is optional and, therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of PHYSICS TESTS, such as those required to:

a. Measure CEA worth;

b. Determine the reactor stability index and damping factor under xenon oscillation conditions;

c. Determine power distributions for nonnormal CEA configurations;

d. Measure rod shadowing factors; and

e. Measure temperature and power coefficients.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS, provided THERMAL POWER is restricted to test power plateau, which shall not exceed 85% RTP.

APPLICABILITY

This LCO is applicable in MODEs 1 and 2 because the reactor must be critical at various THERMAL POWER levels to perform the PHYSICS TESTS described in the LCO section. Limiting the test power plateau to < 85% RTP ensures that LHRs are maintained within acceptable limits.

ACTIONS

A.1

If THERMAL POWER exceeds the test power plateau, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15-minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1 and B.2

If Required Action A.1 cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour, and the reactor must be brought to MODE 3. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal CEA configuration back to within the limits of LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6. Bringing the reactor to MODE 3 within 6 hours increases thermal margin and is consistent with the Required Actions of the power distribution LCOs. The required Completion Time of 6 hours is adequate for performing a controlled shutdown from full power conditions in an orderly manner and without challenging plant systems, and is consistent with power distribution LCO Completion Times.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.1 Verifying that THERMAL POWER is equal to or less than that allowed by the test power plateau, as specified in the PHYSICS TESTS procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. 10 CFR Part 50 2. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978 3. UFSAR