Request for Technical Specifications Change, Transition to General Electric Fuel

ENCLOSURE ONE

Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3

Attachment A
Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3
DESCRIPTION AND SAFETY ANALYSIS
FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit", Commonwealth Edison (ComEd) Company is requesting changes to various Technical Specifications (TS) for Dresden Nuclear Power Station (DNPS) Units 2 and 3 to support a change in fuel vendors from Siemens Power Corporation (SPC) to General Electric (GE) and a transition to the use of GE 14 fuel. In addition, certain poposed changes are requested to improve operational flexibility. The proposed changes affect both our Current Technical Specifications (CTS) and our proposed conversion to Improved Technical Specifications (ITS), described in Reference I.1, which is currently being reviewed by the NRC. These changes, if approved, will be implemented during the next refueling outages at DNPS Units 2 and 3, which are scheduled for October 2001, and September 2002, respectively. The proposed changes include the following:

- Revised thermal limit descriptions to reflect the GE approach to calculating and monitoring these limits.
- Revised control rod scram times to reflect the GE approach to specifying these times. In addition, the CTS control rod operability and scram timing requirements are revised to adopt the ITS approach, which limits the number of control rods with slow scram times, instead of limiting the average control rod scram time. This is necessary to ensure that the cycle-specific core reload analyses are consistent with the approved version of the TS (i.e., CTS or ITS) in effect at the time of implementation of the changes.
- Revised power level at which the Rod Worth Minimizer (RWM) is required to be operable. This revision provides operational flexibility and makes the DNPS TS consistent with other ComEd Boiling Water Reactors (BWRs) and with the proposed ITS conversion.
- Revised references to include GE methods in the Core Operating Limits Report (COLR) description of approved analytical methods.

The DNPS units are expected to operate with reactor cores containing both GE and SPC fuel for several operating cycles. Because of this, the proposed TS changes do not remove requirements related to the use of SPC fuel. These requirements will be deleted in a future license amendment request.

As ComEd's fuel vendor, GE will be performing Critical Power Ratio (CPR) calculations to determine safety limits for the DNPS core reloads. These calculations will apply GE methodology to the remaining SPC fuel. As documented in Reference I.2, GE has requested NRC approval for this application of GE methodology to SPC fuel.

The proposed TS changes are described in detail in Section E of this Attachment. The marked-up TS pages for CTS and ITS are enclosed in Attachment B-1 and B-2, respectively. In addition, the associated TS Bases sections have been revised to be consistent with the TS revisions. The revised TS Bases are included in Attachment E-1 and E-2 for CTS and ITS, respectively.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

The following section discusses the current TS requirements for which a change is requested, referencing CTS and ITS as applicable.

Current Requirements for CTS

1. TS Section 2.1.B, "Thermal Power, High Pressure and High Flow," requires that the Minimum Critical Power Ratio (MCPR) shall not be less than 1.10 for Unit 3 and 1.09 for Unit 2 with cycle exposures less than or equal to 13,800 MWd/MTU and 1.12 for Unit 2 with cycle exposures greater than 13,800 MWd/MTU with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, the MCPR limit shall be increased by 0.01.

2. TS Surveillance Requirement (SR) 4.3.A.2, "Shutdown Margin," requires that the Shutdown Margin (SDM) is to be verified acceptable within 24 hours after detection of a withdrawn control rod that is immovable.
3. TS Section 3/4.3.C, "Control Rod Operability," describes the requirements for control rod operability in operational modes 1, "Power Operation," and 2, "Startup."

4. TS Section 3/4.3.D, "Maximum Scram Insertion Times," requires that the maximum scram insertion time of each control rod shall not exceed 7 seconds and states requirements for demonstrating control rod scram times.

5. TS Section 3/4.3.E, "Average Scram Insertion Times," requires that the average scram time of all operable control rods not exceed specified times and that the average scram times be demonstrated in accordance with TS Section 4.3.D.

6. TS Section 3/4.3.F, "Group Scram Insertion Times," requires that the average scram time for the three fastest rods of all 2x2 control rod groups not exceed specified times and that these times be demonstrated in accordance with TS 4.3.D.

7. TS Section 3/4.3.G, "Control Rod Scram Accumulators," requires that all control rod scram accumulators be operable in operational modes 1,2, and 5, and states requirements for demonstrating operability of the scram accumulators.

8. TS Section 3.3.H, "Control Rod Drive Coupling," requires that all control rods be coupled to their drive mechanisms in operational modes 1,2, and 5.

9. TS Section 3.3.I, Control Rod Position Indication System," requires that all control rod position indicators shall be operable in operational modes 1,2, and 5.

10. TS Section 3.3.L and SR 4.3.L, "Rod Worth Minimizer," require that the RWM be operable prior to reducing thermal power to less than or equal to 20% of rated thermal power.

11. TS Section 3.6.A, "Recirculation Loops," Action 1.a requires that, for single loop operation, the MCPR Safety Limit be increased by 0.01 in accordance with TS Section 2.1.B. Action 1.b requires that, for single loop operation, the MCPR Operating Limit be increased by 0.01 in accordance with TS Section 3.11.C.

12. TS Section 3.11.B, "Transient Linear Heat Generation Rate," requires that the Transient Linear Heat Generation Rate (TLHGR) shall be maintained such that the Fuel Design Limiting Ratio for Centerline (FDLRC) Melt is less than or equal to 1.0. With FDLRC greater than 1.0, actions shall be taken to either 1) restore FDLRC to less than or equal to 1.0, or 2) adjust the flow biased Average Power Range Monitor (APRM) setpoints by 1/FDLRC, or 3) adjust each APRM gain such that the APRM readings are ≥ 100% times the Fraction of Rated Thermal Power (FRTP) times FDLRC.

13. TS Section 6.9.A.6.b, "Core Operating Limits Report (COLR)," requires that the analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC. The specific approved methods are listed.

Requirements for ITS

14. TS Section 3.1.4, "Control Rod Scram Times," requires that each control rod scram time be within the limits specified in Table 3.1.4-1 and that no more than 12 control rods or 2 adjacent rods be "slow" in accordance with the table.

15. TS Section 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," requires that, a) FDLRC shall be less than or equal to 1.0; or b) each required APRM Flow Biased Neutron Flux – High Function Allowable Value shall be modified by 1/FDLRC; or c) each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times the FRTP times FDLRC.

16. TS Section 5.6.5.b, "Core Operating Limits Report (COLR)," requires that the analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC. The specific approved methods are listed.

C. BASES FOR THE CURRENT REQUIREMENTS

1. MCPR Safety Limit (current requirement #1). The fuel cladding integrity Safety Limit is set such that no mechanistic fuel damage is calculated to occur if the limit is not violated. Because the transition boiling correlation is based on a significant quantity of test data, there is 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. 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 traversing incore probe (TIP) measurements.

2. SDM SR (current requirement # 2). The SDM calculations are performed assuming the highest worth control rod fully withdrawn and all others inserted. Upon determination that one control rod is incapable of being fully inserted, the SDM calculation must be re-performed to evaluate the core with the stuck rod at its new position and the highest worth rod re-determined and assumed to be withdrawn. This ensures that the analysis is performed to correctly model the cycle's operation.

3. Control rod operability and scram insertion times (current requirements #3-7). These TS requirements ensure that the performance of the control rods meets the assumptions used in the safety analyses in the event of an accident or transient. The limit on average scram insertion times ensures that the control rod insertion times are consistent with those used in the safety analyses. The negative reactivity insertion rate that results from the limiting average scram time provides the required protection to maintain the MCPR greater than the safety limit. The performance of the individual Control Rod Drives (CRDs) is monitored to assure that scram performance is not degraded. Transient analyses are performed for both the TS scram speed and the nominal scram speed insertion times. These analyses result in the development of the fuel cycle-dependent MCPR operating limits.

4. Control Rod Drive Coupling (current requirement #8). If control rod coupling is maintained, the possibility of a rod drop accident is eliminated.

5. Control Rod Position Indication System (current requirement #9). In order to ensure that the control rod patterns can be followed and therefore that other fuel-related parameters are within their limits, the control rod position indication system must be operable.

6. RWM (current requirement #10). The RWM provides automatic supervision to assure that out-of-sequence control rods will not be inserted or withdrawn. This provides a backup to procedural control of control rod worth to limit maximum control rod worth, thus limiting the consequences of a postulated control rod drop accident (CRDA).

7. Recirculation loops (current requirement #11). The transient analyses of Chapter 15, "Accident and Transient Analysis," of the Updated Final Safety Analysis Report (UFSAR) are performed for single recirculation loop operation to maintain fuel thermal margins during the Abnormal Operational Occurrences (AOOs) analyzed provided the MCPR fuel cladding safety limit is increased by 0.01 as noted by TS Section 2.1.B.

8. TLHGR (current requirement #12). The TLHGR is monitored by FDLRC, which is a SPC limit. Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% Rated Thermal Power (RTP). When FDLRC is greater than 1.0, excessive power peaking exists. To maintain margins similar to those at RTP conditions, the APRM flow biased scram setpoint is decreased by 1/FDLRC. As an alternative, this adjustment may also be accomplished by increasing the gain of the APRM.

9. COLR (current requirement #13 and ITS requirement #16). The approved analytical methods in the TS reflect SPC methodology.

10. Control rod scram times (ITS requirement #14). The scram function of the CRD system controls reactivity changes during AOOs to ensure that specified acceptable fuel design limits are not exceeded. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

11. APRM Gain and setpoint (ITS requirement #15). The operability of the APRMs and their scram setpoints is an initial condition of all safety analyses that assume control rod insertion upon reactor scram. This Limiting Condition for Operation (TS LCO) is provided to require the APRM gain or APRM flow biased neutron flux-high function scram allowable value to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding 1% plastic strain limit. The condition of excessive power peaking is determined by FDLRC. Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% RTP.

D. NEED FOR REVISION OF THE REQUIREMENTS

The revisions to the requirements listed are necessary to support our change of fuel vendors from SPC to GE that will occur during the DNPS Units 2 and 3 refueling outages beginning in October, 2001, and September, 2002, respectively. In addition, certain poposed changes are requested to improve operational flexibility.

1. MCPR Safety Limit and Recirculation Loops (current requirements #1 and 11). The revision is necessary because the value of the difference between the single recirculation loop operation MCPR safety limit and the two recirculation loop operation MCPR safety limit may change as a result of changes in fuel types and reload designs. The actual values of the MCPR safety limits are not changed. However, with a shift to GE analysis methods, the value of the MCPR safety limit for single loop operation will be specified explicitly, rather than as an increment to the two loop operation limit, to properly reflect the fact that these limits are

calculated separately.

2. SDM, control rod operability, scram insertion times, control rod coupling, and control rod position indication (current requirements #2-9 and ITS requirement #14). The revisions are necessary to adopt the appropriate methodology for scram insertion times. CTS reflect an analysis methodology based on limiting the average scram insertion time. In ITS, scram times are controlled by limiting the number of rods with slow insertion times. Since the requested DNPS conversion to ITS is expected to be approved prior to approval of these proposed changes, the ITS approach will be used to analyze upcoming cycles. In order to ensure that the CTS requirements are based on the methodology used for the cycle analysis, the CTS were changed to reflect ITS requirements. This requires changing all of the CTS sections listed, in order to maintain consistency with the ITS proposed changes.

3. RWM (current requirement #10). The revision is needed to maintain consistency with vendor methodologies and to increase operational flexibility. This change also makes the DNPS low power setpoint (LPSP) consistent with the LPSPs for other ComEd BWR's.

4. TLHGR (current requirement #12 and ITS requirement #15). The revisions are necessary to ensure an equivalent level of TLHGR protection for GE fuel as is currently in place for SPC fuel, which monitors the parameter FDLRC. The GE methodology uses the ratio of the Maximum Fraction of Limiting Power Density (MFLPD) to the Fraction of Rated Thermal Power (FRTP) to protect TLHGR for GE fuel. FDLRC is retained because it is still applicable for SPC fuel.

5. COLR (current requirement #13 and ITS requirement #16). The revisions are necessary to ensure that the methods and references reflect the appropriate approved fuel design and analytical methods for developing operating limits.

E. DESCRIPTION OF THE PROPOSED CHANGES

Proposed Changes to CTS

1. TS Section 2.1.B, "Thermal Power, High Pressure and High Flow," is revised to remove the statement that the single loop operation MCPR Safety Limit is 0.01 greater than the two loop operation MCPR Safety Limit. This requirement is replaced with the numerical value for the single loop operation MCPR Safety Limit .

2. TS SR 4.3.A.2, Shutdown Margin," is revised to require that the SDM be verified acceptable within 72 hours of discovering a control rod that is stuck.

3. TS Section 3/4.3.C, "Control Rod Operability," is revised to reflect ITS Section 3.1.3, "Control Rod Operability," requirements, stated in CTS format. Revised TS LCO 3.3.C has incorporated portions of CTS Sections 3.3.D, 3.3.H, and 3.3.I in order to contain all of the requirements for determining the operability of control

rods. The specific changes are shown in the marked-up TS pages in Attachment B-1.

4. a. TS Section 3/4.3.D, "Maximum Scram Times," is revised to reflect ITS Section 3.1.4, "Control Rod Scram Times," requirements, stated in CTS format. The revision reflects a change from specifying the average control rod scram time to specifying the times required for each control rod and limiting the number of slow control rods. The specific changes are shown in the marked-up TS pages in Attachment B-1.

   b. In addition to the changes described in 4.a above, the required scram times are modified to reflect both SPC and GE methodology for ensuring that the scram times reflect the analysis methods used to protect the fuel from exceeding thermal limits. These scram times are included in new TS Table 3.3.D-1.

5. TS Section 3/4.3.E, "Average Scram Insertion Times," is deleted. The average scram time requirement is replaced with the requirement to limit the number of slow rods. The SRs are incorporated in revised TS LCO 3.3.D. The specific changes are shown in the marked-up TS pages in Attachment B-1.

6. TS Section 3/4.3.F, "Group Scram Insertion Times," is deleted. The limitation on group scram times is replaced with the requirement to limit the number of slow rods. The SRs are incorporated in revised TS LCO 3.3.D. The specific changes are shown in the marked-up TS pages in Attachment B-1.

7. TS Section 3/4.3.G, "Control Rod Scram Accumulators," is revised to reflect ITS Section 3.1.5, Control Rod Scram Accumulators," stated in CTS format. The revised specification requires that control rods with inoperable accumulators be declared "slow." The specific changes are shown in the marked-up TS pages in Attachment B-1.

8. TS Section 3/4.3.H, "Control Rod Drive Coupling," is revised to reflect ITS Section 3.1.3 requirements in operational modes 1 and 2. This relocates the requirements for control rod coupling for modes 1 and 2 to revised TS Section 3.3.D. The TS Section remains unchanged for operational mode 5. The specific changes are shown in the marked-up TS pages in Attachment B-1.

9. TS Section 3/4.3.I, Control Rod Position Indication System," is revised to reflect ITS Section 3.1.3 requirements in operational modes 1 and 2. This relocates the requirements for control rod position indication for modes 1 and 2 to revised TS Section 3.3.D. The TS Section is unchanged for operational mode 5. The specific changes are shown in the marked-up TS pages in Attachment B-1.

10. TS Section 3.3.L and SR 4.3.L, "Rod Worth Minimizer," are revised to reduce the thermal power limit at which the RWM shall be operable from 20% to 10%.

11. TS Section 3.6.A, "Recirculation Loops," Action 1.a is revised to remove the requirement that the single loop operation MCPR Safety Limit be increased by 0.01. This is replaced with a requirement to increase the single loop operation MCPR Safety Limit to the value specified in Section 2.1.B. In Action 1.b, the requirement that the single loop operation MCPR Operating Limit be increased by 0.01 is removed and replaced with a requirement to increase the single loop operation MCPR Operating Limit in accordance with the COLR.

12. TS Section 3.11.B, "Transient Linear Heat Generation Rate." The ratio of MFLPD/FRTP is substituted for FDLRC for monitoring GE fuel. The use of FDLRC for monitoring SPC fuel is retained in a footnote added to the TS Section. The use of MFLPD/FRTP to adjust APRM scram settings or APRM gains is similarly substituted.

13. TS Section 6.9.A.6.b, "Core Operating Limits Report", is modified to add GE's NRC-approved analytical methodology document and GE's methodology for determining critical power for SPC fuel. Section I of this Attachment provides references related to NRC approval of these methods.

Proposed Changes to ITS

14. TS Section 3.1.4, "Control Rod Scram Times." Table 3.1.4-1 is revised to add the GE-based ITS timing requirements to the current SPC-based timing requirements. The GE values added are as follows.

Percent Insertion	Scram Times for GE –Analyzed Cores (seconds)
5	0.48
20	0.89
50	1.98
90	3.44

15. TS Section 3.2.4, "APRM Gain and Setpoint." The ratio of MFLPD/FRTP is added for monitoring GE fuel. The use of FDLRC for monitoring SPC is retained.

16. TS Section 5.6.5.b, "Core Operating Limits Report," is modified to add GE's NRC approved analytical methodology document and GE's methodology for determining critical power for SPC fuel. Section I of this Attachment provides references related to NRC approval of this method.

Proposed Changes to both CTS and ITS

17. The definitions of the Fraction of Limiting Power Density (FLPD), MFLPD, and FRTP are added to CTS Section 1.0, "Definitions." The definition of MFLPD is added to ITS Section 1.1, "Definitions."

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

1. MCPR Safety Limit and Recirculation Loops (changes #1 and 11). These are administrative changes. The removal of the specific requirement that the single loop operation MCPR Safety Limit and Operating Limit be 0.01 higher than the two loop operation MCPR Safety Limit and Operating Limits does not change the actual MCPR limits. The revised TS Section 2.1.B specifies both the two loop operation and the single loop operation MCPR Safety Limit. For TS Section 3.6.A, the MCPR Safety and Operating limits are incorporated by reference.

2. SDM (change #2). With a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach COLD SHUTDOWN is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with this postulated additional single failure, sufficient reactivity control remains to reach and maintain HOT SHUTDOWN conditions. Also, a notch test is required by revised TS LCO 3.3.C Action 1.d for each remaining withdrawn control rod to ensure that no additional control rods are stuck. Given these considerations, the time to demonstrate SDM in CTS Section 3.3.C Action 1.c and CTS Section 4.3.A.2 has been extended from 24 hours to 72 hours, and provides a reasonable time to perform the analysis or test. This is consistent with the BWR Improved Standard Technical Specifications (ISTS), Reference I.6.

3. Control Rod Operability and Scram Insertion Times (changes #3-9). The CTS requirements are modified to adopt the ITS methodology for control rod scram timing. These changes make the CTS requirements identical to the ITS requirements for control rod operability and scram timing. The safety analysis for each change is presented below. The alphanumeric designators for the changes refer to the designators shown in the CTS marked-up pages in Attachment B-1. The changes are grouped into categories that are consistent with the standard conventions used in converting CTS to ITS, described in Reference I.6. The categories are explained in Attachment F.

   Revised TS Section 3.3.C (Changes # 3,8,9) - ADMINISTRATIVE CHANGES

   A.1 In the proposed revisions, certain wording preferences or conventions are adopted that do not result in technical changes, either actual or interpretational.

   A.2 The organization of the Control Rod OPERABILITY TS Section (i.e., revised TS LCO 3.3.C) is proposed to include all conditions that can affect the ability of the control rods to provide the necessary reactivity insertion. The proposed TS Section is also simplified as follows.

   1) A control rod is considered "inoperable" only when it is degraded to the point that it cannot provide its scram functions. All

inoperable control rods (except stuck rods) are required to be fully inserted and disarmed.

2) A control rod is considered inoperable and stuck if it is incapable of being inserted. Requirements are retained to preserve SDM for this situation.

3) Special considerations are provided for nonconformance to the analyzed rod position sequence, due to inoperable control rods, at < 10% of RATED THERMAL POWER.

A.3 Not Used

A.4 A Note is added to CTS Section 3.3.C, Actions 1 and 2 (i.e., revised TS LCO 3.3.C footnotes to Actions 1 and 3.a) that allows for bypassing the RWM, if needed for continued operations. This note is informative in that the RWM may be bypassed at any time, provided the proper Actions of CTS Section 3.3.L, the RWM requirements, are taken. This is a human factors consideration to assure clarity of the requirement and allowance.

A.5 The existing phrase, "Immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable," in CTS Section 3.3.C Action 1 and CTS Section 4.3.A.2 has been replaced with the term "stuck" in proposed Action 1 of revised TS LCO 3.3.C. The intent of the existing wording is consistent with the proposed simplification. Details of potential mechanisms by which control rods may be stuck are not necessary for inclusion within the Specification.

A.6 CTS SR 4.3.C.1 pertains to control rods "not required to have their directional control valves disarmed electrically or hydraulically." This phrase thus exempts this surveillance for inoperable control rods. In accordance with TS Section 4.0.C, inoperable control rods are not required to meet this SR and, therefore, CTS Section 4.3.C.1 only applies to OPERABLE control rods. Thus, this phrase is proposed to be deleted.

A.7 These listed surveillances in CTS Section 4.3.C.2 are required by other TS SRs. Repeating a requirement to perform these surveillances is not necessary. Elimination of this cross-reference is therefore administrative.

A.8 CTS Section 3.3.C Actions 1.a.2), 2.b, and 2.c, footnote (a), CTS Section 3.3.H, Action 1.b, footnote (b), and CTS Section 3.3.I, Action 1.c, footnote (b), which permit the directional control valves to be rearmed intermittently, has been deleted since TS Section 3.0.E provides this allowance. Therefore, deletion of this allowance is administrative.

A.9 Not used

A.10 The CTS Section 3.3.D requirement that maximum control rod scram insertion time be ≤ 7 seconds is presented in proposed SR 4.3.C.4, making it a requirement for control rods to be considered OPERABLE. Eliminating the separate Specification for excessive scram time by moving the requirement to an SR does not eliminate any of the requirements, or impose a new or different treatment of the requirements other than those proposed in L.6 below. Therefore, this proposed change is administrative.

A.11 The definition of time zero in CTS Section 3.3.D (i.e., "based on de-energization of the scram pilot valve solenoids as time zero") has been deleted since it is duplicative of the definition of time zero in CTS Section 3.3.E and 3.3.F, which is maintained in proposed footnote (a) to Table 3.3.D-1. No change has been made to the defined time zero; therefore, this deletion is administrative.

A.12 CTS Section 4.3.D, which provides the scram time testing requirements, is addressed in proposed SR 4.3.D. Therefore, proposed SR 4.3.C.4 has been added to require the SRs in 4.3.D to be performed. Changes to the testing requirements located in SR 4.3.D as SRs 4.3.D.1, 4.3.D.2, 4.3.D.3, and 4.3.D.4 are addressed in the safety analysis for SR 4.3.D.

A.13 The CTS Section 3.3.H requirement that control rods be coupled to their drive mechanism is presented in proposed SR 4.3.C.5. As a Surveillance in the Control Rod OPERABILITY TS LCO, it is a requirement for control rods to be considered OPERABLE. The actions for uncoupled control rods continue to be required. See L.5, L.7, L.8, L.9, and L.10 below. Eliminating the separate TS LCO for control rod coupling, by moving the Surveillance and Actions to another TS Section, does not eliminate any requirements or impose a new or different treatment of the requirements other than those separately proposed. Therefore, this proposed change is administrative.

A.14 CTS Section 3.3.H Action 1.a contains the method of restoring coupling integrity to an uncoupled control rod (i.e., insert the control rod drive mechanism to accomplish recoupling). The revised presentation of actions, based on the BWR ISTS, Reference I.6, is proposed to not explicitly detail options to "restore...to OPERABLE." This action is always an option, and is implied in all Actions. Omitting this action is purely editorial.

A.15 CTS Section 3.3.I requires all control rod position indicators to be Operable. The objective of the CTS Section 3.3.I requirement is understood to be related to each control rod. Each specific Action and each SR refer to individual control rods. Therefore, the interpretation of this TS LCO is that each control rod shall have at least one control rod position indication.

The basis of the requirement that each control rod have at least one control rod position indication is presented in SR 4.3.C.1. The effect of relocating the requirement for control rod position indication is to make it a requirement for control rods to be considered OPERABLE. Eliminating the separate TS LCO for control rod position indication by moving the SR and Actions to another Specification does not eliminate any requirements or impose a new or different treatment of the requirements other than those separately proposed. Similarly, CTS Section 3.3.I Action 1 addresses this intent. The proposed SR 4.3.C.1 has combined the CTS Section 3.3.I objective with the CTS Section 3.3.I Action 1 objective to require the position of the control rod be determined. If the position can be determined, the control rod may be considered OPERABLE, and continued operation allowed. This outcome is identical, whether complying with CTS Section 3.3.I Action 1, or meeting proposed SR 4.3.C.1.

Revised TS Section 3.3.C (Changes # 3,8,9) - TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 A proposed Action has been added to CTS Section 3.3.C Action 1.a to require the immediate verification that the stuck control rod separation criteria are met. The actual criteria are specified in the Bases and are applicable to SPC and GE methodologies. The stuck control rod separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods elsewhere in the core adjacent to one another. The description of "slow" control rods is provided in revised TS LCO 3.3.D, "Control Rod Scram Times." The stuck separation criteria ensures local scram reactivity rate assumptions are met.

M.2 CTS Section 3.3.C Actions 1.a.1) and 2.a.1) require the separation criteria to be met only for withdrawn control rods. Action 4 of the revised TS LCO 3.3.C applies to all inoperable control rods whether inserted or withdrawn, and is therefore, more restrictive. This revised separation criteria requirement is necessary to ensure the safety analysis assumptions are met.

M.3 The CTS Section 3.3.C Actions require TS LCO 3.0.C entry (i.e., within one hour, take action to place the unit in an operational mode in which the requirement does not apply) if more than one control rod is stuck. The proposed TS LCO 3.3.C Action 2 maintains the equivalent shutdown action as TS LCO 3.0.C, but also contains an additional requirement in proposed Action 1.b to disarm the stuck control rod. The Bases for this action states that the disarming is to be performed hydraulically. This requirement provides a level of protection to the control rod drive should a scram signal occur. If mechanically bound, the stuck control rod could cause further damage if not hydraulically disarmed. In addition, CTS Section 3.3.C Action 1.a.2)a) allows a stuck control rod to be disarmed electrically. This allowance has been deleted. The stuck control rod can only be disarmed hydraulically. This will also prevent potential damage if a scram signal occurs, since the means by which hydraulic disarming is performed will preclude scram pressure from being applied.

M.4 Not used.

M.5 Proposed SRs 4.3.C.2 and 4.3.C.3 require control rods to be inserted in lieu of the CTS Section 4.3.C.1 requirement for moving the control rods. The existing requirement can be met by control rod withdrawal. It is conceivable that a mechanism causing binding of the control rod that prevents insertion can exist such that a withdrawal test will not detect the problem. Since the purpose of the test is to assure scram insertion capability, restricting the test to only allow control rod insertion provides an increased likelihood of this test detecting a problem that impacts this capability.

M.6 The proposed changes to CTS Section 3.3.C Action 2.a.2) including footnote (b), for non-stuck inoperable control rods, eliminates the check of insertion capability; replacing it with a requirement to fully insert and disarm all inoperable control rods. CTS Section 3.3.C Action 2.a.2), requiring the insertion capability to be verified and allowing the control rod to remain withdrawn, is applicable to conditions such as: 1) one inoperable CRD accumulator, and 2) loss of position indication while below the LPSP. The first condition is addressed in the safety analysis for revised TS LCO 3.3.G. The latter condition would no longer allow the affected control rod to remain withdrawn and not disarmed. This added restriction on control rod(s) with loss of position indication is conservative with respect to scram time and SDM since an inoperable, but not stuck, control rod is not disarmed while it is withdrawn. Actions for inoperable control rods not complying with analyzed rod position sequence (i.e., revised TS LCO 3.3.C Action 4) assure that insertion of these control rods remains appropriately controlled.

<u>Revised TS Section 3.3.C (Changes #3,8,9) - TECHNICAL CHANGES – LESS RESTRICTIVE</u>

LA.1 The details of the recommended procedures for disarming control rod drives (CRDs) specified in CTS Section 3.3.C Actions 1.a.2, with the exception of electrical disarming (i.e., see M.3 above), 2.b, and 2.c, CTS Section 3.3.H Action 1.b, and CTS Section 3.3.I Action 1.c are proposed to be relocated to the Bases. These details are not necessary to ensure the associated CRDs of inoperable control rods are disarmed. Revised TS LCO 3.3.C Actions 1.b and 3.b, which require disarming the associated CRDs of inoperable control rods, are adequate for ensuring associated CRDs and inoperable control rods are disarmed. Therefore, the relocated

details are not required to be in the TS to provide adequate protection of the public health and safety.

LA.2 CTS Section 3.3.I Actions 1.a and 1.b, which determine the position of the control rod, which is now proposed to be an SR for control rod OPERABILITY, can be met a number of ways. Two ways are presented: by using an alternate method and by moving the control rod to a position with an OPERABLE position indicator. These details of methods for determining the position of a control rod are proposed to be relocated to the Bases for the proposed SR 4.3.C.1. This SR, which requires the position of each control rod to be determined every 24 hours, is adequate for ensuring the position of the control rods is determined. Therefore, the relocated details are not required to be in the TS to provide adequate protection of the public health and safety.

L.1 CTS Section 3.3.C Actions 1.a and 2.a are presented in revised TS LCO 3.3.C Action 4 to provide the requirements and actions for the local distribution of inoperable control rods. Three distinct changes are addressed.

1) Revised TS LCO 3.3.C Action 4 is modified by a Note excluding its applicability above 10% RTP. The existing separation requirements for a stuck control rod, in part, account for allowing <u>withdrawn</u> inoperable control rods. (See M.2 above.) To preserve scram reactivity, a stuck rod must be separated from other <u>withdrawn</u> inoperable control rods which may also not scram. In the proposed change, all inoperable control rods which will not scram are required to be fully inserted, and therefore, cannot impact scram reactivity. Therefore, scram reactivity remains preserved at all power levels and is unaffected by this proposed change.

   Separation requirements are required when below 10% RTP because of CRDA concerns related to control rod worth. Above 10% RTP, control rod worths that are of concern for the CRDA are not possible.

2) Revised TS LCO 3.3.C Action 4 also does not require actions for inoperable control rods whose position is in conformance with the analyzed rod position sequence constraints, even if the inoperable control rods are within two cells of each other. As discussed above in the first item of this category of changes, adequate limits to control core reactivity and power distribution above 10% RTP remain with this proposed change. Below 10% RTP, the appropriate core reactivity and power distribution limits are controlled by maintaining control rod positions within the limits of the analyzed rod position sequence and maintaining scram times within the limits of CTS Section 3.3.E and 3.3.F, as modified to

reflect revised TS LCO 3.3.D. If the two inoperable control rods were both "stuck," actions require an immediate shutdown, regardless of their proximity. Therefore, the limitation on the local distribution of inoperable control rods that comply with the analyzed rod position sequence is overly restrictive.

3) Finally, the actions for revised TS LCO 3.3.C Action 4 allow four hours to correct the situation prior to commencing a required shutdown, while CTS Section 3.3.C Actions 1.a and 2.a allow one hour. This increase is proposed in recognition of the actual operational steps involved on discovery of inoperable control rod(s). Time is first required to attempt identification and correction of the problem. Additional time is necessary to fully insert and then disarm the affected control rod(s). After these high priority steps are accomplished, attention can be turned to correcting localized distribution of inoperable control rods that deviate from the analyzed rod position sequence. Given the low probability of a CRDA during this brief proposed time extension, and the desire not to impose excessive time constraints on operator actions that could lead to hasty corrective actions, the proposed extension to this action does not represent a significant safety concern. This is consistent with the BWR ISTS.

L.2 Disarming a control rod as required by CTS Section 3.3.C Action 1.a.2) involves personnel actions by other than control room operating personnel. These processes require coordination of personnel and preparation of equipment, and potentially require anti-contamination "dress-out," in addition to the actual procedure of disarming the control rod. Currently, all these activities must be completed and the control room personnel must confirm completion within the same one hour allowed to insert the control rod. This is proposed to be extended to two hours in revised TS LCO 3.3.C Action 1.b, consistent with the guidance in Reference I.6, in recognition of the potential for excessive haste required to complete this task. The proposed 2 hour time does not represent a significant safety concern as the control rod is already in an acceptable position in accordance with other actions, and the action to disarm is solely a mechanism for precluding the potential for damage to the CRD mechanism.

L.3 CTS Section 4.3.C.1.a, which verifies control rods to be non-stuck, is proposed to be extended from seven days to 31 days for control rods that are not fully withdrawn (i.e., proposed SR 4.3.C.3). This is acceptable given the following.

1) At full power, a large percentage of control rods (i.e., 80% to 90%) are fully withdrawn and would continue to be exercised each week. This represents a significant sample size when looking for an unexpected random event (i.e., a stuck control rod).

2) Operating experience has shown "stuck" control rods to be an extremely rare event while operating.

3) Should a stuck rod be discovered, 100% of the remaining control rods, even those partially withdrawn, must be tested within 24 hours (i.e., revised TS LCO 3.3.C Action 1.d).

L.4 With a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach COLD SHUTDOWN is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with this postulated additional single failure, sufficient reactivity control remains to reach and maintain HOT SHUTDOWN conditions. Also, a notch test is required by revised TS LCO 3.3.C Action 1.d for each remaining withdrawn control rod to ensure that no additional control rods are stuck. Given these considerations, the time to demonstrate SDM in CTS Section 3.3.C Action 1.c and CTS Section 4.3.A.2 has been extended from 24 hours to 72 hours, and provides a reasonable time to perform the analysis or test.

L.5 CTS Section 3.3.C Action 2, for excessive scram speed and certain combinations of conditions with a low pressure on a control rod scram accumulator, CTS Section 3.3.H Action 1, for uncoupled control rods, and CTS Section 3.3.I Action 1, for inoperable control rod position indication, provide actions for inoperable control rods. Both CTS Section 3.3.C Action 2 and CTS Section 3.3.H Action 1 provide a total of two hours to insert and disarm the control rods, while CTS Section 3.3.I provides only one hour. In the proposed revision, all inoperable non-stuck control rods are required to be fully inserted and disarmed as described in M.6 above. The time allowed to complete the insertion is proposed to be extended to 3 hours (i.e., revised TS LCO 3.3.C Action 3.a); for all cases an additional hour is provided to disarm the associated CRD (i.e., revised TS LCO 3.3.C Action 3.b). The additional time provides the necessary time to insert and disarm the control rods in an orderly manner and without challenging plant systems. The RWM may be required to be bypassed to allow the rod to be inserted, therefore, the current action times may not be sufficient under all cases.

In addition, disarming a control rod can involve personnel actions by other than control room operating personnel. This process requires coordination of personnel and preparation of equipment, and potentially requires anti-contamination "dress-out," in addition to the actual procedure of disarming the control rod.

The disarming is proposed to be extended to 4 hours in revised TS LCO 3.3.C Action 3.b, one hour beyond that allowed to insert, consistent with the guidance in the ISTS, in recognition of the potential for excessive haste required to complete this task. The proposed four hour time does not represent a significant safety concern since the control rod will be inserted within three hours, and the action to disarm is solely a mechanism for precluding the potential for future misoperation.

L.6 The CTS Section 3.3.D Action 2 requirement for additional scram time surveillance testing when three or more control rods exceed the maximum scram time is deleted. During normal power operating conditions, scram testing is a significant perturbation to steady state operation, involving significant power reductions, abnormal control rod patterns and abnormal control rod drive hydraulic system configurations. Requiring more frequent scram time surveillance tests is therefore not desirable. Because of the frequent testing of control rod insertion capability (i.e., proposed SR 4.3.C.2 and SR 4.3.C.3) and accumulator OPERABILITY (i.e., proposed SR 4.3.E.1), and the operating history demonstrating a high degree of reliability, the more frequent scram time testing is not necessary to assure safe plant operations. In addition, since the shutdown requirement could have only applied to CTS Section 3.3.D Action 2 (i.e., since a control rod can always be declared inoperable), this part of CTS Section 3.3.D Action 2 has also been deleted.

L.7 Coupling requirements during refueling (i.e., OPERATIONAL MODE 5) specified by CTS Section 3/4.3.H are not necessary since only one control rod can be withdrawn from core cells containing fuel assemblies. The probability and consequences of a single control rod dropping from its fully inserted position to the withdrawn position of the control rod drive are negligible (i.e., reactor will remain subcritical and within the limits of the CRDA assumptions).

L.8 If an uncoupled control rod is not allowed by the RWM to be inserted to accomplish recoupling, CTS Section 3.3.H Action b requires the control rod be inserted. This will require bypassing the RWM and operation with an out-of-sequence control rod. Therefore, coupling attempts are allowed regardless of the RWM allowance because of the short time allowed. If coupling is not established within three hours, the control rod must be fully inserted and disarmed (i.e., revised TS LCO 3.3.C Actions 3.a and 3.b).

L.9 Proposed SR 4.3.C.5 verifies a control rod does not go to the withdrawn overtravel position. An uncoupled control rod would fail to meet this SR. After restoration of a component that caused a failure to meet an SR, the appropriate SRs are performed to demonstrate the OPERABILITY of the affected components. The requirement to verify control rod coupling by observation of nuclear instrumentation response is addressed in L.10 below. As a result, the CTS Section 3.3.H Actions 1.a and 1.a.2) requirements are proposed to be deleted since they are not necessary for ensuring recoupling of the control rod.

L.10 The CTS Section 3.3.H Action 1.a.1) requirement to verify control rod coupling by observing any indicated response of the nuclear instrumentation during withdrawal of a control rod is proposed to be deleted. A response to control rod motion on nuclear instrumentation is indicative that a control rod is following its drive, but gives no indication as to whether or not a control rod is coupled. Likewise, failure to have a response to control rod motion on nuclear instrumentation does not indicate that a rod is uncoupled. Thus, the results from monitoring nuclear instrumentation are inconclusive to use as a verification that the control rod is coupled. Proposed SR 4.3.C.5 requires verification that a control rod does not go to the withdrawn overtravel position. The overtravel feature provides a positive check of coupling integrity since only an uncoupled control rod can go to the overtravel position. This verification is required to be performed any time a control rod is withdrawn to the full out position and prior to declaring a control rod operable after work on the control rod or CRD System that could affect coupling. As a result, SR 4.3.C.5 provides adequate assurance that the control rods are coupled.

L.11 CTS Section 4.3.I.2 requires that the indicated control rod position change during the movement of the control rod drive when performing the control rod movement tests (i.e., CTS Section 4.3.C.1). To perform control rod movement tests required by CTS Section 4.3.C.1 (i.e., proposed SRs 4.3.C.2 and 4.3.C3), position indication must be available. If position indication is not available, this test cannot be satisfied and appropriate actions will be taken for inoperable control rods in accordance with the actions of revised TS LCO 3.3.C. As a result, the requirements for the control rod position indication system are adequately addressed and are proposed to be deleted.

Revised TS Section 3.3.D (Changes # 4, 5, 6) – ADMINISTRATIVE CHANGES

A.1 In the proposed revisions, certain wording preferences or conventions are adopted that do not result in technical changes, either actual or interpretational.

A.2 CTS Section 4.3.D.2 footnote (a), which states that the provisions of TS Section 4.0.D (i.e., the requirement to perform SRs prior to entry into applicable modes) are not applicable, has been deleted since TS Section 4.0.D provides this allowance (i.e., by providing for stated exceptions). Therefore, deletion of this allowance is administrative.

Revised TS Section 3.3.D (Changes # 4, 5, 6) – TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 An additional SR 4.3.D.3, is proposed. This new SR will require a scram time test, which may be done at any reactor pressure, prior to declaring the control rod operable and, thus, enabling its withdrawal during a startup. To allow testing at less than normal operating pressures, a requirement for scram time limits at <800 psig is included (i.e., proposed TS Table 3.3.D-1 footnote (b)). These limits appear less restrictive than the operating limits; however, due to reactor pressure not being available to assist the scram speed, the limits are reasonable for application as a test of operability at these conditions. This ensures the affected control rod retains adequate scram performance over the range of applicable reactor pressures. Since this test, and therefore any limits, are not applied in the existing Specification, this is an added restriction. In addition, the reactor pressure applicability of CTS Section 4.3.D (i.e., proposed SRs 4.3.D.1, 4.3.D.2, and 4.3.D.4) has been changed from > 800 psig to ≥ 800 psig for consistency with the proposed SR.

M.2 The purpose of the control rod scram time TS LCOs is to ensure the negative scram reactivity corresponding to that used in licensing basis calculations is supported by individual control rod drive scram performance distributions allowed by the TS. CTS Sections 3.3.D, 3.3.E, and 3.3.F accomplish the above purpose by placing requirements on maximum individual control rod drive scram times (i.e., seven second requirement), average scram times, and local scram times (i.e., a four control rod group). In the proposed revisions, the negative scram reactivity assumptions are maintained by ensuring that each control rod meets the seven second insertion time and by addressing the number of rods that are slow compared to the requirements of TS Table 3.3.D-1. SPC and GE methodologies treat slow rods slightly differently; this explains the differences in TS Table 3.3.D-1 for SPC and GE analyzed cores. These differences are explained below.

<u>SPC methodology</u>

Because of the methodology used in the design basis transient analysis using one-dimensional neutronics, all control rods are assumed to scram at the same speed, which is the analytical scram time requirement. Performing an evaluation assuming all control rods scram at the analytical limit results in the generation of a scram reactivity versus time curve, the analytical scram reactivity curve. The purpose of the scram time TS LCO is to ensure that, under allowed plant conditions, this analytical scram reactivity will be met. Since scram reactivity cannot be readily measured at the plant, the safety analyses use appropriately conservative scram reactivity versus insertion fraction curves to account for the variation in scram reactivity during a cycle. Therefore, the TS requirements must only ensure the scram times are satisfied.

The first result is that, if all control rods scram at least as fast as the analytical limit, the analytical scram reactivity curve will be met. However, a distribution of scram times (i.e., some slower and some faster than the analytical limit) can also provide adequate scram reactivity. By definition, for a situation where all control rods do not satisfy the analytical scram time limits, the condition is acceptable if the resulting scram reactivity meets or exceeds the analytical scram reactivity curve. This can be evaluated using models which allow for a distribution of scram speeds. It follows that the more control rods that scram slower than the analytical limit, the faster the remaining control rods must scram to compensate for the reduced scram reactivity rate of the slower control rods. Revised TS LCO 3.3.D incorporates this philosophy by specifying scram time limits for each individual control rod instead of limits on the average of all control rods and the average of three fastest rods in all 2x2 control rod groups. This philosophy has been endorsed by the BWR Owners' Group and described in EAS-46-0487, "Revised Reactivity Control Systems Technical Specifications," which has been accepted by the NRC as part of the BWR ISTS. The scram time limits listed in TS Table 3.3.D-1 have margin to the analytical scram time limits listed in EAS-46-0487, Table 3-4 to allow for a specified number and distribution of slow control rods, a single stuck control rod and an assumed single failure. Therefore, if all control rods met the scram time limits found in TS Table 3.3.D-1, the analytical scram reactivity assumptions are satisfied. If any control rods do not meet the scram time limits, revised TS LCO 3.3.D specifies the number and distribution of these slow control rods to ensure the analytical scram reactivity assumptions are still satisfied.

GE Methodology:

GE's approach also uses the BWROG application of EAS-46-0487 and EAS-56-0889, "BWR/2-5 Scram Time Technical Specification," which has been accepted by the NRC as part of the BWR ISTS. Whereas SPC methodology sets scram times that ensure an adequate scram reactivity insertion rate if no more than 12 rods are slow, GE's approach is to set slower scram times and then use actual average rod scram times to calculate the actual scram reactivity. This information is then used to set cycle-specific operating limits.

In both GE and SPC methods, if the number of slow rods is more than 12 or the rods do not meet the separation requirements, the unit must be shutdown within 12 hours. This change is considered more restrictive on plant operation since the proposed individual times are more restrictive than the average times. That is, currently, the average time of all rods or a group can be improved by a few fast scramming rods, even when there may be more than 12 slow rods, as defined in the proposed TS requirement. Therefore, revised TS LCO 3.3.D limits the number of slow rods to 12 and ensures no more than two slow rods occupy adjacent locations.

The maximum scram time requirement in CTS Section 3.3.D has been retained in SR 4.3.C.4 for the purpose of defining the threshold between a slow control rod and an inoperable control rod even though the analyses to determine the TS LCO scram time limits assumed slow control rods did not scram. Proposed Note 2 to TS Table 3.3.D-1 ensures that a control rod is not inadvertently considered "slow" when the scram time exceeds seven seconds.

Revised TS Section 3.3.D (Changes # 4, 5, 6) – TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 Proposed SR 4.3.D.2 will test a representative sample of control rods each 120 days of power operation instead of the CTS Section 4.3.D.3 SR to test 10% of the control rods on a rotating basis. The details of what constitutes a representative sample are proposed to be relocated to the Bases. Revised TS LCO 3.3.D and SR 4.3.D.2 are adequate to ensure scram time testing is performed. Therefore, the relocated details of what constitutes a representative sample are not required to be in the TS to provide adequate protection of the public health and safety.

L.1 CTS Section 4.3.D.1.a requires control rod scram time testing for all control rods prior to exceeding 40% RTP following CORE ALTERATIONS. This means that even if only one bundle is moved (e.g., replacing a leaking fuel bundle mid-cycle), all the control rods are required to be tested. Proposed SR 4.3.D.4 requires control rod scram time testing for only affected control rods following any fuel movement within the affected core cell. This change is acceptable since the objective of testing all of the control rods following CORE ALTERATIONS ensures the overall negative reactivity insertion rate is maintained following refueling activities that may impact a significant number of control rods (e.g., CRD replacement, CRD Mechanism overhaul, or movement of fuel in the core cell). When only a few control rods have been impacted by fuel movement, the effect on the overall negative reactivity insertion rate is insignificant. Therefore, it is not necessary to perform scram time testing for all control rods when only a few control rods have been impacted by fuel movement in the reactor pressure vessel. During a routine refueling outage, it is expected that all core cells will be impacted, thus all control rods will be tested, consistent with current requirements. This fact is stated in the Bases for SR 4.3.D.4. The SRs in 4.3.D are adequate to ensure that the negative reactivity insertion rate assumed in the safety analyses is maintained. Additionally, the reliability of the control rods is increased since this change eliminates unnecessary testing of the control rods.

Revised TS Section 3.3.G (Change #7 – ADMINISTRATIVE CHANGES

A.1 In the proposed revisions, certain wording preferences or conventions are adopted that do not result in technical changes, either actual or interpretational.

A.2 Not used

A.3 Not used

A.4 The revised presentation of CTS Section 3.3.G Action 1.a.1) does not explicitly detail options to restore control rod scram accumulators to OPERABLE status. This action is always an option, and is implied in all actions. Omitting this action is purely editorial.

A.5 Revised TS LCO 3.3.G does not contain the equivalent default action to be in at least HOT SHUTDOWN within the next 12 hours for failure to perform the CTS Section 3.3.G Action 1.a to declare the associated control rod inoperable. There are no circumstances which preclude the possibility of compliance with an action to declare the control rod inoperable. Therefore, deletion of this default action is inconsequential and considered administrative.

A.6 The conditions of CTS SR 4.3.G, which specify when the accumulator surveillance does not have to be performed (i.e., when the associated control rod is inserted and disarmed or scrammed), are duplicative of the allowance currently provided by TS Section 4.0.C. Therefore, the stated exception has been deleted.

A.7 The CTS Section 3.3.G Action 1.c.1) requirement to verify that a control rod drive pump is operating has been maintained, but the method for verifying this has been changed from inserting one control rod one notch to verifying that charging water header pressure is at least 940 psig. These methods both assure that sufficient control rod drive pressure exists to insert the control rods. The proposed method for determining charging water header pressure provides added assurance that the charging water pressure is sufficient to insert all control rods, whereas the existing method only assures that one rod can be inserted. Since the change is merely exchanging one test method for another equivalent or better test method, this change is considered administrative.

A.8 CTS Section 3.3.G Action 1.c requires the affected control rod to be declared inoperable. Once declared inoperable, the CTS Section 3.3.C Actions for an inoperable control rod are required to be taken. The revised TS LCO 3.3.C Actions for an inoperable control rod contain requirements to insert and disarm, as well as a shutdown requirement if the actions are not performed (i.e., revised TS LCO 3.3.C Actions 3.a and 3.b). The revised TS LCO 3.3.G Actions for inoperable accumulators do not need to repeat the revised TS LCO 3.3.C Actions to insert and disarm, or shutdown the unit if the inoperable control rod is not inserted and disarmed. Therefore CTS Section 3.3.G Actions 1.c.2 and 1.d have been deleted. Since this change is a presentation preference only, it is considered administrative.

Revised TS Section 3.3.G (Change # 7) - TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The revised TS LCO 3.3.G Action 1 for an inoperable control rod accumulator only provides an 8 hour allowance to essentially restore the inoperable accumulator if the reactor pressure is sufficiently high to support control rod insertion. CTS Section 3.3.G Action 1.a allows eight hours to restore the inoperable accumulator regardless of the reactor pressure. At reduced reactor pressures, control rods may not insert on a scram signal unless the associated accumulator is OPERABLE. Given the allowances in the proposed TS LCOs 3.3.C and 3.3.D for number and distribution of inoperable and slow control rods, an additional control rod failing to scram due to inoperable accumulator and low reactor pressure for up to eight hours without compensatory action is not justified. Therefore, revised TS LCO 3.3.G Action 1 applies to one inoperable accumulator at sufficiently high reactor pressures. Revised TS LCO 3.3.G Action 1.c applies to one or more inoperable accumulators at lower reactor pressures. At low reactor pressures, only one hour will be provided to restore the inoperable accumulator(s) prior to requiring the associated control rod(s) to be declared inoperable. In addition, charging water header pressure must be ≥940 psig during this one hour, or a reactor scram will be required (i.e., revised TS LCO 3.3.G Action 1.d).

Revised TS Section 3.3.G (Change # 7) - TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 CTS Section 3.3.G Action 1.a.2) requires a control rod to be declared inoperable within eight hours when its associated accumulator is inoperable. An inoperable control rod accumulator affects the associated control rod scram time. However, at sufficiently high reactor pressure, the accumulators only provide a portion of the scram force. With this high reactor pressure, the control rod will scram even without the associated accumulator, although probably not within the required scram times. Therefore, the option to declare a control rod with an inoperable accumulator "slow" when reactor pressure is sufficient is proposed (i.e., revised TS LCO 3.3.G Action 1.a.i) in lieu of declaring the control rod inoperable. Since CTS Section 3.3.G Action 1.a.2) to declare the control rod inoperable allows the control rod to remain withdrawn and not disarmed, revised TS LCO 3.3.G Required Action 1.a.i to declare the control rod "slow" is essentially equivalent. The proposed limits and allowances for numbers and distribution of inoperable and slow control rods, found in revised TS LCO 3.3.C and revised TS LCO 3.3.D, respectively, are appropriately applied to control rods with inoperable accumulators whether declared inoperable or slow. The option for declaring the control rod with an inoperable accumulator "slow" is restricted (i.e., by a Note to revised TS LCO 3.3.G Action 1.a.i and 1.b.ii) to control rods not previously known to be slow. This restriction limits the flexibility to control rods not otherwise known to have an impaired scram capability.

Additionally, with more than one accumulator inoperable, revised TS LCO 3.3.G Actions 1.b and 1.c provide actions similar to revised TS LCO 3.3.G Action 1.a, instead of the CTS Section 3.3.G Action 1.c requirement to declare the associated control rod inoperable immediately. The requirement to declare the associated control rod inoperable is maintained (i.e., revised TS LCO 3.3.G Action 1.b.ii and 1.c.ii), as well as an option to declare the associated control rod "slow" (i.e., revised TS LCO 3.3.G Action 1.b.ii). This added option is only allowed, however, when a sufficiently high reactor pressure exists, since at high reactor pressure there is adequate pressure to scram the rods, even with the accumulator inoperable. The requirement for declaration of control rods as slow, as described in the paragraph above, or inoperable, is limited to one hour in revised TS LCO 3.3.G Action 1.b.ii, and 1.c.ii2, as opposed to the current immediate declaration of inoperability in CTS Section 3.3.G Action 1.c. This provides a reasonable time to attempt investigation and restoration of the inoperable accumulator and is sufficiently short such that it does not increase the risk significance of an Anticipated Transient Without Scram (ATWS) event. Furthermore, the one hour will only be allowed provided the control rod drive header pressure alone is sufficient to insert control rods if a scram is required (i.e., revised TS LCO 3.3.G Actions 1.b.i, 1.c.i, and 1.d).

L.2 CTS Section 3.3.G Action 1.c.1) for inoperable scram accumulators applies to all reactor pressure situations, whether normal operating pressure or zero pressure. These two extremes represent significant differences in whether or not a control rod with an inoperable accumulator will scram. Revised TS LCO 3.3.G reflects this difference and presents actions more appropriate to the actual plant conditions, and, in one instance, includes more restrictive actions (i.e., M.1 above).

CTS Section 3.3.G Action 1.c.1) is intended to identify the situation where additional scram accumulators and eventually all accumulators would be expected to become inoperable. Identification of this sort of common cause is significant in ensuring continued plant safety. In the event reactor pressure is too low, where the control rod with an inoperable accumulator may not scram, it is imperative that immediate action be taken if the charging pressure to all accumulators is lost. This requirement is maintained essentially consistent in revised TS LCO 3.3.G Action 1.c.

However, in the event reactor pressure is sufficiently high (i.e., where the control rod will scram even without the associated accumulator), 20 minutes is proposed in revised TS LCO 3.3.G Action 1.b.1 to ensure control rod accumulator charging water pressure is adequate to support maintaining the remaining accumulators OPERABLE. This 20 minutes allows an appropriate time to attempt restoration of charging pressure if it should be lost. This proposed action is deemed more appropriate than the CTS Section 3.3.G Action 1.c.1) requirement to initiate an immediate reactor scram by placing the reactor mode switch in the shutdown position. The most likely cause of the loss of charging pressure is a trip of the operating CRD pump. Restart of this pump or of the spare CRD pump would restore charging pressure and avoid the plant transient caused by the immediate scram. Since control rod scram capability remains viable solely from the operating reactor pressure, and the most likely result of the 20 minute allowance of revised TS LCO 3.3.G Action 1.b.i is expected to be restoration of charging pressure, upon which time inoperable control rods could be manually inserted and disarmed, operation returned to normal, and a scram transient avoided, the proposed change is deemed acceptable.

4. RWM (change #10). The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the CRDA analysis are not violated. As shown in References I.4 and I.5, when thermal power is greater than 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA.

5. TLHGR and APRM Gain and Setpoint (change #12 and ITS #15). The revisions to add MFLPD/FRTP to the TS ensures that the equivalent level of TLHGR protection is provided for the GE fuel as is currently provided by FDLRC for the SPC fuel. MFLPD represents the maximum fraction of the steady state LHGR limit that exists at a given time in the reactor core. By maintaining the value of this fraction less than one, when adjusted for current reactor power, it is ensured that the steady state LHGR limit is not violated at all powers less than or equal to 100%. This, in turn, ensures that the transient LHGR limit is not violated during AOOs. This is further described in Reference I.3.

6. Control Rod Scram Times (change #14). The revision to add required scram times for GE analyzed cores will maintain all fuel-related parameters within the required thermal limits during all analyzed transients and accidents. The proposed scram times are different from those for SPC analyzed cores because of the difference in calculational approach. Whereas SPC methodology sets scram times that ensure an adequate scram reactivity insertion rate if no more than 12 rods are slow, GE's approach is to set slower scram times and then use actual average rod scram times to calculate the actual scram reactivity. This information is then used to set cycle-specific operating limits.

7. COLR (changes #13 and #16). The references added are either NRC approved methodologies or are expected to be approved by the NRC during the review process for this proposed change.

8. Definitions (change #17). The definitions added represent purely administrative changes.

G. IMPACT ON PREVIOUS SUBMITTALS

The proposed changes affect our previous request for CTS conversion to ITS, which was submitted to the NRC by Reference I.1. As previously described, the marked-up pages of both CTS and ITS have been submitted with this amendment request in Attachments B-1 and B-2. We are requesting NRC approval for the changes to the version of TS that is in effect (i.e., CTS or ITS) at the time this amendment request is approved.

We have reviewed the proposed changes and have determined that there is no impact on any other previous submittals.

H. SCHEDULE REQUIREMENTS

We request approval of the proposed changes prior to September 14, 2001, in order to support core reload with GE fuel during the DNPS Unit 2 refueling outage which is currently scheduled to begin on October 20, 2001.

I. REFERENCES

1. Letter from R.M. Krich (ComEd) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000.

2. Letter from G.A. Watford (GE) to U.S. NRC, "GEXL96 Correlation for ATRIUM 9B Fuel," NEDC-32981P, dated September 26, 2000

3. Letter from G. A. Watford (GE) to U.S. NRC, "Revision 14 to GESTAR II and Its United States Supplement," dated June 9, 2000

4. a. Letter from P. L. Piet (ComEd) to U.S. NRC, "Topical Report for Neutronics Methods for BWR Reload Design Using CASMO/MICROBURN," dated December 31, 1991
   b. Letter from P. L. Piet (ComEd) to U.S. NRC, "Topical Report for Neutronics Methods for BWR Reload Design Using CASMO/MICROBURN," Supplement 1, dated March 24, 1992
   c. Letter from P. L. Piet (ComEd) to U.S. NRC, "Topical Report for Neutronics Methods for BWR Reload Design Using CASMO/MICROBURN," Supplement 2, dated May 22, 1992
   d. Letter from C.P. Patel (U.S. NRC) to ComEd, "Commonwealth Edison Company Topical Report NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," dated March 22, 1993.

5. Letter from T.A. Pickens (BWROG) to NRC, "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," August 15, 1986

6. NUREG-1433, "Standard Technical Specifications for General Electric Plants, BWR 4," revision 1

Attachment B-1
Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3

MARKED-UP CURRENT TECHNICAL SPECIFICATIONS PAGES FOR PROPOSED CHANGES

REVISED PAGES

Revised Marked-Up Pages

I
VI
1-3
1-4
2-1
Insert for page 2-1
3/4.3-1
3/4.3-3
3/4.3-4
3/4.3-5
3/4.3-6
3/4.3-7
3/4.3-8
3/4.3-9
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3/4.3-12
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3/4.3-18
3/4.6-1
3/4.11-2
6-16

Revised Typed Pages

3/4.3-3
3/4.3-4
3/4.3-5
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DEFINITIONS

SECTION PAGE

Section 1 DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

FRACTION OF LIMITING POWER DENSITY (FLPD)

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

FRACTION OF RATED THERMAL POWER (FRTP)

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

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

FUEL DESIGN LIMITING RATIO (FDLRX)

The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum.

FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC)

The FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) shall be the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)

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

LINEAR HEAT GENERATION RATE (LHGR)

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

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)

A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from as close to the sensor as practicable up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total system steps so that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

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

MINIMUM CRITICAL POWER RATIO (MCPR)

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than the following:

Unit 2: 1.09 for cycle exposures less than or equal to 13,800 MWd/MTU and 1.12 for cycle exposures greater than 13,800 MWd/MTU, and

Unit 3: 1.10

with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. ~~During single recirculation loop operation, the MCPR limit shall be increased by 0.01~~. (see insert attached)

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2

ACTION:

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

Insert to Dresden TS 2.1.B

During single recirculation loop operation, the MCPR limit shall not be less than the following:

Unit 2: 1.10 for cycle exposures less than or equal to 13,800 MWd/MTU and 1.13 for cycle exposures greater than or equal to 13,800 MWd/MTU, and

Unit 3: 1.11

with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.38% Δk/k with the highest worth control rod analytically determined, or

2. 0.28% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage. 72 — LCO 3.3.D 2.4

2. Within 24 hours after detection of a withdrawn control rod that is ~~immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable~~ stuck. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the ~~immovable or unscrammable~~ stuck control rod. LCO 3.3.C A.5

3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE. SR 4.3.C.2 SR 4.3.C.3

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

A.4

ACTION: add proposed Required Action 1 Note

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable: A.5

   a. Within ~~one~~ 4 hour: L.1 ACTION 4

      1) Verify that the inoperable control rod, ~~if withdrawn~~, is separated from all other inoperable ~~withdrawn~~ control rods by at least two control cells in all directions. 2 M.2 L.2

      2) Disarm the associated control rod drive (CRD) ~~directional control valves~~[a] ~~either:~~ LA.1 Action 1b

         a) ~~Electrically~~, or M3

         b) ~~Hydraulically by closing the drive water and exhaust water isolation valves.~~

   add proposed ACTION 2 M.3

   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours. ACTION 5

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods ~~not required to have their directional control valves disarmed electrically or hydraulically~~ shall be demonstrated OPERABLE by ~~moving~~ each control rod at least one notch: A.6 M.5

   a. At least once per ~~7~~ 31 days, and L.3

   b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable. Action 1.d

2. ~~All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.~~ A.7

add proposed Required Action 1.a M.1

---

a ~~May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~ A.8

REVISED LCO 3.3. C

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

Action 1.c

c. Comply with Surveillance Requirement 4.3.A.2 within ~~24~~ 72 (not 70) hours ~~or be in HOT~~ L.4

ACTION 5 SHUTDOWN within the next 12 hours.

ACTION 3

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

L.1 ACTION 4

a. If the inoperable control rod(s) is withdrawn, within ~~one~~ 4 hour:

Add proposed Note to Action 4 L.1

1) Verify that the inoperable ~~withdrawn~~ control rod(s) is separated from all other inoperable ~~withdrawn~~ control rods by at least two control cells in all directions, and M.2

add proposed Action footnote (a) A.4

L.5 Action 3.a Action 3.b

2) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the inoperable withdrawn control rod(s) at least one notch by drive water pressure within the normal operating range.[b] M.6

b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated CRD directional control valves[a] either: A.1

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

---

b The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable. M.6

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.8

A.1

REVISED LCO 3.3.C

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

Action 3.b c. If the inoperable control rod(s) is fully inserted, within one hour disarm the associated directional control valves[a] either: (4) LS CRD LA.1

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

ACTION 5 3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 5 4. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.8

REVISED LCO 3.3.C

A.1

A.10

<general reorganization>

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

SR 4.3.C.4

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, ~~based on de-energization of the scram pilot valve solenoids as time zero,~~ shall not exceed 7 seconds.

A.11

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

LCO 3.3.C ACTION 1 or ACTION 3

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and

2. ~~When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.~~

~~With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.~~

L.6

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:

   a. following CORE ALTERATION(s), or

   b. after a reactor shutdown that is greater than 120 days.

2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

<See LCO 3.3.D>

add proposed SR 4.3.C.4

A.12

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

Page 4 of 8

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

Surv. Requirement 4.3.C.5

A.13

All control rods shall be coupled to their drive mechanisms.

Surveillance Requirement 4.3.C.5

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

L retained in specification 3.3.H

ACTION:

L.5

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

L.8

a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

A.14

L.10

(1) Observing any indicated response of the nuclear instrumentation, and

L.9

(2) Demonstrating that the control rod will not go to the overtravel position.

L.8

b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:

ACTION 3

CRD

LA.1

(1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by verifying that the control rod drive does not go to the overtravel position:

1. Deleted.

2. Anytime the control rod is withdrawn to the "Full out" position, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

---

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

Retained in Specification 3.3.H

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

A.8

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

2) Hydraulically by closing the drive water and exhaust water isolation valves.

LA.1

ACTION 5

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

Retained in Specification 3.3.H

Retained in specification 3.3.H

L.7

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

REACTIVITY CONTROL

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE. A.15

SR 4.3.C.1

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a]

Retained in specification 3.3.I

ACTION:

ACTION 3

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either: L.5

   a. Determine the position of the control rod by an alternate method, or

   ~~b. Move the control rod to a position with an OPERABLE position indicator, or~~ LA.2

   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either: CRD LA.1

      1) ~~Electrically, or~~

      2) ~~Hydraulically by closing the drive water and exhaust water isolation valves.~~

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.

2. ~~That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.~~ L.11

3. ~~Deleted.~~

Retained in specification 3.3.I

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status. A.8

REVISED LCO 3.3.C

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

ACTION S

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

a. Move the control rod to a position with an OPERABLE position indicator, or

b. Fully insert the control rod.

Retained in specification 3.3.I

Retained in specification 3.3.I

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

REVISED LCO 3.3.D

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

NOTE to Surveillance Requirements

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and

2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

<See LCO 3.3.D>

4.3 - SURVEILLANCE REQUIREMENTS

M.1

D. Maximum Scram Insertion Times

SR 4.3.D.1, SR 4.3.D.2, SR 4.3.D.4

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

or equal to

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:

   SR 4.3.D.4 a. following CORE ALTERATION(s), or — L.1

   SR 4.3.D.1 b. after a reactor shutdown that is greater than 120 days.

2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and — SR 4.3.D.4

3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION. — SR 4.3.D.2 — LA.1

add proposed SR 4.3.D.3 — M.1

A.2

SR 4.3.D.4

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

Page 1 of 3

REVISED LCO 3.3.D

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

E. Average Scram Insertion Times

Footnote (a) to Table 3.3.D-1

The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

M.2

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

E. Average Scram Insertion Times

SR 4.3.D.1 SR 4.3.D.2 SR 4.3.D.4

The control rod average scram times shall be demonstrated by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

add proposed LCO 3.3.D and Table 3.3.D-1.

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

Foot note (a) to Table 3.3.D-1

~~The average of the scram insertion times, from the fully withdrawn position, for the three fastest control rods of all groups of four control rods in a two-by-two array,~~ based on de-energization of the scram pilot valve solenoids as time zero, ~~shall not exceed any of the following:~~

~~% Inserted From Fully Withdrawn~~	~~Avg. Scram Insertion Times (sec)~~
~~5~~	~~0.398~~
~~20~~	~~0.954~~
~~50~~	~~2.120~~
~~90~~	~~3.800~~

M.2

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

LCO 3.3.D ACTION

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

1. ~~Declare the control rods exceeding the above average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and~~
2. ~~When operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of power operation.~~

M.2

~~With the provisions of the ACTION(s) above not met,~~ be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

SR 4.3.D.1 SR 4.3.D.2 SR 4.3.D.4

All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

add proposed LCO 3.3.D and Table 3.3.D-1

A.1

REVISED LCO 3.3.G

3.3 - LIMITING CONDITIONS FOR OPERATION

LCO 3.3.G **G. Control Rod Scram Accumulators**

All control rod scram accumulators shall be OPERABLE.

APPLICABILITY

OPERATIONAL MODE(s) 1, 2 and 5[a]

ACTION:

1. In OPERATIONAL MODE 1 or 2.

ACTION 1a a. With one control rod scram accumulator inoperable, within 8 hours: with reactor steam dome pressure ≥ 900 psig M.1

1) Restore the inoperable accumulator to OPERABLE status, or A.4

add proposed Required Action 1.a.i

Action 1.a.i 2) Declare the control rod associated with the inoperable accumulator inoperable. L.1

b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours. A.5

M.1

ACTION 1.b.ii
ACTION 1.c.ii c. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and: add proposed Action 1.b.i

within 1 hour L.1

4.3 - SURVEILLANCE REQUIREMENTS

G. Control Rod Scram Accumulators

Each control rod scram accumulator shall be determined OPERABLE at least once per 7 days by verifying that the indicated pressure is ≥940 psig unless the control rod is fully inserted and disarmed, or scrammed. A.6

---

a In OPERATIONAL MODE 5, this specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

Page 1 of 3

A.1

REVISED LCO 3.3.G

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

Action 1.d Note

1) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch. With no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

Action 1.c

ACTION 1.B

L.2

A.7

2) ~~Fully insert the inoperable control rods and disarm the associated directional control valves[b] either:~~

A.8

~~a) Electrically, or~~

~~b) Hydraulically by closing the drive water and exhaust water isolation valves.~~

~~d. With the provisions of ACTION 1.c.2 above not met, be in at least HOT SHUTDOWN within 12 hours.~~

ACTION 2

2. In OPERATIONAL MODE 5[a]:

a. With one withdrawn control rod with its associated scram accumulator inoperable, fully insert the affected control rod and disarm the associated directional control valves[b] within one hour, either:

a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

Page 2 of 3

A.1

Revised LCO 3.3.G

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

1) Electrically, or

ACTION 2

2) Hydraulically by closing the drive water and exhaust water, isolation valves.

b. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

10

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.
2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.
3. In OPERATIONAL MODE 1 prior to reducing THERMAL POWER below 20% of RATED THERMAL POWER:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.

10

---

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

3.6 - LIMITING CONDITIONS FOR OPERATION

A. Recirculation Loops

Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With only one reactor coolant system recirculation loop in operation, within 24 hours either, restore both loops to operation or:

   a. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.B, and

   b. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Operating Limit by 0.01 per Specification 3.11.C, and to the MCPR Operating Limit specified in the COLR, and

   c. Reduce the Average Power Range Monitor (APRM) Flow Biased Neutron Flux Scram and Rod Block and Rod Block Monitor Trip Setpoints to those applicable to single recirculation loop operation per Specifications 2.2.A and 3.2.E.

   d. Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) to single loop operation limits as specified in the CORE OPERATING LIMITS REPORT (COLR).

4.6 - SURVEILLANCE REQUIREMENTS

A. Recirculation Loops

Each pump motor generator (MG) set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with the overspeed setpoints specified in the CORE OPERATING LIMITS REPORT at least once per 18 months.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. TRANSIENT LINEAR HEAT GENERATION RATE

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be maintained such that the FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) is less than or equal to 1.0. Where FDLRC is equal to:

MFLPD / FRTP (b)

e

$$\frac{(LHGR)\ (1.2)}{(TLHGR)\ (FRTP)}$$

e

APPLICABILITY:

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

ACTION:

MFLPD / FRT (b)

With FDLRC greater than 1.0, initiate corrective ACTION within 15 minutes and within 6 hours either:

MFLPD / FRTP (b)

1. Restore FDLRC to less than or equal to 1.0, or
2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by ~~1/FDLRC~~ or
3. ~~Adjust[a] each APRM gain such that the APRM readings are ≥100% times the FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.~~ e

1/FDLRC or FRTP/MFLPD

Adjust (a) each APRM gain such that the APRM readings are ≥ 100% times the FRACTION OF RATED THERMAL POWER (FRTP) times the greater of FDLRC or MFLPD/FRTP. (b)

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

4.11 - SURVEILLANCE REQUIREMENTS

B. TRANSIENT LINEAR HEAT GENERATION RATE

MFLPD/FRTP (b)

The value of FDLRC shall be verified:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with FDLRC greater than or equal to 1.0.
4. The provisions of Specification 4.0.D are not applicable.

MFLPD / FRTP (b)

---

a Provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

b For SPC fuel, FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) is substituted for MFLPD/FRTP. Adjustments are based on the lowest APRM setpoint or highest APRM reading resulting from the two limits.

ADMINISTRATIVE CONTROLS

(12) ANF-1125 (P)(A), ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

(13) EMF-85-74(P), RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model, Supplement 1 (P)(A) and Supplement 2 (P)(A), Siemens Power Corporation, February 1998.

c. The core operating limits report 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 of supplements thereto shall be provided on 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]

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

(15) NEDC-32981P, GEXL96 Correlation for ATRIUM 9B Fuel, September 2000.

3.3 – Limiting Conditions for Operation

C. Control Rod OPERABILITY

Each control rod shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one withdrawn control rod stuck[a]:
   a. Immediately verify that stuck control rod separation criteria are met, and
   b. Within 2 hours, disarm the associated control rod drive (CRD), and
   c. Within 72 hours, perform Surveillance Requirement 4.3.A.2, and
   d. Within 24 hours of discovery of one withdrawn stuck control rod concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM, perform Surveillance Requirement 4.3.C.2 and Surveillance Requirement 4.3.C.3 for each withdrawn OPERABLE control rod.
2. With two or more withdrawn control rods stuck, be in at least HOT SHUTDOWN within 12 hours.
3. With one or more control rods inoperable for reasons other than being stuck in the withdrawn position:
   a. Within 3 hours, fully insert the inoperable control rod(s) [b], and
   b. Within the next 1 hour, disarm the associated CRD(s).

4.3 – Surveillance Requirements

C. Control Rod OPERABILITY

1. The position of each control rod shall be determined at least once per 24 hours.
2. Insert each fully withdrawn control rod at least one notch at least once per 7 days. [c]
3. Insert each partially withdrawn control rod at least one notch at least once per 31 days. [d]
4. Verify each control rod scram time from fully withdrawn to 90% insertion is $\leq$ 7 seconds, in accordance with the frequencies specified in Surveillance Requirements 4.3.D.1, 4.3.D.2, 4.3.D.3, 4.3.D.4, and 4.3.D.5.
5. Verfiy each control rod does not go to the withdrawn overtravel position each time the control rod is withdrawn to the "full out" position and prior to declaring the control rod OPERABLE after work on control rod or CRD system that could affect coupling.

(a) The rod worth minimizer (RWM) may be bypassed as allowed by Specification 3.3.L to allow continued operation.
(b) The RWM may be bypassed as allowed by Specification 3.3.L to allow insertion of inoperable control rod and continued operation.
(c) Not required to be performed until 7 days after the control rod is withdrawn and THERMAL POWER is greater than the low power setpoint of the RWM.
(d) Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the low power setpoint of the RWM.

3.3 – Limiting Conditions for Operation

4. With two or more inoperable control rods not in compliance with analyzed rod position sequence and not separated by two or more OPERABLE control rods [(e)]:

    a. Within 4 hours, restore compliance with analyzed rod sequence or restore the control rod to OPERABLE status.

5. With the required provisions of ACTION 1, 3, or 4 not met, or with nine or more control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

4.3 – Surveillance Requirements

(e) Not applicable when THERMAL POWER > 10% RTP.

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3.3 – Limiting Conditions For Operation

D. Control Rod Scram Times

1. No more than 12 OPERABLE control rods shall be "slow," in accordance with Table 3.3.D-1; and

2. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTIONS:

With the LCO requirements not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 – Surveillance Requirements

D. Control Rod Scram Times [a]

1. Verify each control rod scram time is within the limits of Table 3.3.D-1 with reactor steam dome pressure $\geq$ 800 psig prior to exceeding 40% RTP after each reactor shutdown $\geq$ 120 days.

2. Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.3.D-1 with reactor steam dome pressure $\geq$ 800 psig, at least once per 120 days of cumulative operation in OPERATIONAL MODE 1.

3. Verify each affected control rod scram time is within the limits of Table 3.3.D-1 with any reactor steam dome pressure prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect scram time.

4. Verify each affected control rod scram time is within the limits of Table 3.3.D-1 with reactor steam dome pressure $\geq$ 800 psig prior to exceeding 40% RTP after fuel movement within the affected core cell

5. Verify each affected control rod scram time is within the limits of Table 3.3.D-1 with reactor steam dome pressure $\geq$ 800 psig prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time.

(a) During single control rod scram time surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

Table 3.3.D-1
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this table are considered "slow."

2. Enter applicable ACTIONS of Specification 3.3.C,. "Control Rod Operability," for control rods with scram times > 7 seconds to 90% insertion. These control rods are inoperable, in accordance with Surveillance Requirement 4.3.C.4, and are not considered "slow."

Percent Insertion	Scram Times [a][b] (seconds) When Reactor Steam Dome Pressure $\geq$ 800 psig For SPC Analyzed Cores	Scram Times [a][b] (seconds) When Reactor Steam Dome Pressure $\geq$ 800 psig For GE Analyzed Cores
5	0.36	0.48
20	0.84	0.89
50	1.86	1.98
90	3.25	3.44

(a) Maximum scram times from fully withdrawn position based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure when < 800 psig are within established limits.

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3.3 – Limiting Conditions for Operation

G. Control Rod Scram Accumulators

Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY

OPERATIONAL MODE(s) 1, 2 and 5[a].

ACTIONS:

1. In OPERATIONAL MODE 1 OR 2:
   a. With one control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig:

      i. Within 8 hours, declare the associated control rod scram time "slow," [b] or declare the associated control rod inoperable.

   b. With two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig:

      i. Within 20 minutes from discovery of two or more inoperable accumulators with reactor steam dome pressure ≥ 900 psig concurrent with charging water header pressure < 940 psig, restore charging water header pressure to ≥ 940 psig, and

      ii. Within 1 hour, declare the associated control rod scram time "slow," [b] or declare the associated control rod inoperable.

(a) In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J

4.3 – Surveillance Requirements

G. Control Rod Scram Accumulators

1. Verify each control rod scram accumulator pressure is ≥ 940 psig at least once per 7 days.

(b) Only applicable if the associated control rod scram time was within the limits of Table 3.3.D-1 during the last scram time surveillance.

3.3 – Limiting Conditions for Operation

G. Control Rod Scram Accumulators

c. With one or more control rod scram accumulators inoperable with steam dome pressure < 900 psig:

i. Immediately upon discovery of charging water header pressure < 940 psig, verify all control rods associated with inoperable accumulators are fully inserted, and

ii. Within 1 hour, declare the associated control rod inoperable.

d. With the required provisions of ACTION 1.b.i or 1.c.i not met, immediately place the reactor mode switch in the shutdown position. [c]

2. In OPERATIONAL MODE 5[a]:

a. With one withdrawn control rod and its associated scram accumulator inoperable, fully insert and disarm the affected control rod within one hour. [d]

b. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the shutdown position.

(a) In OPERATIONAL MODE 5, this specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(c) Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

4.3 – Surveillance Requirements

(d) May be armed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status

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3.3 – Limiting Conditions for Operation

H. Control Rod Drive Coupling

All control rod drives shall be coupled to their drive mechanisms

APPLICABILITY:

OPERATIONAL MODE 5 [a]

ACTION:

With a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

b. If recoupling is not accomplished, declare the control rod inoperable, fully insert and disarm the control rod.

(a) In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

4.3 – Surveillance Requirements

H. Control Rod Drive Coupling

Each affected control rod drive shall be demonstrated to be coupled to its drive mechanism by verifying that the control rod does not go to its overtravel position:

1. Anytime the control rod is withdrawn to the "full out" position, and
2. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

3.3 – Limiting Conditions for Operation

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE 5 [(a)]

ACTION:

1. With a withdrawn control rod position indicator inoperable:

   a. Move the control rod to a position with an OPERABLE position indicator, or

   b. Fully insert the control rod.

4.3 – Surveillance Requirements

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying at least once per 24 hours that the position of each control rod is indicated.

(a) In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

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Attachment B-2
Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3

MARKED-UP IMPROVED TECHNICAL SPECIFICATIONS PAGES FOR PROPOSED CHANGES

REVISED MARKED-UP PAGES

1.1-4
3.1.4-3
3.2.4-1
3.2.4-2
5.6-4

REVISED TYPED PAGES

1.1-4
1.1-5
1.1-6
1.1-7
3.1.4-3
3.2.4-1
3.2.4-2
5.6-4
5.6-5

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The MFLPD shall be the largest value of the fraction of limiting power density (FLPD) in the core. The FLPD shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWt.

(continued)

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to 90% insertion. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

PERCENT INSERTION	SCRAM TIMES(a)(b) (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig for SPC analyzed cores	SCRAM TIMES(a)(b) (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig for GE analyzed cores
5	0.36	0.48
20	0.84	0.89
50	1.86	1.98
90	3.25	3.44

(a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure when < 800 psig are within established limits.

SCRAM TIMES (a)(b) (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig for GE analyzed cores

INFORMATION ONLY

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. FDLRC and the ratio of MFLPD to Fraction of RTP (FRTP) shall be less than or equal to 1.0; or

b. Each required APRM Flow Biased Neutron Flux — High Function Allowable Value shall be modified by the lesser of 1/FDLRC or FRTP/MFLPD; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times the ~~Fraction of RTP (FRTP) times FDLRC.~~ higher of FRTP times FDLRC or of MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	------------------NOTE------------------ Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. -- Verify FDLRC ~~is~~ are within limits. and the ratio of MFLPD to FRTP	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter
SR 3.2.4.2	------------------NOTE------------------ Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. -- less than or equal to the lesser of Verify each required: a. APRM Flow Biased Neutron Flux—High Function Allowable Value is modified by 1/FDLRC or FRTP/MFLPD; or b. APRM gain is adjusted such that the APRM reading is ≥ 100% times the higher of FRTP times FDLRC or of MFLPD.	12 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

12. ANF-1125(P)(A), ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

13. EMF-85-74(P), RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model, Supplement 1 (P)(A) and Supplement 2 (P)(A), Siemens Power Corporation, February 1998.

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

15. NEDC-32981P, "GEXL96 Correlation for ATRIUM 9B Fuel," September 2000.

The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The FDLRC and the ratio of MFLPD to FRTP is required to be calculated and compared to 1.0 or APRM gain adjusted or APRM Flow Biased Neutron Flux—High Function Allowable Value modified to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the FDLRC and the ratio of MFLPD to FRTP and, assuming either exceeds 1.0, determine the appropriate APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or Flow Biased Neutron Flux—High Function circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1), MCPR (LCO 3.2.2), and LHGR (LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to APLHGR, MCPR, and LHGR operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 is required when FDLRC or the ratio of MFLPD to FRTP is greater than 1.0, because more rapid changes in power distribution are typically expected.

REFERENCES

1. UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10.

2. UFSAR, Chapter 15.

Attachment F

Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3
CONVENTIONS USED FOR MARK-UPS OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

The annotated CTS control rod specifications pages are marked with sequentially numbered boxes which provide a cross-reference to Attachment A, Section F, "Safety Analysis of the Proposed Changes." The revised Specification is noted on the top right corner of each CTS page, identifying the Specification where the revised requirements are located. Items on the CTS page that are located in one or more revised locations or sections have the appropriate location(s) noted adjacent to the items. When the revised requirement differs from the current requirement, the current requirement being revised is annotated with an alpha-numeric designator. This designator relates to the appropriate subsection of the safety analysis. Each safety analysis subsection provides a justification for the proposed change..

The alpha-numeric designator is based on the category of the change and a sequential number within that category. The revisions are categorized as follows.

A ADMINISTRATIVE - associated with restructuring, interpretation, and complex rearranging of requirements, and other changes not substantially revising an existing requirement.

M TECHNICAL CHANGES - MORE RESTRICTIVE - changes resulting in added restrictions or eliminating flexibility.

L TECHNICAL CHANGES - LESS RESTRICTIVE - changes where requirements are relaxed, relocated, eliminated, or new flexibility is provided. There are two subcategories used in this revision:

LA changes consist of relocation of details out of the TS and into the Bases, Updated Final Safety Analysis Report, Quality Assurance Topical Report, or other plant controlled documents. Typically, this involves details of system design and function or procedural details on methods of conducting a surveillance.

L changes consist of relaxation or elimination of requirements.

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density (FLPD) in the core. The FLPD shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that

(continued)

1.1 Definitions

OPERABLE – OPERABILITY (continued)	are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact until the opening of the trip actuator. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

1.1 Definitions (continued)

TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[a] or Startup/Hot Standby	NA
3	Hot Shutdown[a]	Shutdown	> 212
4	Cold Shutdown[a]	Shutdown	≤ 212
5	Refueling[b]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to 90% insertion. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

PERCENT INSERTION	SCRAM TIMES(a)(b) (seconds) when REACTOR STEAM DOME PRESSURE $\geq$ 800 psig for SPC analyzed cores	SCRAM TIMES(a)(b) (seconds) when REACTOR STEAM DOME PRESSURE $\geq$ 800 psig for GE analyzed cores
5	0.36	0.48
20	0.84	0.89
50	1.86	1.98
90	3.25	3.44

(a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure when < 800 psig are within established limits.

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. FDLRC and the ratio of MFLPD to Fraction of RTP (FRTP) shall be less than or equal to 1.0; or

b. Each required APRM Flow Biased Neutron Flux—High Function Allowable Value shall be modified by the lesser of 1/FDLRC or FRTP/MFLPD; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times the higher of FRTP times FDLRC or of MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------------------NOTE------------------ Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. -- Verify FDLRC and the ratio of MFLPD to FRTP are within limits.	Once within 12 hours after $\geq$ 25% RTP <u>AND</u> 24 hours thereafter
SR 3.2.4.2	------------------NOTE------------------ Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. -- Verify each required: a. APRM Flow Biased Neutron Flux—High Function Allowable Value is modified by less than or equal to the lesser of 1/FDLRC or FRTP/MFLPD; or b. APRM gain is adjusted such that the APRM reading is $\geq$ 100% times the higher of FRTP times FDLRC or of MFLPD.	12 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

12. ANF-1125(P)(A), ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

13. EMF-85-74(P), RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model, Supplement 1 (P)(A) and Supplement 2 (P)(A), Siemens Power Corporation, February 1998.

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

15. NEDC-32981P, "GEXL96 Correlation for ATRIUM 9B Fuel," September 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.6 Reporting Requirements

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Attachment C
Proposed Change to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3
INFORMATION SUPPORTING A FINDING OF
NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), "Issuance of amendment," a proposed amendment to an operating license involves no significant hazards consideration 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.

Commonwealth Edison (ComEd) Company is proposing to modify various Technical Specifications (TS) for Dresden Nuclear Power Station (DNPS), Units 2 and 3 to support a change in fuel vendors from Siemens Power Corporation (SPC) to General Electric (GE). The revisions are proposed to both Current Technical Specifications (CTS) and our requested conversion to Improved Technical Specifications (ITS), which is currently being reviewed by the NRC. The proposed changes are briefly summarized as follows.

Proposed Changes to CTS

1. Administrative Changes. a) CTS Section 2.1.B, "Thermal Power, High Pressure and High Flow," is revised to remove the statement that the single loop operation Minimum Critical Power Ratio (MCPR) Safety Limit is 0.01 greater than the two loop operation MCPR Safety Limit. This requirement is replaced with the numerical values for the single loop operation MCPR Safety Limit. b) In CTS Section 3.6.A, "Recirculation Loops," the MCPR Safety and Operating limits are incorporated by reference. c) CTS Section 6.9.A.6.b, "Core Operating Limits Report," is revised to reflect the approved methodologies that will be used to determine the core operating limits. d) The definitions of the Maximum Fraction of Limiting Power Density (MFLPD), Fraction of Rated Thermal Power (FRTP), and Fraction of Limiting Power Density (FLPD) are added to Section 1.0, "Definitions."

2. Control Rod Operability and Scram Insertion Time Methodology. CTS Sections 3/4.3.C, "Control Rod Operability," 3/4.3.D, "Maximum Scram Insertion Times," 3/4.3.E, "Average Scram Insertion Times," 3/4.3.F, "Group Scram Insertion Times," 3/4.3.G, "Control Rod Scram Accumulators," 3/4.3.H, "Control Rod Coupling," and 3/4.3.I, "Control Rod Position Indication System," are revised to adopt the ITS methodology for control rod operability and scram insertion times. CTS reflects an analysis methodology based on limiting the average scram insertion time. ITS reflects an analysis methodology based on limiting the number of rods with slow insertion times.

3. Control Rod Scram Insertion Times. In addition to change #2 above, scram times are revised to add the required scram times for GE analyzed cores.

4. Rod Worth Minimizer (RWM). CTS Sections 3.3.L and 4.3.L, "Rod Worth Minimizer," are revised to reduce the thermal power limit at which the RWM shall be operable from 20% to 10%.

5. Transient Linear Heat Generation Rate (TLHGR). CTS Section 3.11.B, "Transient Linear Heat Generation Rate," is revised to add the ratio of MFLPD/FRTP to the TS. This ensures that the equivalent level of TLHGR protection is provided for GE fuel as is currently provided by the Fuel Design Limiting Ratio for Centerline (FDLRC) Melt for SPC fuel. The use of FDLRC for monitoring SPC fuel is retained.

Proposed Changes to ITS

1. Administrative Changes. a) TS Section 5.6.5.b, "Core Operating Limits Report," is revised to add references to the GE NRC approved methodologies that will be used for developing operating limits. b) The definition of MFLPD is added to Section 1.1, "Definitions."

2. Control Rod Scram Times. TS Table 3.1.4-1, "Control Rod Scram Times," is revised to add the required scram times for GE analyzed cores to the current requirements for SPC analyzed cores.

3. APRM Gain and Setpoint. In TS Section 3.2.4, "APRM Gain and Setpoint," the ratio of MFLPD/FRTP is added for monitoring GE fuel. The use of FDLRC for monitoring SPC fuel is retained.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below in two separate sections for CTS and ITS.

Proposed Changes to CTS

Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Evaluation of the effect on the probability of an accident previously evaluated:

1. Administrative Changes. The revisions to Current Technical Specifications (CTS) Sections 2.1.B, "Thermal Power, High Pressure and High Flow," and 3.6.A, "Recirculation Loops," regarding the Minimum Critical Power Ratio (MCPR) Safety Limit, the changes to CTS Section 6.9.A.6.b, "Core Operating Limits Report," and the changes to the definitions are administrative changes and will not affect the probability of an accident previously evaluated. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. Control Rod Operability and Scram Insertion Times Methodology. The changes to CTS Sections 3/4.3.C, "Control Rod Operability," 3/4.3.D, "Maximum Scram Insertion Times," 3/4.3.E, "Average Scram Insertion Times," 3/4.3.F, "Group Scram Insertion Times," 3/4.3.G, "Control Rod Scram Accumulators," 3/4.3.H,

"Control Rod Coupling," and 3/4.3.I, "Control Rod Position Indication System," revise the methodology for determining rod operability and control rod scram time requirements for operation. These changes do not physically alter plant systems, structures or components and therefore do not affect the probability of an accident previously evaluated.

3. Control Rod Scram Times. The addition of required scram times for General Electric (GE) analyzed cores does not physically alter plant systems, structures or components and therefore does not affect the probability of an accident previously evaluated.

4. Rod Worth Minimizer (RWM). The revision to CTS Section 3/4.3.L , "Rod Worth Minimizer," lowers the power level at which the analyzed rod position sequence must be followed. This change does not affect plant systems structures, or components. Because there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gram fuel design limit, the probability of an accident is not increased.

5. Transient Linear Heat Generation Rate (TLHGR). The revisions to CTS Section 3.11.B, "Transient Linear Heat Generation Rate," add fuel thermal limits that are monitored to ensure the TLHGR is not violated. These changes do not physically alter plant systems, structures or components and therefore do not affect the probability of an accident previously evaluated.

Evaluation of the effect on the consequences of an accident previously evaluated.

1. Administrative Changes. The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, the changes to CTS Section 6.9.A.6.b regarding the COLR, and the changes to the definitions are administrative changes and will not affect the consequences of an accident previously evaluated. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.
2. Control Rod Operability and Scram Insertion Times Methodology. The revisions to CTS Sections 3/4.3.C, 3/4.3.D, 3/4.3.E, 3/4.3.F, 3/4.3.G, 3/4.3.H, and 3/4.3.I are made to ensure the appropriate scram times are reflected in the TS for GE methodology. The scram timing requirements ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. CTS methods ensure this by limiting scram times for individual rods, the average scram time, and local scram times (i.e., a four control rod group). The proposed revisions, based on the Improved Technical Specification (ITS) methods, ensure this by limiting the scram times for individual rods, the number of slow rods, and the number of adjacent slow rods. Each of these methods ensure equivalent protection of the assumed reactivity insertion rate. Therefore, there is no change to the consequences of a previously evaluated accident or transient.

   In addition, numerous changes to the control rod operability and scram timing requirements were made to reflect the ITS approach to these requirements. These revisions consist of administrative changes, more restrictive changes, and less restrictive changes. The discussion of each of these categories is provided below.

Administrative changes. These consist of restructuring, interpretation, rearranging of requirements, and other changes not substantially revising an existing requirement. Therefore, these changes do not affect the consequences of an accident previously evaluated.

More restrictive changes. These consist of changes resulting in added restrictions or eliminating flexibility. The more restrictive requirements continue to ensure that process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve an increase in the consequences of an accident previously evaluated.

Less restrictive changes. The less restrictive changes involve increasing the time to complete actions, increasing the time intervals between required surveillances, and deleting or revising the applicability of certain actions. The time to complete actions and the surveillance frequencies are not assumed in the analysis of the consequences of any accidents previously evaluated, and therefore cannot increase the consequences of such accidents. The deleted or revised actions are not assumed in the safety analyses for any evaluated accidents. The revised scram timing methods will result in operating thermal limits that will maintain the identical safety limits. Thus, the consequences of the evaluated accidents will not increase.

3. Control Rod Scram Times. Cycle-specific analyses that use the GE methodology scram times will meet all of the same safety limit acceptance criteria. Additionally, for the non-cycle specific events in the Updated Final Safety Analysis Report (UFSAR), GE has determined that there is negligible impact on results of events which are not analyzed on a cycle-specific basis. Therefore, there is no change to the consequences of a previously evaluated accident or transient.

4. RWM. The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the Control Rod Drop Accident (CRDA) analysis are not violated. Compliance with the analyzed rod position sequence, and operability of the RWM is required in Mode 1, "Power Operation," and Mode 2, "Startup," when thermal power is less than or equal to 10% Rated Thermal Power (RTP). When thermal power is greater than 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA. Because the fuel design limit of 280 cal/gm is not exceeded, this change to lower the Low Power Setpoint (LPSP) does not increase the consequences of an accident previously evaluated.

5. TLHGR. The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The changes in this section revise the description of fuel thermal limits that are monitored to ensure the TLHGR limit is not violated. The TLHGR protects the fuel from 1% plastic strain and fuel centerline melt. Because these criteria have not changed, the consequences of

an accident have not changed.

Therefore, the proposed changes to the CTS do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

1. Administrative Changes. The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, the changes to CTS Section 6.9.A.6.b regarding the COLR, and the changes to the definitions are administrative changes and will not create the possibility of a new or different kind of accident. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.
2. Control Rod Operability and Scram Insertion Times Methodology. The changes to CTS Sections 3/4.3.C, 3/4.3.D, 3/4.3.E, 3/4.3.F, 3/4.3.G, 3/4.3.H, and 3/4.3.I revise the control rod operability and scram time requirements for operation. These changes do not physically alter plant systems, structures or components and therefore do not create the possibility of a new or different kind of accident.
3. Control Rod Scram Times. These changes do not physically alter plant systems, structures or components and therefore do not create the possibility of a new or different kind of accident.
4. RWM. The revisions to CTS Section 3/4.3.L lower the power level at which the analyzed rod position sequence must be followed. This change does not affect plant systems structures, or components. Because there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gam fuel design limit, no new or different type of accident is created.
5. TLHGR. The revisions to CTS Section 3.11.B revise the description of fuel thermal limits that are monitored to ensure the TLHGR is not violated. These changes are analytical in nature and do not affect plant systems, structures, or components. Therefore, the changes do not create the possibility of a new or different kind of accident.

Therefore, the proposed changes to the CTS do not create the possibility of a new or different kind of accident from any previously evaluated.

Does the proposed change involve a significant reduction in a margin of safety?

1. Administrative Changes. The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, the changes to CTS Section 6.9.A.6.b, regarding the COLR, and the changes to the definitions are administrative changes and will not reduce the margin of safety. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. Control Rod Operability and Scram Insertion Times Methodology. The revisions to the CTS control rod operability and scram insertion times ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. CTS methods ensure this by limiting scram times for individual rods, the average scram time, and local scram times (i.e., a four control rod group). ITS methods ensure this by limiting the scram times for individual rods, the number of slow rods, and the number of adjacent slow rods. Each of these methods ensure equivalent protection of the assumed reactivity insertion rate. Therefore, the changes do not involve a reduction in the margin of safety.

   In addition, numerous changes to the control rod operability and scram timing requirements were made to reflect the ITS approach to these requirements. These revisions consist of administrative changes, more restrictive changes, and less restrictive changes. The discussion of each of these categories is provided below.
   Administrative changes. These consist of restructuring, interpretation, and complex rearranging of requirements, and other changes not substantially revising an existing requirement. Therefore, these changes do not affect the margin of safety.
   More restrictive changes. These consist of changes resulting in added restrictions or eliminating flexibility. The more restrictive requirements continue to ensure that process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not reduce the margin of safety.
   Less restrictive changes. The less restrictive changes involve increasing the time to complete actions, increasing the time intervals between required surveillances, and deleting or revising the applicability of certain actions. The time to complete actions and the surveillance frequencies have been extended for several reasons, including experience showing low probability of failures, the benefit of allowing time to perform actions without undue haste, or due to compensating changes in other actions. The deleted or revised actions are not assumed in the safety analyses for any evaluated accidents. Thus, there is no significant reduction in the margin of safety.

3. Control Rod Scram Times. The addition of required scram times for GE analyzed cores based on GE analysis methodology does not involve a reduction in the margin of safety. For GE analyzed cores, cycle-specific analyses using the actual averaged scram times provide MCPR operating limits that will ensure the MCPR safety limit is not violated. Therefore, the fuel remains appropriately protected and no margins of safety are reduced.

4. RWM. The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the CRDA analysis are not violated. Compliance with the analyzed rod position sequence, and operability of the RWM is required in Modes 1 and 2 when thermal power is less than or equal to 10% RTP. When thermal power is greater than 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA. Because the fuel design limit of 280 cal/gm is not exceeded

above 10% RTP, this change to reduce the LPSP does not reduce a margin of safety.

5. TLHGR. The addition of the ratio of Maximum Fraction of Limiting Power Density (MFLPD) to the Fraction of Rated Thermal Power (FRTP) provides thermal limit protection for GE fuel. This provides equivalent protection to ensure that the TLHGR limit is maintained. Therefore, the revisions to CTS Section 3.11.B will not reduce a margin of safety.

Therefore, these proposed changes to the CTS do not involve a significant reduction in the margin of safety.

Proposed Changes to ITS

Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Evaluation of the effect on the probability of an accident previously evaluated.

1. Administrative Changes. The revision to Improved Technical Specification (ITS) Section 5.6.5, "Core Operating Limits Report," and the added definitions are purely administrative changes and do not affect the probability or consequences of an accident previously evaluated.

2. Control Rod Scram Times. The revision to ITS Table 3.1.4-1, "Control Rod Scram Times," adds scram time requirements for GE analyzed cores. This change does not physically alter plant systems, structures or components and therefore does not affect the probability of an accident previously evaluated.

3. Average Power Range Monitor (APRM) Gain and Setpoint. The revisions to ITS Section 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," revise the description of fuel thermal limits that are monitored to ensure the TLHGR is not violated. The changes to this section are analytical in nature and do not affect plant systems, structures, or components and therefore will not affect the probability of an accident previously evaluated.

Evaluation of the effect on the consequences of an accident previously evaluated.

1. Administrative Changes. The revision to ITS Section 5.6.5 and the added definitions are purely administrative changes and do not affect the probability or consequences of an accident previously evaluated.

2. Control rod scram times. The revisions to ITS Section 3.1.4, "Control Rod Scram Insertion Times," are made to ensure the appropriate scram times are reflected in the Technical Specifications (TS) for GE methodology. The scram timing requirements ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. Cycle specific analyses that use the GE methodology scram times will meet all of the same safety limit acceptance criteria. Additionally, for the non-cycle specific UFSAR events, GE has determined that there is negligible impact on the results of events which are not

analyzed on a cycle specific basis. Therefore, there is no change to the consequences of a previously evaluated accident or transient due to the TS changes.

3. APRM Gain and Setpoint. The revisions to ITS Section 3.2.4 will not increase the consequences of an accident previously evaluated. The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The changes in this section revise the description of fuel thermal limits that are monitored to ensure the TLHGR limit is not violated. The TLHGR protects the fuel from 1% plastic strain and fuel centerline melt. Because these criteria have not changed, the consequences of an accident have not changed.

Therefore, the proposed changes to the ITS do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

1. Administrative Changes. The revision to ITS Section 5.6.5 and the added definitions are purely administrative changes and therefore do not create the possibility of a new or different kind of accident.

2. Control Rod Scram Insertion Times. The revisions to ITS Section 3.1.4 do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes to these sections revise the control rod scram time requirements for operation. This change does not physically alter plant systems, structures, or components.

3. APRM Gain and Setpoint. The revisions to ITS Section 3.2.4 will not create the possibility of a new or different kind of accident. The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The changes in this section revise the description of fuel thermal limits that are monitored to ensure the TLHGR is not violated.

Therefore, the proposed changes to the ITS do not create the possibility of a new or different kind of accident from any previously evaluated.

Does the proposed change involve a significant reduction in a margin of safety?

1. Administrative Changes. The revision to ITS Section 5.6.5 and the added definitions are purely administrative changes and do not affect the margin of safety.

2. Control Rod Scram Insertion Times. For GE analyzed cores, cycle-specific analyses using the actual averaged scram times provide MCPR operating limits that will ensure the MCPR safety limit is not violated. Therefore, the fuel remains appropriately protected and no margins of safety are reduced.

3. APRM Gain and Setpoint. The addition of MFLPD/FRTP provides thermal limit protection for GE fuel. This provides equivalent protection to ensure that the TLHGR limit is maintained. Therefore, the revisions to ITS Section 3.2.4 will not reduce a margin of safety.

Therefore, these proposed changes to the ITS do not involve a significant reduction in the margin of safety.

Based on the above evaluation, ComEd has concluded that these changes involve no significant hazards consideration.

Attachment D
Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3
INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) Company has evaluated this proposed change against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and Identification of Licensing and Regulatory Actions Requiring Environmental Assessment." ComEd has determined that this proposed change meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9), "Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b), "Issuance of amendment". This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50,"Domestic licensing of production and utilization facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20,"Standards for protection against radiation," or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment C, this proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed change is limited to revision of fuel types and analytical methods. This change does not allow for an increase in the unit power level, does not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed change does not affect actual unit effluents.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

<u>Attachment E-1</u>
Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3

REVISED CURRENT TECHNICAL SPECIFICATIONS BASES PAGES

B 2-2
B 3/4.3-2
B 3/4.3-3
B 3/4.3-4
B 3/4.3-5
Insert pages (3) for B 3/4.3-3,4,5
B 3/4.6-3
B 3/4.11-1
B 3/4.11-2
B 3/4.11-3
Insert page for B 3/4.11-3

BASES

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 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.

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

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% 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.

Insert attached #1

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~~

BASES

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.

- insert attached #2

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

BASES

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 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.

BASES

3/4.3.G Control Rod Scram Accumulators

insert attached #3

~~The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.~~

~~Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.~~

3/4.3.~~H~~ 4 Control Rod Drive Coupling

Insert attached #4

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. ~~Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the~~ The overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

Insert attached #5

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

Insert #1

3/4.3.C Control Rod OPERABILITY

Control rods are components of the control rod drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including anticipated operational occurrences, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System.

This Specification, along with LCO 3.3.D, "Control Rod Scram Times," LCO 3.3.G, "Control Rod Scram Accumulators," and LCO 3.3.L, "Rod Worth Minimizer," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses.

The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure). the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL), the 1% cladding plastic strain fuel design limit. and the fuel design limit during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel design limits during a CRDA.

The stuck control rod separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods elsewhere in the core adjacent to one another.

An inoperable control rod drive must be disarmed. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM or reactor internals. The control rod isolation method should also ensure cooling water to the CRD is maintained.

Insert #2

3/4.3.D Control Rod Scram Times

The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses (as defined in the COLR) can be met.

The scram function of the CRD System protects the MCPR Safety Limit (SL) and the 1% cladding plastic strain fuel design, which ensure that no fuel damage will occur if these limits are not exceeded. At ≥ 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident and, therefore, also provides protection against violating fuel design limits during reactivity insertion accidents. For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

The scram times specified in Table 3.3.D-1 are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met. To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.3.D-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.3.D-1 is accomplished through measurement and interpolation of the "pickup" or "dropout" times of reed switches associated with each of the required insertion positions. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations (face or diagonal).

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 20% of the control rods in the sample tested are determined to be "slow." With more than 20% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 20% criterion (i.e., 20% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample.

When work that could affect the scram insertion time is performed on a control rod or CRD System, or when fuel movement within the reactor pressure vessel occurs, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure ≥ 800 psig. When only a few control rods have been impacted by fuel movement, the effect on the overall negative reactivity insertion rate is insignificant. Therefore, it is not necessary to perform scram time testing for all control rods when only a few control rods have been impacted by fuel movement in the reactor pressure vessel. During a routine refueling outage, it is expected that all core cells will be impacted, thus all control rods will be tested, consistent with current requirements.

Insert #3

<u>3/4.3.G Control Rod Scram Accumulators</u>

The control rod scram accumulators are part of the Control Rod Drive (CRD) System and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen, which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.3D, "Control Rod Scram Times."

The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. OPERABILITY of each individual control rod scram accumulator, along with LCO 3.3.C, "Control Rod OPERABILITY," and LCO 3.3.D, ensures that the scram reactivity assumed in the DBA and transient analyses (as defined in the COLR) can be met. The existence of an inoperable accumulator may invalidate prior scram time measurements for the associated control rod.

Insert #4

<u>3/4.3.H Control Rod Drive Coupling</u>

The requirements for control rod drive coupling during OPERATIONAL MODES 1 and 2 are presented in Specification 3.3.D, "ControlRod OPERABILITY."

Insert #5

<u>3/4.3.I Control Rod Position Indication System (RPIS)</u>

The requirements for control rod position indication during OPERATIONAL MODES 1 and 2 are presented in Specification 3.3.D, "ControlRod OPERABILITY."

BASES

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

For GE methodology.

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. SPC methodology determines the most limiting pressurization transient each cycle. 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.

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.

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 Appendix K of 10 CFR Part 50.

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 Appendix K of 10 CFR 50. 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.

or MFLPD/FRTP

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.

for SPC fuel and MFLPD/FRTP for GE fuel

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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.

or MFLPD/FRTP

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.

or MFLPD/FRTP

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

or MFLPD/FRTP

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

or MFLPD/FRTP   or MFLPD/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.

For SPC Methodology,

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.

BASES

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.

* See Insert to Section 3/4.11.C

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.

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

Insert to Bases Section 3/4.11.C

For GE methodology, the value of τ, which is the measure of the actual scram speed distribution compared with the assumed distribution, is determined. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (Technical Specification scram times) and Option B (realistic scram times) analyses.

Attachment E-2
Proposed Changes to Technical Specifications
for Dresden Nuclear Power Station, Units 2 and 3

REVISED IMPROVED TECHNICAL SPECIFICATIONS BASES PAGES

REVISED MARKED-UP PAGES

B 2.1.1-3
B 2.1.1-4
B 2.1.1-5
B 2.1.1-6
B 3.1.4-3
B 3.2.2-1
B 3.2.2-2
B 3.2.2-4
B 3.2.2-5
B 3.2.4-1
B 3.2.4-2
B 3.2.4-3
B 3.2.4-4
B 3.2.4-5
B 3.2.4-6
Insert pages (3) for B 3.2.4

REVISED TYPED PAGES

B 2.1.1-3
B 2.1.1-4
B 2.1.1-5
B 2.1.1-6
B 3.1.4-3
B 3.2.2-1
B 3.2.2-2
B 3.2.2-4
B 3.2.2-5
B 3.2.4-1
B 3.2.4-2
B 3.2.4-3
B 3.2.4-4
B 3.2.4-5
B 3.2.4-6

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The use of the Siemens Power Corporation correlation (ANFB) is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 (Refs. 2 and 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

The use of the General Electric (GE) Critical power Correlation (GEXL) is valid for critical power calculations at pressures >785 psig and core flows >10% (Ref. 4)

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr (approximately a mass velocity of 0.25 X 10^6 lb/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale critical power test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative. Although the ANFB correlation is valid at reactor steam dome pressures > 600 psia, applications of the fuel cladding integrity SL at reactor steam dome pressure < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

in the ~~ANFB~~ fuel vendor's critical power correlation. References 2, 3, ~~and~~ 4 describe the methodology used in determining the MCPR SL. and 5

The ~~ANFB~~ fuel vendor's critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the ANFB correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the ANFB correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ~~ANFB~~ fuel vendor's correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. 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 indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This

(continued)

BASES

APPLICABLE SAFETY ANALYSES

<u>2.1.1.3 Reactor Vessel Water Level</u> (continued)

reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

<u>2.2</u>

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. ~~5~~ 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.2.2.1.

2. ANF-524(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors, (as specified in Technical Specification 5.6.5).

3. ANF-1125(P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, (as specified in Technical Specification 5.6.5).

5 ~~4~~. ANF-1125(P)(A), Supplement 1, Appendix E, ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).

6 ~~5~~. 10 CFR 100.

4. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR) (as specified in Technical Specification 5.6.5)

BASES

LCO (continued)

("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement and interpolation of the "pickup" or "dropout" times of reed switches associated with each of the required insertion positions. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods (i.e., one pair of control rods in the core) may occupy adjacent locations (face or diagonal).

Table 3.1.4-1 is modified by two Notes which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY — Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, 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 MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow state ($MCPR_f$) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in UFSAR, Chapter 15 (Ref. 5).

to protect

Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 8) and a multichannel thermal hydraulic code (Ref. 9) to analyze slow flow runout transients on a cycle-specific basis. For core flows less than rated, the established MCPR operating limit is adjusted to provide protection of the MCPR SL in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow (if necessary) control by applying appropriate flow dependent MCPR operating limits. The MCPR operating limit for a given flow state is the greater of the rated conditions MCPR operating limit or the flow dependent MCPR operating limit. For automatic flow control, in addition to protecting the MCPR SL during the flow run-up event, protection is provided by the flow dependent MCPR operating limit to prevent exceeding the rated flow MCPR operating limit during an automatic flow increase to rated core flow. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the appropriate $MCPR_f$ or the rated condition MCPR limit.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

For General Electric (GE) methodology, SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, and Option B (realistic scram times) analyses.

SR 3.2.2.2

For Siemens Power Corporation (SPC) methodology,

Because the transient analyses take credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analyses. SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is then determined based on either the applicable limit associated with the scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. ~~The MCPR limit, including the scram insertion times for rated and off-rated flow conditions, are contained in the COLR.~~ This determination must be performed once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual scram speed distribution expected during the fuel cycle.

of the actual scram speed distribution for SPC methodology and of the parameter τ for GE methodology

REFERENCES

1. NUREG-0562, June 1979.

2. XN-NF-524(P)(A), "Advanced Nuclear Fuels Critical Power Methodology for Boiling Water Reactors," (as specified in Technical Specification 5.6.5).

3. UFSAR, Chapter 4.

(continued)

BASES

REFERENCES (continued)

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 15.

6. EMF-94-217(NP), "Boiling Water Reactor Licensing Methodology Summary," Revision 1, November 1995.

7. NFSR-091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).

8. XN-NF-80-19(P)(A), Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).

9. XN-NF-80-19(P)(A), Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors - THERMEX Thermal Limits Methodology Summary Description, (as specified in Technical Specification 5.6.5).

10. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR) (as specified in Technical Specification 5.6.5)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable final design criteria are discussed in UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10 (Ref. 1). This LCO is provided to require the APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

For SPC fuel, t The condition of excessive power peaking is determined by Fuel Design Limit Ratio for Centerline Melt (FDLRC), which is defined as:

See insert #1 section B 3.2.4

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

where LHGR is the Linear Heat Generation Rate, FRTP is the Fraction of Rated Thermal Power, and TLHGR is the Transient Linear Heat Generation Rate limit. The TLHGR limit is specified in the COLR and protects against fuel centerline melting and the fuel cladding 1% plastic strain during transient conditions throughout the life of the fuel.

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% RTP (APRM Fixed Neutron Flux—High Allowable Value). The APRM Flow Biased Neutron Flux—High Function Allowable Value must be adjusted to ensure that the TLHGR limit is not violated for any power distribution. When FDLRC is greater than 1.0, excessive power peaking exists. To maintain margins similar to those at RTP conditions, the APRM Flow Biased Allowable Value is decreased by 1/FDLRC. As an alternative, this adjustment may also be accomplished by increasing the gain of the APRM by FDLRC. Increasing the APRM gain raises the initial APRM reading closer to the Flow Biased Allowable Value such that a scram would be received at the same point in a transient

replace with insert #2 to section B 3.2.4

e

(continued)

BASES

BACKGROUND (continued)

as if the Allowable Value had been reduced. Thus, increasing the APRM gain by FDLRC provides the same degree of protection as reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by 1/FDLRC. Either of these adjustments has effectively the same result as maintaining FDLRC less than or equal to 1.0, and thus, maintains RTP margins for APLHGR, MCPR, and LHGR.

The normally selected APRM Flow Biased Neutron Flux—High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 3 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Neutron Flux—High Function Allowable Value may be reduced during operation when FDLRC

or the combination of THERMAL POWER and MFLPD

indicates an excessive power peaking distribution.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

UFSAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the fuel design limits could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM ~~Flow Biased Neutron Flux—High Function Allowable Value is adjusted downward by 1/FDLRC, or the APRM gain is adjusted upward by FDLRC.~~ Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Neutron Flux—High Function Allowable Value, dependent on the increased peaking that may be encountered.

gain is adjusted upward by the higher of the core limiting value of FDLRC or the ratio of the core limiting MFLPD to FRTP, or the APRM Flow Biased Neutron Flux-High Function Allowable Value is required to be reduced by the lesser of either the reciprocal of the core limiting FDLRC or by the ratio of FRTP to the core limiting MFLPD.

The APRM gain and setpoint satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by multiplying the APRM Flow Biased Neutron Flux—High Function Allowable Value by the lesser of either 1/FDLRC or the ratio of FRTP and the core limiting value of MFLPD; or

c. Increasing APRM gains to cause the APRM to read greater than or equal to 100% times ~~FRTP times FDLRC~~. This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

the higher of the core limiting value of FDLRC times FRTP or the core limiting MFLPD.

(continued)

BASES

LCO (continued)

Insert #3 to B 3.2.4

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during AOOs beginning at any power level and terminating at 120% of RTP. When FDLRC is greater than 1.0, excessive power peaking exists. To compensate for this condition, the APRM Flow Biased Neutron Flux—High Function Allowable Value is adjusted downward by 1/FDLRC or the APRM gain is adjusted upward by FDLRC. When the reactor is operating with the peaking less than the design value, it is not necessary to modify the APRM Flow Biased Neutron Flux—High Function Allowable Value. Modifying the APRM Flow Biased Allowable Value or adjusting the APRM gain is equivalent to maintaining FDLRC less than or equal to 1.0, as stated in the LCO.

For compliance with LCO 3.2.4.b (APRM Flow Biased Neutron Flux—High Function Allowable Value modification) or LCO 3.2.4.c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain adjusted or Allowable Value modified independently of other APRMs that are having their gain adjusted or Allowable Value modified.

APPLICABILITY

or the ratio of MFLPD to FRTP limit

The FDLRC limit, APRM gain adjustment, or APRM Flow Biased Neutron Flux—High Function Allowable Value ~~is~~ are provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3 sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

or the ratio of MFLPD to FRTP

If the APRM gain or Flow Biased Neutron Flux—High Function Allowable Value is not within limits while FDLRC has exceeded or 1.0, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore the FDLRC to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

and the ratio of MFLPD to FRTP

(continued)

and the ratio of MFLPD to FRTP

BASES (continued)

ACTIONS

A.1 (continued)

The 6 hour Completion Time is normally sufficient to restore either the FDLRC to within limits or to adjust the APRM gain or modify the APRM Flow Biased Neutron Flux — High Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

and the ratio of MFLPD to FRTP

B.1

If FDLRC, the APRM gain or Flow Biased Neutron Flux — High Function Allowable Value cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

and the ratio of MFLPD to FRTP

and the ratio of MFLPD to FRTP

The FDLRC is required to be calculated and compared to 1.0 or APRM gain adjusted or APRM Flow Biased Neutron Flux — High Function Allowable Value modified to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the FDLRC and, assuming ~~FDLRC is greater than~~ 1.0, the appropriate APRM gain or APRM Flow Biased Neutron Flux — High Function Allowable Value, and are not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or Flow Biased Neutron Flux — High Function circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1), MCPR (LCO 3.2.2), and LHGR (LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The

determine

either exceeds

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to APLHGR, MCPR, and LHGR operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 is required when FDLRC or the ratio of MFLPD to FRTP is greater than 1.0, because more rapid changes in power distribution are typically expected.

REFERENCES

1. UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10.

2. UFSAR, Chapter 15.

Insert #1 Dresden ITS B3.2.4

For General Electric (GE) fuel, the condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{MFLPD}{FRTP} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing.

Insert #2 to Dresden B 3.2.4

~~BASES~~

~~BACKGROUND (continued)~~

To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or modification of the APRM Neutron Flux - High Function Allowable Value. Either of these adjustments has effectively the same result as maintaining FDLRC and the ratio of MFLPD to FRTP less than or equal to 1.0 and thus maintains RTP margins for APLHGR, MCPR, and LHGR. Adjustments are based on the lowest APRM Neutron Flux - High Function Allowable Value or highest APRM reading resulting from the two methods (GE or Siemens).

~~The normally selected APRM Flow Biased Neutron Flux - High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Neutron Flux - High Function Allowable Value may be reduced during operation when FDLRC or the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.~~

~~APPLICABLE SAFETY ANALYSES~~

~~The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.~~

(continued)

BASES

LCO (continued)

b. Reducing the APRM Flow Biased Neutron Flux-High Function Allowable Value by multiplying the APRM Flow Biased Neutron Flux-High Function Allowable Value by the lesser of either 1/FDLRC or the ratio of FRTP and the core limiting value of MFLPD; or

c. Increasing APRM gains to cause the APRM to read greater than or equal to 100 (%) times the higher of the core limiting value of FDLRC times FRTP or the core limiting MFLPD. This condition is to account for the reduction in margin to the fuel cladding integrity SL ~~and the fuel cladding 1% plastic~~ strain limit.

Insert 3 to Dresden B 3.2.4

For GE fuel, MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. For Siemens fuel, FDLRC times FRTP is the ratio of the LHGR times 1.2 to TLHGR. As power is reduced, if the design power distribution is maintained, MFLPD and FDLRC are reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD and FDLRC are not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Neutron Flux-High Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Neutron Flux-High Function Allowable Value. Adjusting APRM gain or modifying the APRM Flow Biased Neutron Flux-High Function Allowable Value is equivalent to maintaining FDLRC and the ratio of MFLPD to FRTP less than or equal to 1.0, as stated in the LCO.

For compliance with LCO 3.2.4.b (APRM Flow Biased Neutron Flux-High Function Allowable Value modification) or LCO 3.2.4.c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain adjusted or Allowable Value modified independently of other APRMs that are having their gain adjusted or Allowable Value modified.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The use of the Siemens Power Corporation correlation (ANFB) is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 (Refs. 2 and 3). The use of the General Electric (GE) critical power correlation (GEXL) is valid for critical power calculations at pressures > 785 psig and core flows > 10% (Ref. 4). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr (approximately a mass velocity of 0.25 X 10^6 lb/hr-ft^2), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale critical power test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative. Although the ANFB correlation is valid at reactor steam dome pressures > 600 psia, applications of the fuel cladding integrity SL at reactor steam dome pressure < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the fuel vendor's critical power correlation. References 2, 3, 4, and 5 describe the methodology used in determining the MCPR SL.

The fuel vendor's critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the fuel vendor's correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. 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 indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This

(continued)

BASES

APPLICABLE SAFETY ANALYSES	2.1.1.3 Reactor Vessel Water Level (continued) reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.
SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	2.2 Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.2.2.1.

2. ANF-524(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors, (as specified in Technical Specification 5.6.5).

3. ANF-1125(P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, (as specified in Technical Specification 5.6.5).

4. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR), (as specified in Technical Specification 5.6.5).

5. ANF-1125(P)(A), Supplement 1, Appendix E, ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).

6. 10 CFR 100.

BASES

LCO (continued)

("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement and interpolation of the "pickup" or "dropout" times of reed switches associated with each of the required insertion positions. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods (i.e., one pair of control rods in the core) may occupy adjacent locations (face or diagonal).

Table 3.1.4-1 is modified by two Notes which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods may be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY — Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analyses. Therefore, 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 MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, 9, and 10. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow state ($MCPR_f$) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in UFSAR, Chapter 15 (Ref. 5).

Flow dependent MCPR limits are determined to protect slow flow runout transients on a cycle-specific basis. For core flows less than rated, the established MCPR operating limit is adjusted to provide protection of the MCPR SL 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 (if necessary) by applying appropriate flow dependent MCPR operating limits. The MCPR operating limit for a given flow state is the greater of the rated conditions MCPR operating limit or the flow dependent MCPR operating limit. For automatic flow control, in addition to protecting the MCPR SL during the flow run-up event, protection is provided by the flow dependent MCPR operating limit to prevent exceeding the rated flow MCPR operating limit during an automatic flow increase to rated core flow. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the appropriate $MCPR_f$ or the rated condition MCPR limit.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a low recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analyses take credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analyses. For Siemens Power Corporation (SPC) methodology, SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is then determined based on either the applicable limit associated with the scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The MCPR limit, including the scram insertion times for rated and off-rated flow conditions, are contained in the COLR. For General Electric (GE) methodology, SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4) and Option B (realistic scram times) analyses. This determination of the actual scram speed distribution for SPC methodology and of the parameter τ for GE methodology must be performed once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.2 (continued)

may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual scram speed distribution expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.
2. XN-NF-524(P)(A), "Advanced Nuclear Fuels Critical Power Methodology for Boiling Water Reactors," (as specified in Technical Specification 5.6.5).
3. UFSAR, Chapter 4.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. EMF-94-217(NP), "Boiling Water Reactor Licensing Methodology Summary," Revision 1, November 1995.
7. NFSR-091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).
8. XN-NF-80-19(P)(A), Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronics Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).
9. XN-NF-80-19(P)(A), Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors - THERMEX Thermal Limits Methodology Summary Description, (as specified in Technical Specification 5.6.5).
10. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR), (as specified in Technical Specification 5.6.5).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable final design criteria are discussed in UFSAR, Sections 3.1.2.2.1, 3.1.2.2.4, 3.1.2.3.1, and 3.1.2.3.10 (Ref. 1). This LCO is provided to require the APRM gain or APRM Flow Biased Neutron Flux—High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

For General Electric (GE) fuel, the condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{MFLPD}{FRTP} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing.

For SPC fuel, the condition of excessive power peaking is determined by Fuel Design Limit Ratio for Centerline Melt (FDLRC), which is defined as:

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

where LHGR is the Linear Heat Generation Rate, FRTP is the Fraction of Rated Thermal Power, and TLHGR is the Transient Linear Heat Generation Rate limit. The TLHGR limit is

(continued)

BASES

BACKGROUND (continued)

specified in the COLR and protects against fuel center line melting and the fuel cladding 1% plastic strain during transient conditions throughout the life of the fuel.

To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMS or modification of the APRM Neutron Flux—High Function Allowable Value. Either of these adjustments has effectively the same result at maintaining FDLRC and the ratio of MFLPD to FRTP less than or equal to 1.0 and thus maintains RTP margins for APLHGR, MCPR, and LHGR. Adjustments are based on the lowest APRM Neutron Flux—High Function Allowable Value or highest APRM reading resulting from the two methods (GE or Siemens).

The normally selected APRM Flow Biased Neutron Flux—High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 3 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Neutron Flux—High Function Allowable Value may be reduced during operation when FDLRC or the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

(continued)

BASES *(continued)*

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

UFSAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the fuel design limits could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the higher of the core limiting value of FDLRC or the ratio of the core limiting MFLPD to FRTP, or the APRM Flow Biased Neutron Flux—High Function Allowable Value is required to be reduced by the lesser of either the reciprocal of the core limiting FDLRC or by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Neutron Flux—High Function Allowable Value, dependent on the increased peaking that may be encountered.

The APRM gain and setpoint satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM Flow Biased Neutron Flux—High Function Allowable Value by multiplying the APRM Flow Biased Neutron Flux—High Function Allowable Value by the lesser of either 1/FDLRC or the ratio of FRTP and the core limiting value of MFLPD; or

c. Increasing APRM gains to cause the APRM to read greater than or equal to 100(%) times the higher of the core limiting value of FDLRC times FRTP or the core limiting MFLPD. This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

For GE fuel, MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. For Siemens fuel, FDLRC times FRTP is the ratio of the LHGR times 1.2 to TLHGR. As power is reduced, if the design power distribution is maintained, MFLPD and FDLRC are reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD and FDLRC are not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Neutron Flux—High Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Neutron Flux—High Function Allowable Value. Adjusting APRM gain or modifying the APRM Flow Biased Neutron Flux—High Function Allowable Value is equivalent to maintaining FDLRC and the ratio of MFLPD to FRTP less than or equal to 1.0, as stated in the LCO.

For compliance with LCO 3.2.4.b (APRM Flow Biased Neutron Flux—High Function Allowable Value modification) or LCO 3.2.4.c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain adjusted or Allowable Value modified independently of other APRMs that are having their gain adjusted or Allowable Value modified.

(continued)

BASES (continued)

APPLICABILITY

The FDLRC or the ratio of MFLPD to [illegible] APRM gain adjustment, or APRM Flow Biased [illegible]—High Function Allowable Value are provided [illegible] the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violate[illegible] basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3 [illegible] these limits exists below 25% RTP and, [illegible] rements are only necessary when the react[illegible] 25% RTP.

ACTIONS

A.1

If the APRM gain or Flow Biased Neut[illegible]ux—High Function Allowable Value is not within limits while FDLRC or the ratio of MFLPD to [illegible] the fuel cladding integrit[illegible] plastic strain limit may be reduced. [illegible]tion should be taken to restore [illegible] of MFLPD to FRTP to within its required limit or make acceptable APRM adjustments such that [illegible] within the assumed margin of the [illegible]

The 6 hour Completion Time is norm[illegible]fficient to restore either the FDLRC [illegible] FRTP to within limits or to adjust the APRM ga[illegible] APRM Flow Biased Neutron Flux—[illegible] Value to within limits and is acceptable [illegible] probability of a transient or Design [illegible]taneously with the LCO not met.

B.1

If FDLRC and the ratio of MF[illegible] gain or Flow Biased Neutron Flux—High Function Allowable Value cannot be restored to within its re[illegible] limits within the associated Completion Time [illegible] brought to a MODE or other specified [illegible] LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. [illegible] Completion Time is reasonable, based on operating [illegible] to reduce THERMAL POWER to < 25% RTP in an order[illegible] without challenging plant system[illegible]

(continued)