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ComEd

RS-00-101

September 29, 2000

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Request for Technical Specifications Change, Transition to General Electric Fuel

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

In accordance with 10 CFR 50.90, "Application for amendment license or construction permit," Commonwealth Edison (ComEd) Company is requesting various changes to the Technical Specifications (TS) of Facility Operating License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station (DNPS), Facility Operating License Nos. NPF-11 and NPF-18 for the LaSalle County Station (LCS), and Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station (QCNPS). The proposed changes are 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 proposed changes are requested to improve operational flexibility and allow extended fuel burnup. The proposed changes affect both our Current Technical Specifications (CTS) and our proposed conversion to Improved Technical Specifications (ITS), described in Reference (1), which is currently being reviewed by the NRC. These changes, if approved, will be implemented beginning with the Fall 2001 refueling outages at DNPS and LCS. The proposed changes include the following.

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- 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 TS references to include GE methods in the list of approved analytical methods.

- Revised requirement for adjusting thermal limits when operating in Single Loop Operation to refer to the safety limits and the Core Operating Limits Report (COLR).

- For the DNPS CTS, a revised power level at which the Rod Worth Minimizer (RWM) is required to be operable.

- For the LCS TS, a revised Local Power Range Monitor (LPRM) calibration frequency.

- For the QCNPS TS, addition of an NRC-approved SPC methodology to support operation of the SPC fuel up to exposures anticipated for the QCNPS operating cycle that begins in February 2002, concurrent with the transition to GE 14 fuel.

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

This request is subdivided into three enclosures as follows.

1. Enclosure 1 applies to DNPS and consists of attachments A through F as described below.

2. Enclosure 2 applies to LCS and consists of attachments A through F as described below.

3. Enclosure 3 applies to QCNPS and consists of attachments A through F as described below.

Each of the above enclosures contains the following attachments.

1. Attachment A gives a description and safety analysis of the proposed changes.

2. Attachments B-1 and B-2 include, respectively, the marked-up CTS and ITS pages with the proposed changes indicated.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91(a)(1) which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92(c).

4. Attachment D provides information supporting an Environmental Assessment.

5. Attachments E-1 and E-2 include, respectively, the marked-up CTS and ITS Bases pages with the proposed changes indicated.

6. Attachment F provides a description of the conventions used in marking up portions of the CTS Control Rod TS Sections. These conventions are identical to the conventions used in our proposed conversion of CTS to ITS.

These proposed changes have been reviewed by the Plant Operations Review Committees at each of the three facilities and the Nuclear Safety Review Boards in accordance with the Quality Assurance Program.

ComEd is notifying the State of Illinois of this application request for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning his letter, please contact Mr. Allan R. Haeger at (630) 663-6645.

Respectfully,

R.M. Krich
Vice President, Regulatory Services

Attachments: Affidavit
<u>Enclosure 1: Proposed Changes to Technical Specifications for Dresden Nuclear Power Stations, Units 2 and 3</u>
Attachment A: Description and Safety Analysis for Proposed Changes
Attachment B-1: Marked-Up CTS Pages for Proposed Changes
Attachment B-2: Marked-Up ITS Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment

Attachment E-1: Marked-Up CTS Bases Pages for Proposed Changes
Attachment E-2: Marked-Up ITS Bases Pages for Proposed Changes
Attachment F: Conventions used for Mark-Ups of Current Technical Specifications (CTS)

Enclosure 2: Proposed Changes to Technical Specifications for LaSalle County Station, Units 1 and 2
Attachment A: Description and Safety Analysis for Proposed Changes
Attachment B-1: Marked-Up CTS Pages for Proposed Changes
Attachment B-2: Marked-Up ITS Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E-1: Marked-Up CTS Bases Pages for Proposed Changes
Attachment E-2: Marked-Up ITS Bases Pages for Proposed Changes
Attachment F: Conventions used for Mark-Ups of Current Technical Specifications (CTS)

Enclosure 3: Proposed Changes to Technical Specifications for Quad Cities Nuclear Power Stations, Units 1 and 2
Attachment A: Description and Safety Analysis for Proposed Changes
Attachment B-1: Marked-Up CTS Pages for Proposed Changes
Attachment B-2: Marked-Up ITS Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E-1: Marked-Up CTS Bases Pages for Proposed Changes
Attachment E-2: Marked-Up ITS Bases Pages for Proposed Changes
Attachment F: Conventions used for Mark-Ups of Current Technical Specifications (CTS)

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - LaSalle Nuclear Power Station
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

bcc Dresden Project Manager - NRR
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Regulatory Assurance Manager – LaSalle County Station
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STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	Docket Numbers
Dresden Nuclear Power Station Units 2 and 3	)	50- 237 and 50-249
LaSalle County Station Units 1 and 2	)	50- 373 and 50-374
Quad Cities Nuclear Power Station Units 1 and 2	)	50- 254 and 50-265

SUBJECT: Request for Technical Specifications Change, Transition to General Electric Fuel

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R.M. Krich
Vice President, Regulatory Services

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 29th day of September, 2000.

Notary Public

" OFFICIAL SEAL "
Joseph V. Sipek
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Request for Technical Specifications Change, Transition to General Electric Fuel

ENCLOSURE THREE

Proposed Changes to Technical Specifications
for Quad Cities Nuclear Power Station, Units 1 and 2

Attachment A
Proposed Changes to Technical Specifications
for Quad Cities Nuclear Power Station, Units 1 and 2
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 Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2 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 and allow extended fuel burnup. 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 QCNPS Units 1 and 2, which are scheduled for October 2002 and February 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.
- Addition of an NRC approved SPC methodology to support operation of the SPC fuel up to exposures anticipated for the Quad Cities Unit 2 Cycle 17.

The QCNPS 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 all methodology related to the use of SPC fuel. Appropriate SPC methodology 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 QCNPS 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 sections discuss the current TS requirements for which a change is requested, referencing CTS and ITS as applicable.

Current Requirements for CTS

1. TS 2.1.B, "Thermal Power, High Pressure and High Flow", requires that the MCPR shall not be less than 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. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

2. TS Surveillance Requirement (SR) 4.3.A.2, "Shutdown Margin," requires that the 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 Section 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, "Refueling," 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 3/4.6.A, "Recirculation Loops", Action 1.a. requires that the MCPR Safety Limit be increased by 0.01 for operation with one reactor coolant system recirculation loop. Action 1.b. requires that the MCPR operating limit be increased by 0.01 for operation with one reactor coolant system recirculation

loop.

11. 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 to be taken are either 1) restore FDLRC to less than or equal to 1.0 or 2) adjust the flow biased 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 times FDLRC. A footnote requires the use of 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.

12. TS Section 6.9.A.6.b, "Core Operating Limits Report", 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

13. 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.

14. TS 5.6.5.b, "Core Operating Limits Report" 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 practical 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 Sections ensure that the performance of the control rods meet the assumptions used in the safety analyses. 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 assuming the TS scram speed insertion times and the nominal scram speed insertion times (if applicable). 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. Recirculation Loops (current requirements #10). The transient analyses of Chapter 15, "Accident and Transient Analysis," of the Updated Final Safety Analysis Report (UFSAR) have been 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 as noted by TS Section 2.1.B, i.e., by 0.01.

7. Transient Linear Heat Generation Rate (current requirement #11). 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 the MCPR does not become less than the fuel cladding safety limit or that greater than or equal to 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted when the value of MFLPD/FRTP or FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

8. Core Operating Limits report (current requirement s #12 and ITS requirement #14). The approved analytical methods in the TS reflect NRC approved methodology applicable to Quad Cities.

9. Control rod scram times (ITS requirement #13). 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.

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 QCNPS refueling outages beginning in February 2002 and October 2002, respectively. In addition, certain poposed changes are requested to improve operational flexibility and allow extended fuel burnup.

1. MCPR Safety Limit and Recirculation Loops (current requirements #1 and 10). 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 and Scram Insertion Times (current requirements # 2-9, and ITS requirement #13). The revisions are necessary to adopt the appropriate GE methodology for scram insertion times. CTS reflect an analysis methodology based on limiting the average scram insertion time. ITS limits the number of rods with slow insertion times. Since the requested QCNPS 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 are revised to reflect ITS requirements. This requires changing all of the CTS Sections listed, in order to maintain consistency with the ITS proposed changes.

3. TLHGR (current requirement #11). The revisions are necessary to highlight the use of the ratio of MFLPD/FRTP for monitoring TLHGR for GE fuel. The revision retains the use of both FDLRC and MFLPD/FRTP, but moves the use of MFLPD/FRTP from a footnote to the body of the TS Section. The use of FDLRC for SPC fuel is moved to the footnote.

4. COLR (current requirement #12 and #14). The SPC NRC approved methodology is needed to support the design and operation of future Quad Cities Units 1 and 2 reloads which contain ATRIUM-9B fuel. The NRC approved methodology permits the use of extended burnup limits for ATRIUM 9 designs.

E. DESCRIPTION OF THE PROPOSED CHANGES

Proposed Changes to CTS

1. TS Section 2.1.B, "Thermal Power, High Pressure and 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 Limiting Condition for Operation (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 LCOTS 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," requirements, stated in CTS format. The revised TS Section 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 3.3.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 3.6.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.

11. TS Section 3.11.B, "Transient Linear Heat Generation Rate," retains the use of both FDLRC and MFLPD/FRTP, but moves the use of MFLPD/FRTP from a footnote to the body of the TS Section. The use of FDLRC for SPC fuel is moved to the footnote.

12. TS Section 6.9.A.6.b, "Core Operating Limits Report", is revised to add an NRC approved methodology needed to support the design and operation of future Quad Cities reloads containing ATRIUM-9B fuel. Also added is GE's methodology for determining critical power for SPC fuel.

Proposed Changes to ITS

13. 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

14. TS Section 5.6.5, "Core Operating Limits Report", is revised to add an NRC approved methodology needed to support the design and operation of future Quad Cities reloads containing ATRIUM-9B fuel. The methodology permits the use of extended burnup limits for ATRIUM 9 designs. Also added is GE's methodology for determining critical power for SPC fuel.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGE

1. MCPR Safety Limit and Recirculation Loops (changes #1 and 10). 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 3.3.C Action 1.c and CTS 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 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 LCOTS 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 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 3.3.L, the RWM TS Section, 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 3.3.C Action 1 and CTS 4.3.A.2 has been replaced with the term "stuck" in proposed ACTION 1 of revised TS LCO 3.3.C. The objective 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 TS Section.

A.6 CTS 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 4.3.C.1 only applies to OPERABLE control rods. Thus, this phrase is proposed to be deleted.

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

A.8 CTS 3.3.C Actions 1.a.2), 2.b, and 2.c, footnote (a), CTS 3.3.H, Action 1.b, footnote (b), and CTS 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 3.3.D requirement that maximum control rod scram insertion time be $\leq$ 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 TS Section for excessive scram time by moving the requirement to a 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 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 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 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 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 4.3.D.

A.13 The CTS 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 LCOTS 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 LCOTS 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 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 3.3.I requires all control rod position indicators to be Operable. The objective of the CTS 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 LCOTS TS LCO is that each control rod shall have at least one control rod position indication.

The essence 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 LCOTS TS LCO for control rod position indication 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. Similarly, CTS 3.3.I Action 1 addresses this objective. The proposed SR 4.3.C.1 has combined the CTS 3.3.I objective with the CTS 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 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 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 LCOTS TS LCO 3.3.D, "Control Rod Scram Times." The stuck separation criteria ensures local scram reactivity rate assumptions are met.

M.2 CTS 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 LCOTS TS LCO 3.3.C applies to all inoperable control rods (i.e., when $\leq$ 10% Rated Thermal Power (RTP); see L.1 below) 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 3.3.C Actions require TS LCOTS TS LCO 3.0.C (i.e., within one hour, take action to place the unit in an operational mode in which the requirement does not apply) entry 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 necessary 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 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 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 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 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.

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

LA.1 The details of the recommended procedures for disarming control rod drives (CRDs) specified in CTS 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 3.3.H Action 1.b, and CTS 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 3.3.I Actions 1.a and 1.b, which determine the position of the control rod, which is now proposed to be a Surveillance 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 Surveillance 4.3.C.1. This Surveillance, 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 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 withdrawn inoperable control rods. i.e., See M.2 above.) To preserve scram reactivity, a stuck rod must be separated from other withdrawn 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 3.3.E and 3.3.F i.e., 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 4 hours to correct the situation prior to commencing a required shutdown, while CTS 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 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 two hour time does not represent a significant safety concern as the control rod is already in an acceptable position i.e., 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 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 3.3.C Action 1.c and CTS 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 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 3.3.H Action 1, for uncoupled control rods, and CTS 3.3.I Action 1, for inoperable control rod position indication, provide actions for inoperable control rods. Both CTS 3.3.C Action 2 and CTS 3.3.H Action 1 provide a total of two hours to insert and disarm the control rods, while CTS 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 three 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 Rod Worth Minimizer may be required to be bypassed to allow the rod to be inserted, therefore, the current action times may not be sufficient in 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 out of services, 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 four 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 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 3.3.D Action 2 (i.e., since a control rod can always be declared inoperable), this part of CTS 3.3.D Action 2 has also been deleted.

L.7 Coupling requirements during refueling (i.e., OPERATIONAL MODE 5) specified by CTS 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 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 (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 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 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 Control Rod Drive 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 4.3.I.2 requires that the indicated control rod position change during the movement of the CRD when performing the control rod movement tests (i.e., CTS 4.3.C.1). To perform control rod movement tests required by CTS 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 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., 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 TS Section, this is an added restriction. In addition, the reactor pressure applicability of CTS 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 CRD scram performance distributions allowed by the TS. CTS 3.3.D, 3.3.E, and 3.3.F accomplish the above purpose by placing requirements on maximum individual CRD 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 TS Table 3.3.D-1. SPC and GE methodologies treat slow rods slightly differently; this explains the differences in the Table 3.3.D-1 for SPC and GE analyzed cores. These differences are explained below.

SPC methodology:

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 must only ensure the scram times are satisfied.

The first obvious 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 four control rod groups. This philosophy has been endorsed by the BWR Owners' Group (BWROG) and described in report 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 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 meet the scram time limits found in 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 reports 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 Section. Therefore, revised TS LCO 3.3.D limits the number of slow rods to 12 and ensures no more than 2 slow rods occupy adjacent locations.

The maximum scram time requirement in CTS 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. Note 2 to Table 3.3.D-1 ensures that a control rod is not inadvertently considered "slow" when the scram time exceeds 7 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 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 4.3.D.1.a requires control rod scram time testing for all control rods prior to exceeding 40% RTP following CORE ALTERATIONS. This effectively 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 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 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 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 3.3.G Action 1.c.1) requirement to verify that a CRD 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 CRD 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 3.3.G Action 1.c requires the affected control rod to be declared inoperable. Once declared inoperable, the CTS 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 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 eight hour allowance to essentially restore the inoperable accumulator if the reactor pressure is sufficiently high to support control rod insertion. CTS 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 $\geq$ 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 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 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 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 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 CRD header pressure alone is sufficient to insert control rods if a scram is required (i.e., revised TS LCO 3.3.G Action 1.b.i, 1.c.i, and 1.d).

L.2 CTS 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 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 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.

3. Transient Linear Heat Generation Rate (change #11). This is an administrative change. The revision retains the use of both FDLRC and MFLPD/FRTP, but moves the use of MFLPD/FRTP from a footnote to the body of the TS Section. The use of FDLRC for SPC fuel is moved to the footnote. This relocation does not result in any different use of these thermal limits and is therefore administrative.

4. Control Rod Scram Times (change #13). 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.

5. COLR (change #12 and #14) The basis for adding the NRC approved methodology to the TS is to allow use of the NRC approved extended burnup limits. The RODEX2A Supplements 1 and 2 supports licensing applications up to 62,000 MWd/MTU rod-average burnup and fuel rod/assembly/channel growth models and analytical methods up to 54,000 MWd/MTU assembly-average burnup. The extended burnup limits will support future operation with ATRIUM-9B fuel. The addition of the GE methodology for critical power determination for SPC fuel represents a methodology that is expected to receive NRC approval during the review process for these proposed changes.

G. IMPACT ON PREVIOUS SUBMITTALS

The proposed changes affect our previous request for TS 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 Attachment B. 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 January 1, 2002, in order to support core reload with GE fuel during the QCNPS refueling outage which is currently scheduled to begin early in February 2002.

I. REFERENCES

1. Letter from R.M. Krich (Commonwealth Edison Company) 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 Quad Cities Nuclear Power Station, Units 1 and 2

MARKED-UP CURRENT TECHNICAL SPECIFICATIONS PAGES FOR PROPOSED CHANGES

REVISED PAGES

Revised Marked-Up Pages

VI
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
3/4.3-10
3/4.3-11
3/4.3-12
3/4.3-13
3/4.3-14
3/4.3-15
3/4.6-1
3/4.11-2
6-16a
Insert for 6-16a

Revised Typed Pages

3/4.3-3
3/4.3-4
3/4.3-5
3/4.3-6
3/4.3-7
3/4.3-8
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3/4.3-10
3/4.3-11
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4.3	REACTIVITY CONTROL	
3/4.3.A	SHUTDOWN MARGIN (SDM)	3/4.3-1
3/4.3.B	Reactivity Anomolies	3/4.3-2
3/4.3.C	Control Rod OPERABILITY	3/4.3-3
3/4.3.D	~~Maximum Scram Insertion Times~~ Control Rod Scram Times	3/4.3-6
3/4.3.E	~~Average Scram Insertion Times~~ (deleted)	~~3/4.3-7~~
3/4.3.F	~~Group Scram Insertion Times~~ (deleted)	~~3/4.3-8~~
3/4.3.G	Control Rod Scram Accumulators	3/4.3-9
3/4.3.H	Control Rod Drive Coupling, shutdown	3/4.3-12
3/4.3.I	Control Rod Position Indication System, shutdown	3/4.3-14
3/4.3.J	Control Rod Drive Housing Support	3/4.3-16
3/4.3.K	Scram Discharge Volume (SDV) Vent and Drain Valves	3/4.3-17
3/4.3.L	Rod Worth Minimizer (RWM)	3/4.3-18
3/4.3.M	Rod Block Monitor (RBM)	3/4.3-19
3/4.3.N	Economic Generation Control (EGC) System	3/4.3-20

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 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. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

1.12

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

3.3 - 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 (see revised LCO 3.3.C, L

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. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.

   Stu

   LCO: A.

   stuck

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

REVISED LCO 3.3.C

A.1

A.2

REACTIVITY CONTROL <general reorganization> **CR OPERABILITY 3/4.3.C**

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

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 M.1

   L.1 a. Within one hour: Add proposed Required Action 1.a

   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. M.2

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

   a) Electrically, or M.3

   b) Hydraulically by closing the drive water and exhaust water isolation valves. LA.1

   Add proposed ACTION M.3

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

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

SR 4.3.C.2 SR 4.3.C.3 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

   SR 4.3.C.2 a. At least once per 7 days[b] for each fully withdrawn control rod, and at least once per 31 days[b] for each partially withdrawn control rod, and note(c) SR 4.3.C.3 note(d)

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

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

A.8

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

ctd b Not required to be performed until 7 days (for fully withdrawn) or 31 days (for partially withdrawn) after the control rod is withdrawn and above the low power setpoint of the RWM.

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 hours or be in HOT SHUTDOWN within the next 12 hours.

ACTION 5

72 L.4

(next 70)

ACTION 3

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

add proposed Note to Action 4 L.1

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

L.1

ACTION 4

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

A.4

add proposed Action Note (b)

L.5

Actions 3.a and 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 directional control valves[(a)] either:

1) ~~Electrically, or~~

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

CRD LA.1

M.6

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

A.8

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

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: L.5 4 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.8

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

REVISED LCO 3.3.C

A.1

(general organization) A.10

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,~~ A.11 shall not exceed 7 seconds.

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

L.6

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 above not met, be in at least HOT SHUTDOWN within 12 hours.~~

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.

A.12 add proposed SR 4.3.C.4

see LCO 3.3.D

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

REVISED LCO 3.3.C

L

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

SR 4.3.C.5

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

A.13

APPLICABILITY:

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

retained in specification 3.3.H

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours: L.5

   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and: L.8 A.14 L.9

      1) Observing any indicated response of the nuclear instrumentation, and L.10

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

   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: L.8 CRD LA.1

ACTION 3

      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

SR 4.3.C.5 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.

retained in specification 3.3.

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.

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

Page 5 of 8

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

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

4.3 - SURVEILLANCE REQUIREMENTS

Retained in specification 3.3.H

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.~~

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

L.7

REVISED LCO 3.3.C

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

A.15

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

LA.2

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

LA.2

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

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

SR 4.3.G.1 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.

3. Deleted.

L.11

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.

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

A.8

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

ACTION 5 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.

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 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:~~ NOTE to Surveillance Requirement

or equal to

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

L.1

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

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

add proposed SR

M.1

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

SR 4.3.D.2

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

LA.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

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

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

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

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.

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

Page 2 of 3

REVISED LCO 3.3.D

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

Footnote (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.

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.

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

M.2

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

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

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

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

M.2

Page 3 of 3

REVISED LCO 3.3.G

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

G. Control Rod Scram Accumulators

LCO 3.3G 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, with reactor scram done pressure ≥ 900 psig within 8 hours:

1) Restore the inoperable accumulator to OPERABLE status, or

A.4

M.1 add proposed 1.a.1 Required Action L.1

2) Declare the control rod associated with the inoperable accumulator inoperable.

Action 1.a.ii

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

A.5

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:

M.1 add proposed Action 1.b.i L.1

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

REVISED LCO 3.3.G

A.1

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.

L.2

A.7

Action 1.C

ACTION 1d

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

a) ~~Electrically, or~~

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

A.8

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.

REVISED LCO 3.3.G

A.1

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

1) Electrically, or

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

Action 2

3. 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.

3.6 - 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.

   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)[a] is less than or equal to 1.0. Where FDLRC is equal to:

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

MFLPD[(a)] / FRTP

APPLICABILITY:

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

ACTION: MFLPD/FRTP[(a)]

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

MFLPD/FRTP[(a)]

1. Restore FDLRC to less then 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 FRTP/MFLPD[(a)]
3. Adjust[b] each APRM gain such that the APRM readings are ≥100 times the FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.

Adjust[(b)] 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[(a)].

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[(a)]

The value of FDLRC[a] 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

SPC

FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)

MFLPD/FRTP

a For ~~GE~~ fuel, MFLPD/FRTP is substituted for FDLRC. Adjustments are based on the lowest APRM setpoint or highest APRM reading resulting from the two limits.

b 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.

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

(15) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

(16) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

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

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

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

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

(22) – (insert attached)

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

6.9.B Special Reports

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

Insert for page 6-16a

(22) NEDC – 32981 – P, "GEXL96 Correlationfor 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 $\geq$ 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 $\geq$ 900 psig:

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

4.3 – Surveillance Requirements

(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.

(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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<u>Attachment B-2</u>
Proposed Changes to Technical Specifications
for Quad Cities Nuclear Power Station, Units 1 and 2

MARKED-UP IMPROVED TECHNICAL SPECIFICATIONS PAGES FOR PROPOSED CHANGES

<u>REVISED MARKED-UP PAGES</u>

3.1.4-3
5.6-5

<u>REVISED TYPED PAGES</u>

3.1.4-3
5.6-5
5.6-6

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

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

16. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

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

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

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

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

21. 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.

22. NEDC-32981P, "GEXL96 Correlation for ATRIUM 9B Fuel,"

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.

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.

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.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

16. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

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

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

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

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

21. 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.

22. 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 Changes to Technical Specifications
for Quad Cities Nuclear Power Station, Units 1 and 2

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 Quad Cities Nuclear Power Station (QCNPS), 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) In CTS Section 3.11.B, "Transient Linear Heat Generation Rate," is revised to move the use of the ratio of the Maximum Fraction of Limiting Power Density to the Fraction of Rated Thermal Power (MFLPD/FRTP), which is GE's method for monitoring TLHGR, from a footnote to the body of the TS Section. The use of the Fuel Design Ratio for Centerline (FDLRC) Melt for SPC fuel is moved to the footnote. d) In CTS Section 6.9.A.6.b and ITS Section 5.6.5, the addition of the NRC approved RODEX2A Supplements 1 and 2 is an administrative change because it adds a methodology with has been demonstrated to meet all applicable design criteria.

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.

INFORMATION SUPPORTING A FINDING OF
NO SIGNIFICANT HAZARDS CONSIDERATION

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 to the current requirements for SPC analyzed cores.

Proposed Change to ITS

1. 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.

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 Section 3.11.B, "Transient Linear Heat Generation Rate," regarding the surveillance to monitor Transient linear Heat Generation Rate (TLHGR) using either the ratio of the Maximum Fraction of Limiting Power Density (MFLPD) to the Fraction of Rated Thermal Power (FRTP) or the Fuel Design Limiting Ratio for Centerline (FDLRC) Melt, and the addition of the NRC approved RODEX2A methodology, 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.

INFORMATION SUPPORTING A FINDING OF
NO SIGNIFICANT HAZARDS CONSIDERATION

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 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. The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The administrative changes to Section 3.11.B revise the description of fuel thermal limits that are monitored to ensure the TLHGR limit is not violated. 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. The NRC approved burnup extension for RODEX2A has been demonstrated to meet all applicable design criteria. Therefore, the addition of the NRC approved RODEX2A methodology does not increase the consequences of an accident previously evaluated.
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 TS Sections 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.

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 revisions to CTS Section 3.11.B to revise the description of TLHGR, and the addition of the NRC approved RODEX2A methodology 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.

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, and the changes to Section 3.11.B regarding the surveillance to monitor TLHGR, and the addition of the NRC approved

RODEX2A methodology 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 TS Sections 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.

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

Proposed Change 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 Change. The addition of the NRC approved RODEX2A methodology is an administrative change and will not affect the probability of an accident previously evaluated. This change does not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

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.

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

1. Administrative Change. The NRC approved burnup extension for RODEX2A has been demonstrated to meet all applicable design criteria. Therefore, the addition of the NRC approved RODEX2A methodology does not increase the 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 TS for General Electric (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 events in the Updated Final Safety Analysis Report (UFSAR), 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.

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 Change. The addition of the NRC approved RODEX2A methodology is an administrative change and will not create the possibility of a new or different kind of accident. This change does not affect plant systems, structures, or components. No plant mitigating systems or functions are affected

by this change.

2. <u>Control Rod Scram Insertion Times.</u> 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.

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. <u>Administrative Change</u>. The addition of the NRC approved RODEX2A methodology is an administrative change and will not reduce the margin of safety. This change does not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by this change.

2. <u>Control Rod Scram Insertion Times.</u> 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.

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 Quad Cities Nuclear Power Station, Units 1 and 2
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 SR, 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 revised methodologies for determining core thermal limits and control rod scram times and various related changes that are either administrative or that do not reduce any margins of safety. 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 Quad Cities Nuclear Power Station, Units 1 and 2

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)
B 3/4.6-3
B 3/4.11-2
Insert page
B 3/4.11-3
Insert page

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 x 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 x 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 operating 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 an approved 3-D core simulator code. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

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.

BASES

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

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

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

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

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

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

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

- insert attached #2

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 cycle dependent TSSS MCPR limits and NSS MCPR limits presented in the COLR. Results of the control rod scram tests performed during the current cycle are used to determine the operating limit for MCPR. Following completion of each set of scram 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 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~~ overtravel position feature provides a positive check, as only uncoupled drives may reach this position. The

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 (i.e., one pair of control rods for the reactor) may occupy adjacent locations (face or diagonal). For reactor steam dome pressures < 800 psig, scram times are specified in the ~~DATR~~QATR.

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 3.3.D-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

3/4.3.G Control Rod Scram Accumulators

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

3/4.3.H Control Rod Drive Coupling

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

Insert #5

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

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

BASES

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.

Each safety/relief valve is equipped with diverse position indicators which monitor the tailpipe acoustic vibration and temperature. Either of these provide sufficient indication of safety/relief valve position for normal operation.

3/4.6.G Leakage Detection Systems

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Limits on leakage from the reactor coolant pressure boundary are required so that appropriate action can be taken before the integrity of the reactor coolant pressure boundary is impaired. Leakage detection systems for the reactor coolant system are provided to alert the operators when leakage rates above the normal background levels are detected and also to supply quantitative measurement of leakage rates.

BASES

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).

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 the MCPR does not become less than the fuel cladding safety limit or that ≥1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of MFLPD or FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

SPC Fuel

The Fuel Design Limiting Ratio for Centerline Melt (FDLRC) is incorporated to protect the above criteria at all power levels considering events which cause the reactor power to increase to 120% of rated thermal power.

insert #1

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

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

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

Insert #1

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

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

where LHGR is the LINEAR HEAT GENERATION RATE, AND tlhgr IS THE transient linear heat generation rate. The TLHGR is specified in the CORE OPERATING LIMITS REPORT.

BASES

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. 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 utilizes measured data that is faster than the times required by the Technical Specifications, 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 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.

Insert #2 ←

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 greater 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.

Insert #2

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 Quad Cities Nuclear Power Station, Units 1 and 2

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 3.2.2-2
B 3.2.2-4

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 3.2.2-2
B 3.2.2-4
B 3.2.2-5

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 > 0.1×10^6 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 28×10^3 lb/hr (approximately a mass velocity of 0.25×10^6 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 > 28×10^3 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)

fuel vendor's

in the ~~ANFB~~ critical power correlation. References 2, 3, 4, ~~and~~ 5 describe the methodology used in determining the MCPR SL. and 6

fuel vendor's

The ~~ANFB~~ 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 reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that

(continued)

BASES

APPLICABLE SAFETY ANALYSES

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

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. ~~6~~ 7). 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 3.1.2.1.

2. ANF-524(P)(A), Revision 2, Supplement 1, Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, (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~~ 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 ~~5~~. EMF-1125(P)(A), Supplement 1, Appendix C, ANFB Critical Power Correlation Application for Coresident Fuel, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).

7 ~~6~~. 10 CFR 100.

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

FOR INFORMATION ONLY

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 may occupy adjacent locations (face or diagonal). (i.e., one pair of control rods in the core)

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)

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 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.

(if necessary)

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.

REFERENCES

1. NUREG-0562, June 1979.

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

(continued)

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

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 > 0.1×10^6 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 28×10^3 lb/hr (approximately a mass velocity of 0.25×10^6 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 > 28×10^3 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

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, 5, and 6 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 reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that

(continued)

BASES

APPLICABLE SAFETY ANALYSES	2.1.1.3 Reactor Vessel Water Level (continued) 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. 7). 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 3.1.2.1.

2. ANF-524(P)(A), Revision 2, Supplement 1, Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, (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. EMF-1125(P)(A), Supplement 1, Appendix C, ANFB Critical Power Correlation Application for Coresident Fuel, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).

7. 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)

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 may change during the cycle or after

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.2 (continued)

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. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (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), Revision 1, "Boiling Water Reactor Licensing Methodology Summary," 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).

Attachment F
Proposed Changes to Technical Specifications
for Quad Cities Nuclear Power Station, Units 1 and 2

CONVENTIONS USED FOR MARK-UP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

The annotated CTS control rod TS 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 TS Section is noted on the top right corner of each CTS page, identifying the TS Section 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.

Request for Technical Specifications Change, Transition to General Electric Fuel

ENCLOSURE TWO

Proposed Changes to Technical Specifications for LaSalle County Station, Units 1 and 2

Attachment A
Proposed Changes to Technical Specifications for
LaSalle County Station, Units 1 and 2
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 LaSalle County Station (LCS) Units 1 and 2 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 refueling outages at LCS Units 1 and 2 which are scheduled for November 2001 and November 2002, respectively. The proposed changes include the following.

- 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 LPRM calibration frequency.

- Revised requirement for adjusting thermal limits when operating in Single Loop Operation to refer to the safety limits and the Core Operating Limits Report (COLR).

- In CTS, removed references for SPC analysis methodology.

The LCS units are expected to operate with reactor cores containing both GE and SPC fuel for several operating cycles. Because of this, the proposed ITS changes do not remove the analytical methodologies related to SPC fuel, since ITS is a common document for both units. These methodologies will be retained until both units are operating with cores analyzed with GE methods. In CTS, however, references related to SPC analysis methods were removed, because LCS CTS are specific for Units 1 and 2. If the CTS changes were to be approved, LCS would implement these changes unit-specific, only during the refueling outage in which a GE analyzed core would be loaded. For both CTS and ITS, methodology references related to the mechanical analyses of the SPC fuel will be retained until SPC fuel no longer exists in the core.

As ComEd's fuel vendor, GE will be performing Critical Power Ratio (CPR) calculations to determine safety limits for the LCS 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 TS requirements for which a change is requested, referencing CTS and ITS as applicable.

Current Requirements for CTS

1. TS Section 4.1.1.c, "Shutdown Margin," requires that the shutdown margin (SDM) shall be determined to be adequate within 12 hours after detection of a withdrawn control rod that is immovable, including an increased allowance for the worth of the immovable control rod.

2. TS Section 3/4.1.3.1, "Control Rod Operability," requires that all control rods shall be operable in operational conditions 1, "Power Operation," and 2, "Startup." With one control rod immovable, within one hour, separation criteria are to be verified and the control rod is to be disarmed; in addition, the SDM is to be verified adequate within 12 hours. The immovable control rod is to be made operable within 48 hours or the reactor is to be in hot shutdown within the next 12 hours. With one or more control rods scrammable but otherwise inoperable, separation criteria and insertion capability are to be immediately verified for withdrawn inoperable rods, or else the rods are to be inserted and disarmed. If the provisions for inoperable control rods cannot be met, the reactor is to be in hot shutdown within 12 hours. With more than 8 control rods inoperable, the reactor is to be in hot shutdown within 12 hours. With one or more scram discharge volume (SDV) vent or drain line valves inoperable the associated line must be isolated within seven days for one valve in the line inoperable or within eight hours with two valves in the line inoperable. Otherwise be in hot shutdown within the next 12 hours. The Surveillance Requirements (SRs) require that a) the SDV valves be demonstrated operable by position verification once per 31 days and by cycling once per 92 days; b) withdrawn control rods be verified operable by moving at least one notch once per seven days and once per 24 hours when any rod is immovable; c) all control rods shall be demonstrated operable by performing SRs 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6, and 4.1.3.7; d) the

SDV shall be verified operable at least once per 18 months by verifying that that SDV valves respond to a scram signal.

3. TS Section 3/4.1.3.2, "Control Rod Maximum Scram Insertion Times," requires that the maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05 shall not exceed 7.0 seconds. With one or more control rod scram insertion times exceeding 7 seconds, the control rods are to be declared inoperable. If three or more rods are thus declared inoperable, the maximum scram insertion times of at least 10% of the control rods is to be performed at least once per 60 days of power operation. With the provisions not met, the reactor is to be in hot shutdown within 12 hours. The SRs require that the scram insertion times be demonstrated a) prior to exceeding 40% of rated thermal power following core alterations or a shutdown greater than 120 days, b) for affected control rods following maintenance or modification, and c) for at least 10% of the rods on a rotating basis each 120 days.

4. TS Section 3/4.1.3.3, "Control Rod Average Scram Insertion Times," requires that 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 specified values. With the scram insertion times exceeding limits, the reactor is to be in hot shutdown within 12 hours. The SRs require that control rods be demonstrated operable in accordance with SR 4.1.3.2.

5. TS Section 3/4.1.3.4, "Four Control Rod Group Scram Insertion Times," requires that the average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed specified values. With the control rod scram times greater than the limits, the control rods are to be declared inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group. When operation is continued in this situation, SR 4.1.3.2.C is to be performed at least once per 60 days of power operation. With the provisions not met, the reactor is to be in hot shutdown within 12 hours. The SRs require that control rods be demonstrated operable in accordance with SR 4.1.3.2.

6. TS Section 3/4.1.3.5, "Control Rod Scram Accumulators," requires that all control rod scram accumulators be operable. In operational condition 1 or 2, with one control rod scram accumulator inoperable, within eight hours, the accumulator is to be made operable or the associated control rod is to be declared inoperable. With more than one control rod scram accumulator inoperable, the associated control rods are to be declared inoperable and, a) if the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD (CRD) pump is operating. With no CRD pump operating, immediately place the reactor mode switch in the shutdown position; and b) fully

insert the inoperable control rods. With the provisions for inoperable scram accumulators not met, the reactor is to be in hot shutdown within 12 hours. In operational condition 5, "Refueling," with one withdrawn control rod and associated scram accumulator inoperable the control rod is to be inserted and, within one hour, disarmed. With more than one withdrawn control rod and associated scram accumulator inoperable or with no CRD pump operating, immediately place the reactor mode switch in the shutdown position. The SRs require that each scram accumulator be demonstrated operable at least once per seven days by verifying that indicated pressure is $\geq$ 940 psig.

7. TS Section 3/4.1.3.6, "Control Rod Drive Coupling," requires that all control rods be coupled to their drive mechanisms in operational conditions 1,2, and 5.

8. TS Section 3/4.1.3.7, Control Rod Position Indication," requires that the control rod position indication system shall be operable in operational modes 1,2, and 5.

9. TS 3/4.3.1 Table 4.3.1.1-1, "Reactor Protection System Instrumentation," note (f), requires that the Local Power Range Monitors (LPRMs) be calibrated every 1000 Effective Full Power Hours (EFPH).

10. TS 3/4.4.1.1 "Recirculation Loops," Action a.1.b requires that the Minimum Critical Power Ratio (MCPR) safety limit be increased by 0.01 for operation with one reactor coolant system recirculation loop. Action a.1.c. requires that the MCPR operating limit be increased by 0.01 for operation with one reactor coolant system recirculation loop.

11. TS Section 6.6.A.6, "Core Operating Limits Report (COLR)," requires that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The specific methods are listed.

<u>Requirements for ITS</u>

12. TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," SR 3.3.1.1.8 requires that the LPRMS are calibrated every 1000 EFPH.

13. 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.

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

C. BASES FOR THE CURRENT REQUIREMENTS

1. SDM (current requirement #1). Operation with inoperable control rods is permitted provided that SDM margin requirements are maintained. The requirement to verify SDM within 12 hours is consistent with NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5)," revision 4.

2. Control Rod Operability, Scram Insertion Times, Scram Accumulators, Control Rod Coupling, And Control Rod Position Indication (current requirements # 2 - 8). The specifications in Section 3/4.1.3 ensure that (a) the minimum shutdown margin is maintained, (b) the control rod insertion times are consistent with those used in the accident analysis, and (c) the potential effects of the rod drop accident are limited. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit during the limiting power transient analyzed in Section 15.0 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the fuel cladding safety limit. The occurrence of scram times longer than those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

3. Reactor Protection System Instrumentation (current requirement #9 and ITS requirement #12). The LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) system. This establishes the relative local flux profile for appropriate representative input to the APRM System and core monitoring system. The 1000 EFPH frequency is based on operating experience with LPRM sensitivity changes.

4. Recirculation Loops (current requirement #10). The transient analyses in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) are performed for single recirculation loop operation to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR fuel cladding safety limit is increased as noted by TS Section 2.1.2, "Thermal Power, High Pressure and High Flow."

5. COLR (current requirements #11 and 14). The list of approved methods provides documentation in TS of the approved methods allowed for use in determining core operating limits.

6. Control Rod Operability And Scram Insertion Times (ITS requirement #13). The scram function of the CRD system controls reactivity changes during anticipated operational occurrences 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.

D. NEED FOR REVISION OF THE REQUIREMENTS

The revisions to the requirements listed are requested to support our change of fuel vendors from SPC to GE that will occur during the LCS Units 1 and 2 refuel outages beginning November 2001, and November 2002, respectively. In addition, certain poposed changes are requested to improve operational flexibility.

1. SDM, Control Rod Operability, Scram Insertion Times, Scram Accumulators, Control Rod Coupling, And Control Rod Position Indication (current requirements # 1 - 8 and ITS requirement #13). The revisions are necessary to adopt the appropriate GE methodology for scram insertion times. CTS reflect an analysis methodology based on limiting the average scram insertion time. ITS limits the number of rods with slow insertion times. Since the requested LCS 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 are proposed to reflect ITS requirements. This requires changing all of the CTS sections listed, in order to maintain consistency with the ITS proposed changes.

   The addition of scram times for GE analyzed cores in CTS and ITS is required to ensure that the required scram times reflect the appropriate analysis methodology.

2. Reactor Protection System Instrumentation (current requirement #9 and ITS requirement #12). The revision to this section is requested to extend an unnecessarily restrictive SR interval. The MCPR Safety limit analyses for the ComEd Boiling Water Reactors (BWRs) have been performed assuming a 2500 EFPH LPRM calibration interval to support the 2000 EFPH LPRM calibration intervals for previous cycles and will continue to support 2000 EFPH.

3. COLR (current requirements #11 and 14). The revision to this section is necessary to remove references to SPC methodology that will no longer be applicable. The LCS units are expected to operate with reactor cores containing

both GE and SPC fuel for several operating cycles. In CTS, however, references related to SPC analysis methods were removed, because LCS CTS are specific for Units 1 and 2. If the CTS changes were to be approved, LCS would implement these changes prior to startup from the refueling outage in which the first GE analyzed core reload batch would be loaded. Thus, SPC analysis methods would no longer be applicable at the time of implementation.

4. Recirculation Loops (current requirement #10). The revision is necessary because the value of the difference between the single loop operation MCPR safety limit and the two 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. The value of the MCPR safety limit for single loop operation will be specified explicitly in the TS, rather than as an increment to the two loop operation limit, to properly reflect the fact that these limits are calculated separately.

E. DESCRIPTION OF THE PROPOSED CHANGES

Proposed Changes to CTS

1. TS Section 4.1.1.C, "Shutdown Margin," is revised to require that the SDM be verified acceptable within 72 hours of discovering a control rod that is stuck.

2. TS Section 3/4.1.3.1, "Control Rod Operability," is revised to reflect ITS Section 3.1.3, "Control Rod Operability," requirements, stated in CTS format. The revised TS Section has incorporated portions of CTS Sections 3/4.1.3.2, 3/4.1.3.3, 3/4.1.3.4, 3/4.1.3.5, 3/4.1.3.6, and 3/4.1.3.7 in order to contain all of the requirements for determining the operability of control rods. The portions of the TS Section concerning SDV vent and drain lines and associated valves have been located in a new proposed TS Section 3/4.1.3.3, "Scram Discharge Volume." The specific changes are shown in the marked-up TS pages in Attachment B-1.

3. a. TS Section 3/4.1.3.2, "Control Rod Maximum Scram Insertion 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 3.a above, the required scram times are modified to include the required scram times based on GE methodology. These scram times are included in proposed TS Table 3.1.3.2-1.

4. TS Section 3/4.1.3.3, "Control Rod Average Scram Insertion Times," is deleted. The average scram time requirement is replaced with the requirement to limit the number of slow rods. The SR s are incorporated in revised TS Section 3/4.1.3.2. The specific changes are shown in the marked-up TS pages in Attachment B-1.

5. TS Section 3/4.1.3.4, "Four Control Rod 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 SR s are incorporated in revised TS Section 3/4.1.3.2. The specific changes are shown in the marked-up TS pages in Attachment B-1.

6. TS Section 3/4.1.3.5, "Control Rod Scram Accumulators," is revised to reflect ITS Section 3.1.5, "Control Rod Scram Accumulators," requirements, stated in CTS format. The revised TS Section 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.

7. TS Section 3/4.1.3.6, "Control Rod Drive Coupling," is revised to reflect ITS Section 3.1.3 requirements in operational conditions 1 and 2. This relocates the requirements for control rod coupling for operational conditions 1 and 2 to revised TS Section 3.1.3.1. The TS Section remains unchanged for operational condition 5. The specific changes are shown in the marked-up TS pages in Attachment B-1.

8. TS Section 3/4.1.3.7, Control Rod Position Indication," is revised to reflect ITS Section 3.1.3 requirements in operational conditions 1 and 2. This relocates the requirements for control rod position indication for operational conditions 1 and 2 to revised TS Section 3.1.3.1. The TS Section remains unchanged for operational condition 5. The specific changes are shown in the marked-up TS pages in Attachment B-1.

9. TS 3/4.3.1, "Reactor Protection System Instrumentation," Table 4.3.1.1-1, note (f), is revised to change the calibration interval requirement for the LPRMs from 1000 EFPH to 2000 EFPH.

10. TS 3/4.4.1.1 "Recirculation Loops", Action a.1.b is modified to refer to Section 2.1.2 of CTS for the MCPR safety limit for operation in Single Loop; this removes the requirement to add 0.01. Action a.1.c is modified to refer to the Core Operating Limits Report (COLR) for the MCPR operating limit for operation in single loop operation.

11. TS Section 6.6.A.6, "Core Operating Limits Report," is modified to remove references #1,2,3,4,5,7,8,9,11,13,14,16,23,24,and 25 and to add 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 ITS

12. TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," SR 3.3.1.1.8 was revised to change the calibration interval requirement for the LPRMs from 1000 EFPH to 2000 EFPH.

13. 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.

Position Inserted From Fully Withdrawn	Scram Times for GE analyzed Cores(seconds)
45	0.52
39	0.86
25	1.91
5	3.44

14. TS Section 5.6.5, "Core Operating Limits Report," is revised to add GE's methodology for determining critical power for SPC fuel. Section I of this Attachment provides references related to NRC approval of this method.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

1. SDM (change #1). 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 Section 3/4.1.3.1 Action a.3 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 4.1.1.c has been extended from 12 hours to 72 hours, and provides a reasonable time to perform the analysis or test. This is consistent with the BWR ISTS, Reference I.6.

2. Control rod operability and scram insertion times (changes #2 – 8). 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 3/4.1.3.1 – 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 Section 3/4.1.3.1) 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 (i.e., scram insertion times, coupling integrity, and ability to determine position). 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 The portions of CTS 3/4.1.3.1 concerning SDV have been located in proposed new TS Section 3.1.3.3, "Scram Discharge Volume. This is to maintain consistency with the submitted ITS. None of the requirements were changed. Therefore, this represents a purely administrative change.

A.4 A Note is added to CTS 3.1.3.1, Actions a and b (i.e., Revised Actions Notes a and C.1) 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 3.1.4.1, the RWM TS Section, are taken. This is a human factors consideration to assure clarity of the requirement and allowance.

A.5 The existing phrase of "being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable" in CTS 3.1.3.1 Action a has been replaced with the term "stuck" in proposed Condition A of revised TS Section 3/4.1.3.1. The objective 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. A similar phrase in CTS 4.1.1.c has also been changed to "stuck."

A.6 CTS 3.1.3.1 Actions a.1.b), b.1.b), and b.2.a), footnote *, CTS 3.1.3.6 Action a.1.b) footnote **, and CTS 3.1.3.7 Action a.3.b) footnote **, which permit the directional control valves to be rearmed intermittently, have been deleted since TS Section 3.0.6 provides this allowance. Therefore, deletion of this allowance is administrative.

A.7 Not used.

A.8 CTS 4.1.3.1.2 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. Currently, inoperable control rods are already not required to meet this Surveillance in accordance with CTS 4.0.3, and therefore, CTS 4.1.3.1.2 only applies to OPERABLE control rods. Therefore, this phrase is proposed to be deleted since it is not needed.

A.9 These listed Surveillances in CTS 4.1.3.1.3 are required by other TS Sections. Repeating a requirement to perform these Surveillances is not necessary. Elimination of this "cross-reference" is therefore administrative.

A.10 Not used.

A.11 The CTS 3.1.3.2 requirement that maximum control rod scram insertion time be $\leq$ 7 seconds is presented in proposed SR 4.1.3.1.4, making it a requirement for control rods to be considered OPERABLE. Eliminating the separate TS Section for excessive scram time by moving the requirement to a SR does not eliminate any of the requirements, or impose new or different treatment of the requirements other than those proposed in Section L.8 below. Therefore, this proposed change is administrative.

A.12 The definition of time zero in CTS 3.1.3.2 (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 3.1.3.3 and 3.1.3.4, which is maintained in proposed footnote (a) to proposed Table 3.1.3.2-1. No change has been made to the defined time zero, therefore, this deletion is administrative.

A.13 CTS 4.1.3.2, which provides the scram time testing requirements, is addressed in revised TS Section 3/4.1.3.2. Therefore, proposed SR 4.1.3.1.4 has been added to require the SRs in revised TS Section 3/4.1.3.2 to be performed. Changes to the testing requirements in SRs 4.1.3.2.1, 4.1.3.2.2, 4.1.3.2.3, 4.1.3.2.4, and 4.1.3.2.5 4 are addressed in the safety analysis for revised TS Section 3/4.1.3.2.

A.14 The CTS 3.1.3.6 requirement that control rods be coupled to their drive mechanism is presented in proposed SR 4.1.3.1.5. As a Surveillance in the Control Rod OPERABILITY Limiting Condition for Operation (LCO), it is a requirement for control rods to be considered OPERABLE. The actions for uncoupled control rods continue to be required as discussed in L.4, L.9, L.10, L.11, and L.12 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.15 CTS 3.1.3.6 Action a.1.a) contains the method of restoring coupling integrity to an uncoupled control rod (i.e., insert the CRD 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.16 CTS 4.1.3.6.c addresses the requirement to perform coupling checks after performing activities which could have affected coupling integrity. This Surveillance must be completed prior to allowing the control rod to be considered OPERABLE. The consideration of OPERABILITY is more clearly presented in the proposed editorial rewrite of CTS 4.1.3.6.c into the Frequency for proposed SR 4.1.3.1.5. Therefore, CTS 4.1.3.6.a is redundant. "CORE ALTERATIONS that could have affected the CRD coupling integrity" is a subset of the CTS 4.1.3.6.c requirement, "maintenance ...which could have affected the CRD coupling integrity." Performance of the integrity verification prior to control rod OPERABILITY, which is the understanding of CTS 4.1.3.6.c as presented in the proposed SR 4.1.3.1.5, bounds "prior to reactor criticality." Therefore, elimination of CTS 4.1.3.6.a is administrative and represents no change in requirements.

A.17 The objective of the CTS 3.1.3.7 requirement is understood to be related to each control rod. The Applicability footnote "*", each specific action within Action a, Action b, and each SR all 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 proposed SR 4.1.3.1.1 of revised TS Section 3/4.1.3.1, "Control Rod OPERABILITY." 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 Surveillance 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 3.1.3.7 Actions a.1 and a.2 address this objective . The proposed SR 4.1.3.1.1 has combined the CTS 3.1.3.7 objective with the CTS 3.1.3.7 Actions a.1 and a.2 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 3.1.3.7 Action a.1 or a.2, or meeting proposed SR 4.1.3.1.1.

A.18 Not used.

A.19 The requirements of CTS 3.1.3.7 Action a.3.(a)2) are now covered by the Note to revised TS Section 3/4.1.3.1 Action C.1, which states, in part, that RWM may be bypassed as allowed by TS LCO 3.1.4.1. Therefore, an explicit Action in this TS Section to verify the position and bypassing of RWM is not needed.

Attachment A
Proposed Changes to Technical Specifications for
LaSalle County Station, Units 1 and 2
DESCRIPTION AND SAFETY ANALYSIS
FOR PROPOSED CHANGES

Revised TS Section 3/4.1.3.1 – TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.1.3.1 Actions a.1.a) and b.1.a)1) require the separation criteria to be met only for withdrawn control rods. Action d of the revised TS Section 3/4.1.3.1 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.2 The CTS 3.1.3.1 Actions require TS LCO 3.0.3 entry if more than one control rod is stuck. The proposed revised TS Section 3/4.1.3.1 Action b maintains the equivalent shutdown action as TS LCO 3.0.3, but also contains an additional requirement to disarm the stuck control rod (i.e., revised TS Section 3/4.1.3.1 Action a.2). The Bases for this Action requires the disarming to be performed hydraulically. This additional requirement provides a necessary level of protection to the CRD should a scram signal occur. If mechanically bound, the stuck control rod could cause further damage if not hydraulically disarmed. Disarming normally would preclude control rod insertion on a scram signal; however, since this control rod is stuck, this effect of disarming is moot. In addition, CTS 3.1.3.1 Action a.1.b) 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.3 The proposed changes to CTS 3.1.3.1 Action b.1.a)2) including footnote **, 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 3.1.3.1 Action b.1.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 low power setpoint. The first condition is addressed in the safety analysis for proposed TS Section 3/4.1.3.5. 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 Section 3/4.1.3.1 Action d) assure that insertion of these control rods remain appropriately controlled.

M.4 Not used.

M.5 Proposed SR 4.1.3.1.2 and SR 4.1.3.1.3 require control rods to be inserted in lieu of the CTS 4.1.3.1.2 requirement for "moving." 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 CTS 3.1.3.7 Action a.3.(a)1) requires a control rod to be declared inoperable when THERMAL POWER is within the low power setpoint and one or more control rod position indicators is inoperable and control rod position is unknown. CTS 3.1.3.1 Action b for inoperable rods provides the option to verify the insertion capability, and then allows the control rod to remain withdrawn. The proposed changes to the Actions for non-stuck inoperable control rods (i.e., revised TS Section 3/4.1.3.1 Action c) eliminates the check of insertion capability; replacing it with a requirement to fully insert and disarm all inoperable control rods. The effect on the CTS 3.1.3.7 Action a.3.(a)1) for control rods with position unknown, when below the low power setpoint, is to eliminate the option to leave the control rod withdrawn and continue to operate. The control rod will be required to be inserted and disarmed, regardless of the power level, which is currently the requirement if power is greater than the low power setpoint. 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.

<u>Revised TS Section 3/4.1.3.1 – TECHNICAL CHANGES - LESS RESTRICTIVE</u>

LA.1 The details of the recommended procedures for disarming CRDs specified in CTS 3.1.3.1 Actions a.1.b), b.1.b), and b.2.a), CTS 3.1.3.6 Action a.1.b), and CTS 3.1.3.7 Action a.3.(b) 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 Section 3/4.1.3.1 Actions a.2 and c.2, 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 3.1.3.7 Actions a.1 and a.2, which determine the position of the control rod that are now proposed to be a Surveillance for control rod OPERABILITY (i.e., refer to A.17 above) can be met a number of ways. Three ways are presented: by moving the control rod, by single notch movement, to a position with an OPERABLE position indicator, then returning it, by single notch movement, to its original position and periodically verifying no control rod drift alarm, 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 Surveillance (i.e., SR 4.1.3.1.1). SR 4.1.3.1.1, 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 3.1.3.1 Actions a.1 and b.1.a)1) are presented in revised TS Section 3/4.1.3.1 Action d to provide the requirements and actions for the local distribution of inoperable control rods. Three distinct changes are addressed:

1) Revised TS Section 3/4.1.3.1 Action d is modified by a Note excluding its applicability above 10% rated thermal power (RTP). The existing separation requirements for a stuck control rod, in part, account for allowing withdrawn inoperable control rods. (i.e., refer to M.3 above.) To preserve scram reactivity, a stuck rod must be separated from other withdrawn inoperable control rods which may also not scram. In the TS, all inoperable control rods which will not scram or cannot be verified to scram (e.g., loss of position indication) 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 Control Rod Drop Accident (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 Section 3/4.1.3.1 Action d 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 Sections 3.1.3.2, CTS 3.1.3.3, and 3.1.3.4 (i.e., as modified to reflect

revised TS Section 3/4.1.3.2). 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 Section 3/4.1.3.1 Action d allow 4 hours to correct the situation prior to commencing a required shutdown, while CTS 3.1.3.1 Action a.1 allows 1 hour and Action b.1.a) requires immediate action. 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 (some operational considerations may be necessary to adjust control rod patterns and/or power levels), 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 3.1.3.1 Action a.1.b) 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 Section 3/4.1.3.1 Action a.2, consistent with the BWR ISTS, Reference I.6, in recognition of the potential for excessive haste required to complete this task. The proposed two 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 3.1.3.1 Action a.3, which requires restoration of a stuck control rod within 48 hours, is being deleted. In addition, a new Action is being added (i.e., revised TS Section 3/4.1.3.1 Action a.1), since the proposed TS Section now allows continued operation with a stuck control rod. With a single withdrawn control rod stuck, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. During a transient, a single stuck control rod in addition to an assumed single failure will have no significant impact on the established operating limits. SDM must still be met, accounting for the loss of negative reactivity due to the stuck control rod. The stuck rod must also meet certain separation criteria (i.e., revised TS Section 3/4.1.3.1 Action a.1). Prompt action is required to confirm no additional stuck control rods exist (i.e., revised TS Section 3/4.1.3.1 Action a.3). Therefore, continued operation is proposed to be allowed, as are MODE changes in accordance with TS Section 4.0.4.

L.4 All inoperable non-stuck control rods are required to be fully inserted and disarmed Refer to M.3 above. The time allowed to complete the insertion is proposed to be extended to 3 hours for all cases. In the existing Actions for an uncoupled control rod (i.e., CTS 3.1.3.6 Action a.1.b)), time is provided to recouple and, if unsuccessful, insert the control rod before entering CTS 3.1.3.1 Action b.1. Two hours are currently allowed to perform these Actions. CTS 3.1.3.1 Action b.1.b) provides no additional time to disarm the control rod (i.e., total of two hours to insert and an immediate time to disarm). Uncoupled control rod actions are proposed to be addressed by revised TS Section 3/4.1.3.1 Action c, as are other non-stuck inoperable control rods. This existing three hour allowance, before requiring an inoperable (i.e., uncoupled) control rod to be inserted, is the time found in the revised TS Section 3/4.1.3.1 Action c.1 for control rod insertion. For consistency of presentation, this three hour limitation is also proposed for all other instances of inoperable control rods. These other instances (e.g., loss of position indication, excessive scram speed, certain combinations of conditions with a low pressure on a control rod scram accumulator) also warrant a minimal time to attempt restoration prior to inserting and disarming. It is for these other instances that the extended time to insert are proposed. Since these instances do not represent loss of SDM, and are limited to a total of no more than 8 inoperable control rods, the extended time does not represent a significant safety concern.

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. 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. The disarming is proposed to be extended to four hours in revised TS Section 3/4.1.3.1 Action c.2 -- one hour beyond that allowed to insert. This is consistent with the BWR ISTS, Reference I.6, 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 is already in its required position in accordance with other actions, and the action to disarm is solely a mechanism for precluding the potential for future misoperation.

L.5 CTS 4.1.3.1.2.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 4.1.3.1.3). This is acceptable given the following:

1) At full power, a large percentage of control rods (i.e., typically 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 partially withdrawn, must be tested within 24 hours (i.e., revised TS Section 3/4.1.3.1 Action a.3).

L.6 CTS 4.1.3.1.2.b requires a daily notch test in the event power operation is continuing with an immovable control rod and the plant is operating at greater than the low power setpoint of the rod worth minimizer. The TS requires the control rod notch test only once within 24 hours after the plant is operating at greater than the low power setpoint of the rod worth minimizer (i.e., revised TS Section 3/4.1.3.1 Action A.3). The purpose of the control rod notch test on each withdrawn OPERABLE control rod is to ensure that a generic problem does not exist and that control rod insertion capability remains. The single performance of the control rod notch test satisfies the same function as the daily notch test of the CTS without requiring the additional testing.

L.7 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 Section 3/4.1.3.1 Action a.3 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 4.1.1.c has been extended from 12 hours to 72 hours, and provides a reasonable time to perform the analysis or test.

L.8 The CTS 3.1.3.2 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 CRD 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 4.1.3.1.2 and SR 4.1.3.1.3) and accumulator OPERABILITY (i.e., proposed SR 4.1.3.5.1), 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 (i.e., "Otherwise, be in at least HOT SHUTDOWN within 12 hours") could have only applied to CTS 3.1.3.2 Action 2, since a control rod can always be declared inoperable, this part of the CTS 3.1.3.2 Action has also been deleted.

L.9 Not used.

L.10 If an uncoupled control rod is not allowed by the RWM to be inserted to accomplish recoupling, CTS 3.1.3.6 Action a.1.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 Section 3/4.1.3.1 Actions c.1 and c.2). Also, because of the limited time allowed to recouple, the number of attempts (i.e., currently limited to one by CTS 3.1.3.6 Action a.1.b) does not need to be restricted. The number of attempts to recouple a control rod may be restricted by plant procedures, which consider the potential for equipment damage during successive recoupling attempts.

L.11 Proposed SR 4.1.3.1.5 verifies a control rod does not go to the withdrawn overtravel position. An uncoupled control rod would fail to meet SR 4.1.3.1.5. After restoration of a component that caused a failure to meet a required 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.12 below. As a result, CTS 3.1.3.6 Action a.1.a.2) requirements are proposed to be deleted since they are not necessary for ensuring recoupling of the control rod.

L.12 The CTS 3.1.3.6 Action a.1.a)1) and CTS 4.1.3.6 requirements 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 4.1.3.1.5 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 control rod drive (CRD) System that could affect coupling. As a result, SR 4.1.3.1.5 provides adequate assurance that the control rods are coupled.

L.13 CTS 3.1.3.7 Action a.1 provides methods for determining the position of a control rod whose position indicator is inoperable. These methods require determining position of the control rod by moving the control rod, by single notch movement, to a position with an OPERABLE position indicator, then returning the control rod, by single notch movement, to its original position, and verifying no rod drift alarm is annunciated every 12 hours. The 12 hour requirement to verify no rod drift alarm is being deleted. The TS will require the rod position to be determined every 24 hours (i.e., proposed SR 4.1.3.1.1). Thus, if the method of CTS 3.1.3.7 Actions a.1.(a) and (b) is being used to determine the position of a control rod, it will have to be performed every 24 hours. Currently, it has to be performed only once, then the rod drift alarm is used to verify the rod has not moved. In addition, the alarm provides annunciation in the control room and will alarm if any control rod moves; the alarm is not associated with any one single control rod. The probability of a control rod with an inoperable indicator moving is no different than the probability of a control rod with Operable indicators moving. There are numerous controls/indicators available, that would make a mispositioned control rod readily apparent to the operator, such that appropriate actions could be taken, even without the verification of the alarm. This deletion is also consistent with Reference I.6, which, while not listing this specific method in the Bases of SR 4.1.3.1.1, does specify that other appropriate methods to determine rod position can be used.

L.14 Current SRs for the control rod position indication system (i.e., CTS 4.1.3.7.b, 4.1.3.7.c, and 4.1.3.7.d) require that the control rod position indication system be determined OPERABLE during the performance of the control rod movement tests (i.e., CTS 4.1.3.1.2) and the control rod withdrawal for coupling verifications (i.e., CTS 4.1.3.6.b) prior to startup, and each time a control rod is fully inserted. To perform control rod movement tests required by CTS 4.1.3.1.a (i.e., proposed SR 4.1.3.1.2 and SR 4.1.3.1.3) and control rod coupling verifications required by CTS 4.1.3.6.b (i.e., proposed SR 4.1.3.1.5), position indication must be available. If position indication is not available, these tests cannot be satisfied and appropriate actions will be taken for inoperable control rods in accordance with the Actions of revised TS Section 3/4.1.3.1. As a result, the requirements for the control rod position indication system are adequately addressed by the requirements of revised TS Section 3/4.1.3.1 and associated SR 4.1.3.1.2, SR 4.1.3.1.3, and SR 4.1.3.1.5 and are proposed to be deleted.

Revised TS Section 3/4.1.3.2 – 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.

Revised TS Section 3/4.1.3.2 – TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The pressure at which the control rods must be tested in CTS 4.1.3.2 (i.e., proposed SRs 4.1.3.2.1, 4.1.3.2.2, and 4.1.3.2.4) has been changed from ≥ 950 to ≥ 800 psig. This pressure corresponds to the limiting pressure for CRD scram testing for the LaSalle 1 and 2 CRD System. "Limiting" refers to the maximum scram times experienced at or below this pressure because of the competing effects of the reactor vessel pressure and the accumulator pressure scram forces. The scram time requirements are related to transients analyzed at rated reactor pressure (i.e., assumed to be ≥ 950 psig); however, if the scram times are demonstrated at pressures above 800 psig, the measured times are conservative with respect to the conditions assumed in the design basis transient and accident analyses.

M.2 In the CTS 4.1.3.2.b SR "for specifically affected" control rods, deleting the flexibility provided in CTS 4.1.3.2 to delay post-maintenance testing until reactor pressure is ≥ 950 psig is proposed, to ensure adequate testing is performed prior to declaring the control rod operable, which could include prior to entering MODE 2. In support of the proposed restriction, an additional SR, SR 4.1.3.2.3, 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. 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 pressure. Since this test, and therefore any limits, are not applied in the existing TS Section, this is an added restriction. Furthermore, the CTS 4.1.3.2.b scram time test requirement (i.e., proposed SR 4.1.3.2.4) performed at normal operating reactor pressure, is additionally required to be performed prior to exceeding 40% RTP. This places a finite time on the test if maintenance was performed on the control rod in MODE 1 or 2, and ensures the control rod scram times are within the analyzed limits prior to full power operation. It is noted that if the control rod remains inoperable, which requires it to be inserted and disarmed, until normal operating pressures, a single scram time test will satisfy both SRs (i.e., SR 3.1.4.3 and SR 3.1.4.4).

M.3 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 CRD scram performance distributions allowed by the Technical Specifications. CTS 3.1.3.2, 3.1.3.3, and 3.1.3.4 accomplish the above purpose by placing requirements on maximum individual CRD scram times (i.e., seven second requirement), average scram times, and local scram times (i.e., a four control rod group).

SPC Methodology

Because of the methodology used in the design basis transient analysis, 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 must only ensure the scram times are satisfied.

The first obvious 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, 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 Section 3/4.1.3.2 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 four control rod groups. This philosophy has been endorsed by the BWR Owners' Group (BWROG) and described in report 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 proposed Table 3.1.3.2-1 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 Proposed Table 3.1.3.2-1, the analytical scram reactivity assumptions are satisfied. If any control rods do not meet the scram time limits, revised TS Section 3/4.1.3.2 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 set cycle-specific operating limits.

In both SPC and GE methodology, 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 Section. Therefore, revised TS Section 3/4.1.3.2 limits the number of slow rods to 12 and ensures no more than 2 slow rods occupy adjacent locations.

The maximum scram time requirement in CTS 3.1.3.2 has been retained in revised TS Section 3/4.1.3.1 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 3.1.3.2-1 ensures that a control rod is not inadvertently considered "slow" when the scram time exceeds seven seconds.

Revised TS Section 3/4.1.3.2 – TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 Proposed SR 4.1.3.2.2 will test a representative sample of control rods each 120 days of power operation instead of the CTS 4.1.3.2.c 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 Section 3/4.1.3.2 and SR 4.1.3.2.2 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 4.1.3.2.a requires control rod scram time testing for all control rods prior to exceeding 40% RTP following CORE ALTERATIONS, except for normal control rod movement (i.e., footnote *). This effectively 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 4.1.3.2.4 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 except for normal control rod movement 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 4.1.3.2.4. The Surveillances of revised TS Section 3/4.1.3.2 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 for the control rods.

Revised TS Section 3/4.1.3.5 – ADMINISTRATIVE

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 method for disarming control rods is proposed to be deleted, since either method of disarming (i.e., electrically or hydraulically) is allowed, except in cases of a stuck rod (see discussion M.2 in Revised TS Section 3.1.3.1). This is consistent with the BWR ISTS, Reference I.6.

A.3 Not used.

A.4 The revised presentation of CTS 3.1.3.5 Action a.1.a)1) based on the BWR ISTS, Reference I.6, does not explicitly detail options to "restore...to OPERABLE status." This action is always an option, and is implied in all Actions. Omitting this action from the TS is purely editorial.

A.5 Proposed TS Section 3/4.1.3.5 does not contain the equivalent "default" action (i.e., "Otherwise, be in at least HOT SHUTDOWN within the next 12 hours") for failure to perform the CTS 3.1.3.5 Action a.1.a)2) 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 (i.e., CTS Action a.1.b)) is inconsequential and considered administrative.

A.6 The CTS 3.1.3.5 Action a.2.a) requirement to verify that a CRD pump is operating has been maintained, but the method for verifying this has been changed from inserting one control rod one notch by drive water pressure within the normal operating range to verifying that charging water header pressure is at least 940 psig. These methods both assure that sufficient CRD 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.7 CTS 3.1.3.5 Action a.2 requires the affected control rod to be declared inoperable. Once declared inoperable, the CTS 3.1.3.1 Actions for an inoperable control rod are required to be taken. The CTS 3.1.3.1 and revised TS Section 3/4.1.3.1 Actions for an inoperable control rod contain the requirements to insert and disarm, as well as a shutdown requirement if the Actions are not performed. The proposed TS Section 3/4.1.3.5 Actions for inoperable accumulators do not need to repeat the revised TS Section 3/4.1.3.1 Actions to insert and disarm, or shutdown the unit if the inoperable control rod is not inserted and disarmed. Therefore, CTS 3.1.3.5 Action a.2.b) has been deleted. Since this change is a presentation preference only, it is considered administrative.

A.8 These conditions of CTS 4.1.3.5.a, which specify when the accumulator SR 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 4.0.3. Therefore, the stated exception has been deleted.

Revised TS Section 3/4.1.3.5 – TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The Proposed TS Section 3/4.1.3.5 Action a.1.a) for an inoperable control rod accumulator only provides an eight hour allowance to essentially restore the inoperable accumulator if the reactor pressure is sufficiently high to support control rod insertion. CTS 3.1.3.5 Action a.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 revised TS Section 3/4.1.3.1 and revised TS Section 3/4.1.3.2 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, proposed TS Section 3/4.1.3.5 Action a.1 applies to one inoperable accumulator at sufficiently high reactor pressures. proposed TS Section 3/4.1.3.5 Action a.3 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., proposed TS Section 3/4.1.3.5 Action a.4).

Revised TS Section 3/4.1.3.5 – TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 CTS 3.1.3.5 Action a.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 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., proposed TS Section 3/4.1.3.5 Action a.1.a) in lieu of declaring the control rod inoperable. Since CTS 3.1.3.5 Action a.1.a)2) to declare the control rod inoperable allows the control rod to remain withdrawn and not disarmed, proposed TS Section 3/4.1.3.5 Action a.1.a) to declare the control rod "slow" is essentially equivalent. The proposed limits and allowances for numbers and distribution of inoperable and slow control rods (i.e., found in revised TS Section 3/4.1.3.1 and revised TS Section 3/4.1.3.2, 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 proposed TS Section 3/4.1.3.5 Action a.1.a) and a.2.b) 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, proposed TS Section 3/4.1.3.5 Actions a.2 and a.3 provide actions similar to proposed TS Section 3/4.1.3.5 Action a.1, instead of the CTS 3.1.3.5 Action a.2 requirement to declare the associated control rod inoperable immediately. The requirement to declare the associated control rod inoperable (i.e., CTS 3.1.3.5 Action a.2) is maintained (i.e., proposed TS Section 3/4.1.3.5 Actions a.2.b) and a.3.b), as well as an option to declare the associated control rod "slow" (i.e., proposed TS Section 3/4.1.3.5 Action a.2.a)). 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, as opposed to the current immediate declaration of inoperability in CTS 3.1.3.5 Action a.2. 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 CRD header pressure alone is sufficient to insert control rods if a scram is required (i.e., proposed TS Section 3/4.1.3.5 Actions a.2.a) and a.3.a)).

L.2 CTS 3.1.3.5 Action a.2.a) 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. Proposed TS Section 3/4.1.3.5 reflects this difference and present Actions more appropriate to the actual plant conditions, in one instance, includes more restrictive Actions - refer to M.1 above.

CTS 3.1.3.5 Action a.2.a) is intended to identify the situation where additional scram 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, such that 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 proposed TS Section 3/4.1.3.5 Action a.3.

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 TS Section 3/4.1.3.5 Action a.2.a) 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 3.1.3.5 Action a.2.a) 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 water 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 proposed TS Section 3/4.1.3.5 Action a.2.a) is expected to be restoration of charging water 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.

3. Reactor Protection System Instrumentation (changes #9 and 12). A 2500 EFPH calibration interval has been assumed in MCPR Safety limit calculations for both LCS units with SPC methodology and will continue to be assumed for MCPR safety limit calculations for both LCS units with GE methodology. Therefore, it is appropriate to change the calibration frequency from 1000 EFPH to 2000 EFPH at LCS. This change is also consistent with CTS and ITS at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station, which both specify a calibration interval of 2000 EFPH.

4. Recirculation Loops (change #10). This is an administrative change. 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. TS Section 2.1.2 specifies both the two loop operation and the single loop operation MCPR safety limit. TS Section 3/4.1.1 refers to the COLR, which specifies the MCPR Operating limits.

5. COLR (changes #11 and 14). This is an administrative change. The deletion of methods that are no longer applicable has no adverse impact on safety. The addition of NRC approved methodology ensures that core thermal limits are appropriately determined.

6. Control Rod Scram Times (change #13). 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.

G. IMPACT ON PREVIOUS SUBMITTALS

The proposed changes affect the previous request for TS 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 Attachment B. 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 October 31, 2001, in order to support core reload with GE fuel during the LaSalle Unit 1 refueling outage which is currently scheduled to begin late November 2001. If CTS changes are approved, we will implement the changes separately for Units 1 and 2 upon startup from refuel outages scheduled in November 2001, and November, 2002, respectively. If ITS changes are approved, we will implement the changes for both Units upon startup from the Unit 1 refuel outage mentioned above.

Attachment A
Proposed Changes to Technical Specifications for
LaSalle County Station, Units 1 and 2
DESCRIPTION AND SAFETY ANALYSIS
FOR PROPOSED CHANGES

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
LaSalle County Station, Units 1 and 2

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3/4.2.4	LINEAR HEAT GENERATION RATE	3/4 2-5

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

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

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

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN, and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

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

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within 12 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

72

stuck

stuck

See safety analysis for specification 3/4.1.3.1 A.5

A.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

<general reorganization> A.2

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

add proposed Required Action Note A.4

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

1. Within 1 hour:

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

b) Disarm the associated directional control valves* either: control rod drive (CRD)

Action a.2 and a.3 L.2

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves. L.3

c) Comply with Surveillance Requirement 4.1.1.c. add proposed Required Action A.1

Action e 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours. L.3

Action c b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above: add proposed Action B

1. If the inoperable control rod(s) is withdrawn: M.2

a) Immediately verify: L.4

Action d 1) That the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rod(s) by at least two control cells in all directions, and L.1 M.1

2) The insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range**. M.3

Action c.1 and c.2

b) Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves* either: CRD LA.1

1) Electrically, or

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

add proposed Action C.1 Note A.4

---

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.6

**The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable. M.3

A.1

Revised LCO 3.1.3.1

ACTION (Continued)

Action c.2 2. If the inoperable control rod(s) is inserted: L.4

a) Within ~~1~~ 4 hour disarm the associated directional control valves either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves. CRD LA.1

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

Action e c. With more than ~~8~~ 9 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d#. With one or more SDV vent or drain lines with one valve inoperable,

1. Isolate## the associated line within 7 days.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

e#. With one or more SDV vent or drain lines with both valves inoperable, A.3

1. Isolate## the associated line within 8 hours.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. At least once per 31 days verifying each valve to be open**, and A.3

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

SR 4.1.3.1.2 4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods ~~not required to have their directional control valves disarmed electrically~~ A.8
SR 4.1.3.1.3 ~~or hydraulically~~ shall be demonstrated OPERABLE by moving each control rod at least one notch: M.5

a. At least once per 7 days, and L.5

Action a.4 b. ~~At least once per~~ 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference. L.6

~~*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~ A.6

**These valves may be closed intermittently for testing under administrative control.

#Separate Action statement entry is allowed for each SDV vent and drain line.

##An isolated line may be unisolated under administrative control to allow draining and venting of the SDV. A.3

A.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7. A.9

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per 18 months by verifying that the drain and vent valves: A.3

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open after the scram signal is reset.

A.1

REACTIVITY CONTROL SYSTEM

Revised LCO 3.1.3.1

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION <General organization> A.11

SR 4.1.3.1.4 3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds. A.12

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION a or c With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and

2. ~~Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.~~ L8

~~Otherwise, be in at least HOT SHUTDOWN within 12 hours.~~

SURVEILLANCE REQUIREMENTS <add proposed SR 3.1.3.4> A.13

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS* or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods 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

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

<See LCO 3.1.3.2>

*Except normal control rod movement.

A.1

LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

SR 4.1.3.1.5 3.1.3.6 All control rods shall be coupled to their drive mechanisms. A.14

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*. Retained in LCO 3.1.3.6

ACTION:

Action C a. In OPERATIONAL CONDITIONS 1 and 2 with one control rod not coupled to its associated drive mechanism: L.4

1. Within 2 hours, either: L.10

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.15 L.11

(1) Observing any indicated response of the nuclear instrumentation, and L.12

2) Demonstrating that the control rod will not go to the overtravel position. L.11

Action C b) If recoupling is not accomplished on the first attempt or, if not permitted by the RWM then until permitted by the RWM, declare the control rod inoperable and insert the control rod and disarm the associated directional control valves either: L.10 CRD LA.1

1) Electrically, or

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

Action e 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours, either: Retained in LCO 3.1.3.6

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

2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

a) Electrically, or

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

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2. Retained in LCO 3.1.3.6

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.6

A.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS

SR 4.1.3.1.5 4.1.3.6 A control rod shall be demonstrated to be coupled to its drive mechanism by ~~observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then~~ verifying that the control rod drive does not go to the overtravel position: L.12

a. ~~Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,~~ A.16

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. 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.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

SR 4.1.3.1.b 3.1.3.7 The control rod position indication system shall be OPERABLE. A.17

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*. Retained in LCO 3.1.3.7

ACTION:

ACTION C a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within one hour:

1. Determine the position of the control rod by:

(a) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator, LA.2

(b) Returning the control rod, by single notch movement, to its original position, and

(c) Verifying no control rod drift alarm at least once per 12 hours, or L.13

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

3. When THERMAL POWER is:

Action C.1 Note (a) Within the low power setpoint of the RWM:

1) Declare the control rod inoperable. M.6

2) Verify the position and bypassing of control rods with inoperable "Full in" and/or "Full out" position indicators by a second licensed operator or other technically qualified member of the unit technical staff. A.19

(b) Greater than the low power setpoint of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

1) Electrically, or

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

CRD LA.1

ACTION E 4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

Retained in LCO 3.1.3.7

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE STATUS. A.6

A.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

Retained in LCO 3.1.3.7

SURVEILLANCE REQUIREMENTS

SR 4.1.3.1.1 4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

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

~~b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and~~

~~c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.~~

~~d. That the control rod position indicator corresponds to the control rod position indicated by the "Full in" position indicator:~~

~~1. Prior to each reactor startup, and~~

~~2. Each time a control rod is fully inserted.~~

L.14

Retained in LCO 3.1.3.7

*At least each withdrawn control rod not applicable to control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES A.1

Revised LCO 3.1.3.2

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

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

(See LCO 3.1.3.1)

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be 800 demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and during single control rod scram time tests, the control rod drive pumps isolated from the accumulators: M.1

Note to Surveillance Requirements

SR 4.1.3.2.4, SR 4.1.3.2.1 a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

SR 4.1.3.2.5 b. For specifically affected individual control rods prior to exceeding 40% RTP 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 M.2

add proposed SR 4.1.3.2.3 M.2

SR 4.1.3.2.2 c. For at least 10% of the control rods on a rotating basis at least once per 120 days of operation. LA.1

SR 4.1.3.2.4 *Except normal control rod movement L.1

Page 1 of 6

A.1

Revised LCO 3.1.3.2

REACTIVITY CONTROL SYSTEM

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

Add proposed LCO 3.1.3.2 and Table 3.1.3-1

M.3

~~3.1.3.3 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:

Footnote(a) to Table 3.1.3

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 4.1.3.2.1, SR 4.1.3.2.2, and SR 4.1.3.2.4

A.1

Revised LCO 3.1.3.2

REACTIVITY CONTROL SYSTEM

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.1.3.2 and Table 3.1.3.2-1

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Footnote(a) to Table 3.1.3.2-1

M.3

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

M.3

1. Declare the control rods with the slower than 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. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 4.1.3.2.1, SR 4.1.3.2.2, SR 4.1.3.2.4

Page 3 of 6

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	Q	R	1, 2, 5
9. Turbine Stop Valve - Closure[i]	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low[i]	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

2000

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).

(g) Measure and compare core flow to rated core flow.

(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

(i) At least once per 18 months, verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low Trip Functions are not bypassed when THERMAL POWER is $\geq$ 25% of RATED THERMAL POWER. Specification 4.0.2 applies to this 18-month interval.

* The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

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

1. Within four (4) hours:

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

b) Increase ~~the~~ MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.2, and
to the applicable single loop operation MCPR Safety Limit

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

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

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

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

b. With no reactor coolant recirculation loops in operation:

1. Take the ACTION required by Specification 3.4.1.5, and

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

Monthly Operating Report (Continued)

A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.

6. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

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

(2) The minimum Critical Power Ratio (MCPR) scram time, dependent MCPR limits, and power and flow dependent MCPR limits for Technical Specification 3.2.3. Effects of analyzed equipment out of service are included.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

(4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. For LaSalle County Station Unit 1, the topical reports are:

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

(2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

(3) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

(4) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

Core Operating Limits Report (Continued)

(5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.

(1) ~~(6)~~ Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

(7) Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.

(8) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

(9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

(2) ~~(10)~~ Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.

(11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

(3) ~~(12)~~ RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

(13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

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

(4) ~~(15)~~ Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

Core Operating Limits Report (Continued)

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(16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

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

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

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

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

(9) ~~(21)~~ Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

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

e

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

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

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

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

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. 0.38% delta k/k with the highest worth rod analytically determined, or

b. 0.28% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

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

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

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN, and insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

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

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within 500 MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the specified limit.

c. Within ~~12~~ 72 hours after detection of a withdrawn control rod that is ~~immovable, as a result of excessive friction or mechanical interference, or is untrippable~~ stuck, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the ~~immovable or untrippable~~ stuck control rod.

See Safety analysis for Specification 3/4.1.3.1

A.5

A.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM
3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY <general reorganization> A.2

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

add proposed Required Action A.1 Note A.4

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

1. Within 1 hour:

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

Actions a.2 and a.3 L.2

b) Disarm the associated directional control valves* either: control rod drive (CRD)

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves. LA.1

c) Comply with Surveillance Requirement 4.1.1.c. add proposed Action a.1 L.3

Action e 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours. M.2

Action c b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above: add proposed Action b

1. If the inoperable control rod(s) is withdrawn:

a) Immediately verify: L.4

(Action d) 1) That the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rod(s) by at least two control cells in all directions, and L.1 M.1

Actions c.1 and c.2

2) The insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range**. M.3

b) Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves* either: CRD LA.1

1) Electrically, or

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

add proposed Action c.1 Note

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.4

**The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable. A.6 M.3

A.1

REACTIVITY CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

Action c.2 2. If the inoperable control rod(s) is inserted: L.4

a) Within ~~1~~ 4 hour disarm the associated directional control valves either: CRD LA.1

1) Electrically, or

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

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

Action e c. With more than ~~8~~ 9 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d[#]. With one or more SDV vent or drain lines with one valve inoperable, A.3

1. Isolate[##] the associated line within 7 days.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

e[#]. With one or more SDV vent or drain lines with both valves inoperable,

1. Isolate[##] the associated line within 8 hours.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by: A.3

a. At least once per 31 days verifying each valve to be open[**], and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

SR 4.1.3.1.2 4.1.3.1.2 When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed A.8
SR 4.1.3.1.3 electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch: M.5

a. At least once per 7 days, and L.5

Action a.4 b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference. L.6

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.6

**These valves may be closed intermittently for testing under administrative control. A.3

#Separate Action statement entry is allowed for each SDV vent and drain line.

##An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

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A.1

Revised LCO 3.1.3.1

SURVEILLANCE REQUIREMENTS (Continued)

~~4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.~~ A.9

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE at least once per 18 months by verifying that the drain and vent valves: A.3

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open after the scram signal is reset.

 A.1 Revised LCO 3.1.3.1

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

<general organization> A.11

SR 4.1.3.1.4 3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds. A.12

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION a or c With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds. L.8

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

SURVEILLANCE REQUIREMENTS

add proposed SR 3.1.3.4 A.13

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS* or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods 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

c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

<See LCO 3.1.3.2>

---

*Except normal control rod movement.

A.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

SR 4.1.3.1.5 3.1.3.6 All control rods shall be coupled to their drive mechanisms. A.14

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*. Retained in LCO 3.1.3.6

ACTION:

Action C a. In OPERATIONAL CONDITIONS 1 and 2 with one control rod not coupled to its associated drive mechanism: L.4

1. Within 2 hours, either: L.10

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.15 L.11

(1) Observing any indicated response of the nuclear instrumentation, and L.12

(2) Demonstrating that the control rod will not go to the overtravel position. L.11

Action C b) If recoupling is not accomplished on the first attempt or, if not permitted by the RWM then until permitted by the RWM, declare the control rod inoperable and insert the control rod and disarm the associated directional control valves either: L.10 CRD LA.1

1) Electrically, or

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

Action E 2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours, either: Retained in LCO 3.1.3.

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

2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves** either:

a) Electrically, or

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

---

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2. Retained in LCO 3.1.3.6

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status. A.6

A.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS

SR 4.1.3.1.5 4.1.3.6 A control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position: L.12

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity, A.16

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. 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.1

Revised LCO 3.1.3.1

REACTIVITY CONTROL SYSTEM

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

SR 4.1.3.1.1 3.1.3.7 The control rod position indication system shall be OPERABLE. A.17

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*. Retained in LCO 3.1.3.7

ACTION:

Action C a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable within one hour:

1. Determine the position of the control rod by:

(a) ~~Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,~~ LA.2

(b) ~~Returning the control rod, by single notch movement, to its original position, and~~

(c) ~~Verifying no control rod drift alarm at least once per 12 hours, or~~ L.13

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

3. When THERMAL POWER is:

Action C.1 Note

(a) Within the low power setpoint of the RWM:

(1) Declare the control rod inoperable. M.6

(2) Verify the position and bypassing of control rod with inoperable "Full in" and/or "Full out" position indicators by a second licensed operator or other technically qualified member of the unit technical staff. A.19

b) Greater than the low power setpoint of the RWM, declare the control rod inoperable, insert the control rod and disarm the associated ~~directional control valves**~~ either: CRD LA.1

~~(1) Electrically, or~~

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

Action e 4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2. Retained in LCO 3.1.3.7

~~**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.~~ A.6

A.1

Revised LCO 3.1.3.1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

Retained in LCO 3.1.3.7

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

SURVEILLANCE REQUIREMENTS

SR 4.1.3.1.1 4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

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

L.14

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.

d. That the control rod position indicator corresponds to the control rod position indicated by the "Full in" position indicator:

1. Prior to each reactor startup, and
2. Each time a control rod is fully inserted.

Retained in LCO 3.1.3.7

*At least each withdrawn control rod not applicable to control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

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REACTIVITY CONTROL SYSTEM

Revised LCO 3.1.3.2

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES $\boxed{A.1}$

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 05, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds:

1. Declare the control rod(s) with the slow insertion time inoperable, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

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

See LCO 3.1.3.1

SURVEILLANCE REQUIREMENTS

800 — M.1

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

Logic is surveillance requirements

SR 4.1.3.2.4

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

prior to exceeding 40% RTP — M.2

SR 4.1.3.2.1

SR 4.1.3.2.5

b. For specifically affected individual control rods 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

add proposed SR 4.1.3.2.3 — M.2

SR 4.1.3.2.2

c. For at least 10% of the control rods, on a rotating basis at least once per 120 days of operation.

LA.1

SR 4.1.3.2.4

Except normal control rod movement. — L.1

Page 4 of 6

A.1

Revised LCO 3.1.3.2

REACTIVITY CONTROL SYSTEM

~~CONTROL ROD AVERAGE SCRAM~~ INSERTION TIMES

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.1.3.2 and Table 3.1.3.2-1

3.1.3.3 ~~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:~~

Footnote(s) to Table 3.1.3.2-1

M.3

~~Position Inserted From Fully Withdrawn~~	~~Average Scram Insertion Time (Seconds)~~
~~45~~	~~0.43~~
~~39~~	~~0.86~~
~~25~~	~~1.93~~
~~05~~	~~3.49~~

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 4.1.3.2.1, SR 4.1.3.2.2, and SR 4.1.3.2.4

Page 5 of 6

A.1

Revised LCO 3.1.3.2

REACTIVITY CONTROL SYSTEM

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

add proposed LCO 3.1.3.2, and Table 3.1.3.2-1

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, ~~shall not exceed any of the following:~~

Footnote(s) to Table 3.1.3.2-1

M.3

~~Position Inserted From Fully Withdrawn~~	~~Average Scram Insertion Time (Seconds)~~
~~45~~	~~0.45~~
~~39~~	~~0.92~~
~~25~~	~~2.05~~
~~05~~	~~3.70~~

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

Action a

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

M.3

1. ~~Declare the control rods with the slower than 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. ~~Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.~~

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

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.1.3.2.

SR 4.1.3.2.1, SR 4.1.3.2.2, SR 4.1.3.2.4

A.1

Revised LCO 3.1.3.5

REACTIVITY CONTROL SYSTEM

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*

ACTION:

ACTION A a. In OPERATIONAL CONDITION 1 or 2:

1. With one control rod scram accumulator inoperable: and reactor pressure ≥ 900 psig M.1

a) Within 8 hours, either:

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

add proposed Required Action A.1

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

~~b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.~~ A.5

ACTIONS and 2. With more than one control rod scram accumulator inoperable, declare the associated control rod inoperable and: withdrawn M.1 L.1

add proposed Required Action B.2.b

a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD pump is operating ~~by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range~~ or place the reactor mode switch in the Shutdown position. Action L.2 A.6

a.4. note

Action A4

~~b) Insert the inoperable control rods and disarm the associated directional control valves either:~~ A.7

~~1) Electrically, or~~

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

~~Otherwise, be in at least HOT SHUTDOWN within 12 hours.~~

Action b. In OPERATIONAL CONDITION 5 with:

1. One withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within 1 hour, ~~either:~~

~~a) Electrically, or~~ A.2

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

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

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

A.1

Revised LCO 3.1.3.5

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

SR 4.1.3.5.1 a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed. A.8

A.1

Revised LCO 3.1.3.5

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

Action a

1. With one control rod scram accumulator inoperable: and reactor pressure ≥900 psig M.1

a) Within 8 hours, either:

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

add proposed Required Action A.1

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

b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours. A.5

A.1

Actions b and c

2. With more than one control rod scram accumulator inoperable, declare the associated control rod inoperable and: within 1 hour L.1

add proposed Required Action A.2.b

a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one CRD pump is operating by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range or place the reactor mode switch in the Shutdown position. L.2 A.6

Action a.4 Note

Action a.4

b) Insert the inoperable control rods and disarm the associated directional control valves either: A.7

1) Electrically, or

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

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

Action

b. In OPERATIONAL CONDITION 5 with:

1. One withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within 1 hour, either:

a) Electrically, or

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

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

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Revised LCO 3.1.3.5

REACTIVITY CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

SR 4.1.3.5.1 a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig ~~unless the control rod is inserted and disarmed or scrammed~~. AB

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	Q	R	1, 2, 5
9. Turbine Stop Valve - Closure[i]	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low[i]	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 2000 effective full power hours (EFPH).
(g) Measure and compare core flow to rated core flow.
(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.
(i) At least once per 18 months, verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low Trip Functions are not bypassed when THERMAL POWER is ≥ 25% of RATED THERMAL POWER. Specification 4.0.2 applies to this 18-month interval.

* The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

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

1. Within four (4) hours:

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

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

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

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

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

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

b. With no reactor coolant recirculation loops in operation:

1. Take the ACTION required by Specification 3.4.1.5, and

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

Core Operating Limits Report (Continued)

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

(2) The minimum Critical Power Ratio (MCPR) scram time dependent MCPR limits, and power and flow dependent MCPR limits for Technical Specification 3.2.3. Effects of analyzed equipment out of service are included.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

(4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. For LaSalle County Station Unit 2, the topical reports are:

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

(2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.

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(3) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation November 1990.

(4) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

(5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.

(1) ~~(6)~~ Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

(7) Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.

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(8) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

Core Operating Limits Report (Continued)

e (9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

(2) ~~(10)~~ Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.

e (11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

(3) ~~(12)~~ RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

(13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

e (14) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

(4) ~~(15)~~ Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

e (16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

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

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

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

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

(9) ~~(21)~~ Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

Core Operating Limits Report (Continued)

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

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

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

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

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

3.1.3.1 Each control rod shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one withdrawn control rod stuck*:

    1. Immediately verify that stuck control rod separation criteria are met, and

    2. Within 2 hours, disarm the associated control rod drive (CRD), and

    3. Within 72 hours, perform Surveillance Requirement 4.1.1.c, and

    4. 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 4.1.3.1.2 and Surveillance Requirement 4.1.3.1.3 for each withdrawn OPERABLE control rod.

b. With two or more withdrawn control rods stuck, be in at least HOT SHUTDOWN within 12 hours.

c. With one or more control rods inoperable for reasons other than being stuck in the withdrawn position:

    1. Within 3 hours, fully insert the inoperable control rod(s) **, and

    2. Within the next 1 hour, disarm the associated CRD(s).

d. 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 ***:

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

e. With the required provisions of ACTION a, c, or d not met, or with nine or more control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

---

* The rod worth minimizer (RWM) may be bypassed as allowed by Specification 3/4.1.4 to allow continued operation.

** The RWM may be bypassed as allowed by Specification 3/4.1.4 to allow insertion of inoperable control rod and continued operation.

*** Not applicable when THERMAL POWER > 10% RTP.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each control rod shall be determined at least once per 24 hours.

4.1.3.1.2 Insert each fully withdrawn control rod at least one notch at least once per 7 days.###

4.1.3.1.3 Insert each partially withdrawn control rod at least one notch at least once per 31 days.####

4.1.3.1.4 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 4.1.3.2.1, 4.1.3.2.2, 4.1.3.2.3, 4.1.3.2.4, and 4.1.3.2.5.

4.1.3.1.5 Verify 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.

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.
####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.

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3.1.3.2.a No more than 12 OPERABLE control rods shall be "slow," in accordance with Table 3.1.3.2-1; and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTIONS:

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

SURVEILLANCE REQUIREMENTS: *

4.1.3.2.1 Verify each control rod scram time is within the limits of Table 3.1.3.2-1 with reactor steam dome pressure ≥ 800 psig prior to exceeding 40% RTP after each reactor shutdown ≥ 120 days.

4.1.3.2.2 Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.3.2-1 with reactor steam dome pressure ≥ 800 psig, at least once per 120 days of cumulative operation in OPERATIONAL CONDITION 1.

4.1.3.2.3 Verify each affected control rod scram time is within the limits of Table 3.1.3.2-1 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.

4.1.3.2.4 Verify each affected control rod scram time is within the limits of Table 3.1.3.2-1 with reactor steam dome pressure ≥ 800 psig prior to exceeding 40% RTP after fuel movement within the affected core cell.

4.1.3.2.5 Verify each affected control rod scram time is within the limits of Table 3.1.3.2-1 with reactor steam dome pressure ≥ 800 psig prior to exceeding 40% RTP after work on control rod or CRD System that could affect scram time.

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

Table 3.1.3.2-1
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 3.1.3.1, "Control Rod Operability," for control rods with scram times > 7 seconds to 90% insertion. These control rods are inoperable, in accordance with Surveillance Requirement 4.1.3.1.4, and are not considered "slow."

Notch Position	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
45	0.41	0.52
39	0.80	0.86
25	1.77	1.91
05	3.20	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.

3.1.3.3 Each scram discharge volume (SDV) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION:

a#. With one or more SDV vent or drain lines with one valve inoperable,

1. Isolate## the associated line within 7 days.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

b#. With one or more SDV vent or drain lines with both valves inoperable,

1. Isolate## the associated line within 8 hours.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 The scram discharge volume vent and drain valves shall be demonstrated OPERABLE by:

a. At least once per 31 days verifying each valve to be open*, and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.3.2 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume vent and drain valves OPERABLE at least once per 18 months by verifying that the drain and vent valves:

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open after the scram signal is reset.

Separate Action statement entry is allowed for each SDV vent and drain line.
An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.
* These valves may be closed intermittently for testing under administrative control.

3.1.3.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5 *.

ACTIONS:

a. In OPERATIONAL CONDITION 1 OR 2:

1. With one control rod scram accumulator inoperable with reactor steam dome pressure $\geq$ 900 psig:

    a) Within 8 hours, declare the associated control rod scram time "slow," ** or declare the associated control rod inoperable.

2. With two or more control rod scram accumulators inoperable with reactor steam dome pressure $\geq$ 900 psig:

    a) Within 20 minutes from discovery of two or more inoperable accumulators with reactor steam dome pressure $\geq$ 900 psig concurrent with charging water header pressure < 940 psig, restore charging water header pressure to $\geq$ 940 psig, and

    b) Within 1 hour, declare the associated control rod scram time "slow," ** or declare the associated control rod inoperable.

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

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

    b) Within 1 hour, declare the associated control rod inoperable.

4. With the required provisions of ACTION a.2.a) or a.3.a) not met, immediately place the reactor mode switch in the shutdown position.***

b. In OPERATIONAL CONDITION 5*:

1. With one withdrawn control rod and its associated scram accumulator inoperable, fully insert and disarm the affected control rod within one hour.****

2. 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.

---

* In OPERATIONAL CONDITION 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Only applicable if the associated control rod scram time was within the limits of Table 3.1.3.2-1 during the last scram time surveillance.

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

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

4.1.3.5.1 Verify each control rod scram accumulator pressure is $\geq$ 940 psig at least once per 7 days.

3.1.3.6 All control rod drives shall be coupled to their drive mechanisms

APPLICABILITY: OPERATIONAL CONDITION 5 *

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.

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 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:

a. Anytime the control rod is withdrawn to the "full out" position, and

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

* In OPERATIONAL CONDITION 5, this specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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LIMITING CONDITION FOR OPERATION

3.1.3.7 All control rod position indicators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 5 *

ACTION:

a. With a withdrawn control rod position indicator inoperable:

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

2. Fully insert the control rod.

SURVEILLANCE REQUIREMENTS

4.1.3.7.1 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.

* In OPERATIONAL CONDITION 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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3.1.3.1 Each control rod shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one withdrawn control rod stuck*:

   1. Immediately verify that stuck control rod separation criteria are met, and

   2. Within 2 hours, disarm the associated control rod drive (CRD), and

   3. Within 72 hours, perform Surveillance Requirement 4.1.1.c, and

   4. 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 4.1.3.1.2 and Surveillance Requirement 4.1.3.1.3 for each withdrawn OPERABLE control rod.

b. With two or more withdrawn control rods stuck, be in at least HOT SHUTDOWN within 12 hours.

c. With one or more control rods inoperable for reasons other than being stuck in the withdrawn position:

   1. Within 3 hours, fully insert the inoperable control rod(s) **, and

   2. Within the next 1 hour, disarm the associated CRD(s).

d. 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 ***:

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

e. With the required provisions of ACTION a, c, or d not met, or with nine or more control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

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* The rod worth minimizer (RWM) may be bypassed as allowed by Specification 3/4.1.4 to allow continued operation.

** The RWM may be bypassed as allowed by Specification 3/4.1.4 to allow insertion of inoperable control rod and continued operation.

*** Not applicable when THERMAL POWER > 10% RTP.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each control rod shall be determined at least once per 24 hours.

4.1.3.1.2 Insert each fully withdrawn control rod at least one notch at least once per 7 days.###

4.1.3.1.3 Insert each partially withdrawn control rod at least one notch at least once per 31 days.####

4.1.3.1.4 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 4.1.3.2.1, 4.1.3.2.2, 4.1.3.2.3, 4.1.3.2.4, and 4.1.3.2.5.

4.1.3.1.5 Verify 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.

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.

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.

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3/4.1.3.4 DELETED

3.1.3.2.a No more than 12 OPERABLE control rods shall be "slow," in accordance with Table 3.1.3.2-1; and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTIONS:

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

SURVEILLANCE REQUIREMENTS: *

4.1.3.2.1 Verify each control rod scram time is within the limits of Table 3.1.3.2-1 with reactor steam dome pressure $\geq$ 800 psig prior to exceeding 40% RTP after each reactor shutdown $\geq$ 120 days.

4.1.3.2.2 Verify, for a representative sample, each tested control rod scram time is within the limits of Table 3.1.3.2-1 with reactor steam dome pressure $\geq$ 800 psig, at least once per 120 days of cumulative operation in OPERATIONAL CONDITION 1.

4.1.3.2.3 Verify each affected control rod scram time is within the limits of Table 3.1.3.2-1 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.

4.1.3.2.4 Verify each affected control rod scram time is within the limits of Table 3.1.3.2-1 with reactor steam dome pressure $\geq$ 800 psig prior to exceeding 40% RTP after fuel movement within the affected core cell.

4.1.3.2.5 Verify each affected control rod scram time is within the limits of Table 3.1.3.2-1 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.

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

Table 3.1.3.2-1
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 3.1.3.1, "Control Rod Operability," for control rods with scram times > 7 seconds to 90% insertion. These control rods are inoperable, in accordance with Surveillance Requirement 4.1.3.1.4, and are not considered "slow."

Notch Position	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
45	0.41	0.52
39	0.80	0.86
25	1.77	1.91
05	3.20	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.

3.1.3.3 Each scram discharge volume (SDV) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION:

a#. With one or more SDV vent or drain lines with one valve inoperable,

1. Isolate ## the associated line within 7 days.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

b#. With one or more SDV vent or drain lines with both valves inoperable,

1. Isolate## the associated line within 8 hours.
2. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 The scram discharge volume vent and drain valves shall be demonstrated OPERABLE by:

a. At least once per 31 days verifying each valve to be open*, and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.3.2 The scram discharge volume shall be determined OPERABLE by demonstrating the scram discharge volume vent and drain valves OPERABLE at least once per 18 months by verifying that the drain and vent valves:

a. Close within 30 seconds after receipt of a signal for control rods to scram, and

b. Open after the scram signal is reset.

Separate Action statement entry is allowed for each SDV vent and drain line.
An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.
* These valves may be closed intermittently for testing under administrative control.

3.1.3.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5 *.

ACTIONS:

a. In OPERATIONAL CONDITION 1 OR 2:

1. With one control rod scram accumulator inoperable with reactor steam dome pressure $\geq$ 900 psig:

   a) Within 8 hours, declare the associated control rod scram time "slow," ** or declare the associated control rod inoperable.

2. With two or more control rod scram accumulators inoperable with reactor steam dome pressure $\geq$ 900 psig:

   a) Within 20 minutes from discovery of two or more inoperable accumulators with reactor steam dome pressure $\geq$ 900 psig concurrent with charging water header pressure < 940 psig, restore charging water header pressure to $\geq$ 940 psig, and

   b) Within 1 hour, declare the associated control rod scram time "slow," ** or declare the associated control rod inoperable.

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

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

   b) Within 1 hour, declare the associated control rod inoperable.

4. With the required provisions of ACTION a.2.a) or a.3.a) not met, immediately place the reactor mode switch in the shutdown position.***

b. In OPERATIONAL CONDITION 5*:

1. With one withdrawn control rod and its associated scram accumulator inoperable, fully insert and disarm the affected control rod within one hour.****

2. 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.

---

* In OPERATIONAL CONDITION 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Only applicable if the associated control rod scram time was within the limits of Table 3.1.3.2-1 during the last scram time surveillance.

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

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

4.1.3.5.1 Verify each control rod scram accumulator pressure is $\geq$ 940 psig at least once per 7 days.

3.1.3.6 All control rod drives shall be coupled to their drive mechanisms

APPLICABILITY: OPERATIONAL CONDITION 5 *

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.

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 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:

a. Anytime the control rod is withdrawn to the "full out" position, and

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

* In OPERATIONAL CONDITION 5, this specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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3.1.3.7 All control rod position indicators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 5*

ACTION:

a. With a withdrawn control rod position indicator inoperable:

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

   2. Fully insert the control rod.

SURVEILLANCE REQUIREMENTS

4.1.3.7.1 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.

* In OPERATIONAL CONDITION 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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<u>Attachment B-2</u>
Proposed Changes to Technical Specifications for LaSalle County Station, Units 1 and 2

MARKED-UP IMPROVED TECHNICAL SPECIFICATIONS PAGES FOR PROPOSED CHANGES

<u>REVISED PAGES</u>

<u>MARKED UP REVISED PAGES</u>
3.1.4-3
3.3.1.1-4
5.6-5

<u>TYPED REVISED PAGES</u>
3.1.4-3
3.3.1.1-4
5.6-6

Table 3.1.4-1
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 notch position 05. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	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.
45	0.41	0.52
39	0.80	0.86
25	1.77	1.91
05	3.20	3.43

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

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs
SR 3.3.1.1.7	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	2000 1000 effective full power hours
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	92 days

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

16. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

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

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

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

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

21. Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

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

23. BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

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

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

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

(continued)

Table 3.1.4-1
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 notch position 05. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	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
45	0.41	0.52
39	0.80	0.86
25	1.77	1.91
05	3.20	3.43

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

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.5	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs
SR 3.3.1.1.7	-----------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. --------------------------------------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	2000 effective full power hours
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	92 days

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

26. 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.

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 Changes to Technical Specifications for
LaSalle County Station, Units 1 and 2
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 LaSalle County Station (LCS) Units 1 and 2 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 being reviewed by the NRC. The proposed changes are briefly summarized as follows:

Proposed Changes to CTS

1. SDM, Control Rod Operability and Scram Insertion Time Methodology. CTS Sections 4.1.1.c, "Shutdown Margin," 3/4.1.3.1, "Control Rod Operability," 3/4.1.3.2, "Maximum Scram Insertion Times," 3/4.1.3.3, "Average Scram Insertion Times," 3/4.1.3.4, "Group Scram Insertion Times," 3/4.1.3.5, Control Rod Scram Accumulators," 3/4.1.3.6, "Control Rod Coupling," and 3/4.1.3.7, "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.

2. Control Rod Scram Insertion Times. In addition to change #1 above, scram times are revised to add the required scram times for GE analyzed cores.

3. Local Power Range Monitor Calibration (LPRM) Frequency. CTS Section 3/4.3.1, "Reactor Protection System Instrumentation," is revised to reduce the frequency of calibration of the LPRMs from once every 1000 EFPH to once every 2000 EFPH.

4. Recirculation Loops. CTS Section 3/4.4.1., "Recirculation Loops," is revised to refer the safety limits section and the Core Operating Limits Report (COLR) for the value of the Minimum Critical Power Ratio (MCPR) during single loop operation.

5. COLR. CTS 6.6.A.6, "Core Operating Limits Report," is revised to remove references to SPC methodology that will no longer be applicable and add GE's methodology for determining critical power for SPC fuel.

Proposed Changes to ITS

1. Control Rod Scram Times. ITS 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.

2. LPRM Calibration Frequency. ITS Section 3.3.1.1, "Reactor Protection System Instrumentation," SR 3.3.1.1.8, is revised to reduce the frequency of calibration of the LPRMs from once every 1000 EFPH to once every 2000 EFPH.

3. COLR. ITS Section 5.6.5, "Core Operating Limits Report," is revised to add GE's methodology for determining critical power for SPC fuel.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

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

Evaluation of effect on the probability of an accident:

Proposed Changes to CTS

1. SDM, Control Rod Operability and Scram Insertion Time Methodology. The changes to CTS Sections 4.1.1.c, 3/4.1.3.1, 3/4.1.3.2, 3/4.1.3.3, 3/4.1.3.4, 3/4.1.3.5, 3/4.1.3.6, and 3/4.1.3.7, 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.

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

3. Local Power Range Monitor Calibration (LPRM) Frequency. The change to Section 3/4.3.1 only revises the calibration frequency requirement for core monitoring instrumentation. Core monitoring instrumentation is not an accident initiator. No other plant systems, structures or components are affected by this changed. Therefore the probability of an accident is not increased.

4. Recirculation Loops. The changes to CTS Section 3/4.4.1 are administrative changes and do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes. Therefore the probability of an accident is not increased.

5. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable and addition of NRC approved methodology has no adverse impact on safety. Therefore the probability of an accident is not increased.

Proposed Changes to ITS

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

2. LPRM Calibration Frequency. The change to SR 3.3.1.1 only revises the calibration frequency requirement for core monitoring instrumentation. Core monitoring instrumentation is not an accident initiator. No other plant systems, structures or components are affected by this changed. Therefore the probability of an accident is not increased.

3. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable and addition of NRC approved methodology has no adverse impact on safety. Therefore the probability of an accident is not increased.

Evaluation of effect on the consequences of an accident:

Proposed Changes to CTS

1. SDM, Control Rod Operability and Scram Insertion Time Methodology. The changes to CTS Sections 4.1.1.c, 3/4.1.3.1, 3/4.1.3.2, 3/4.1.3.3, 3/4.1.3.4, 3/4.1.3.5, 3/4.1.3.6, and 3/4.1.3.7 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 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 UFSAR accident or transient.

   In addition, numerous changes to the control rod operability and scram timing specifications 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.

2. Control Rod Scram Insertion 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 UFSAR events, 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.

3. LPRM Calibration Frequency. The change to Section 3/4.3.1 does not affect the consequences of an accident. The larger calibration interval, 2000 EFPH, (2500 EFPH is assumed) is factored into the uncertainties used in the analyses for the cycle specific MCPR safety limit and will continue to be factored into the MCPR safety limit uncertainties for future cycle calculations. Therefore, the potential impacts on the MCPR safety limit as a result of this change are already included in the calculations. Including these uncertainties in the MCPR safety limit ensures that the fuel is properly protected from the consequences of accidents or transients with LPRM calibration intervals of up to 2500 EFPH. No other plant systems, structures or components are affected by this changed.

4. Recirculation Loops. The changes to CTS Section 3/4.4.1 are administrative changes and do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes. Therefore the consequences of an accident are not increased.

5. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable has no adverse impact on safety. Therefore the consequences of an accident are not increased.

Proposed Changes to ITS

1. 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 UFSAR events, 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.

2. LPRM Calibration Frequency. The change to ITS SR 3.3.1.1 does not affect the consequences of an accident. The larger calibration interval, 2000 EFPH (i.e., 2500 EFPH is assumed), is factored into the uncertainties used in the analyses for the cycle specific MCPR safety limit and will continue to be factored into the MCPR safety limit uncertainties for future cycle calculations. Therefore, the potential impacts on the MCPR safety limit as a result of this change are already included in the calculations. Including these uncertainties in the MCPR safety limit ensures that the fuel is properly protected from the consequences of accidents or transients with LPRM calibration intervals of up to 2500 EFPH. No other plant systems, structures or components are affected by this changed.

3. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable and addition of NRC approved methodology has no adverse impact on safety. Therefore the consequences of an accident are not increased.

Therefore, the proposed changes 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?

Proposed Changes to CTS

1. SDM, Control Rod Operability and Scram Insertion Time Methodology. The changes to CTS Sections 4.1.1.c, 3/4.1.3.1, 3/4.1.3.2, 3/4.1.3.3, 3/4.1.3.4, 3/4.1.3.5, 3/4.1.3.6, and 3/4.1.3.7 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.

2. Control Rod Scram Insertion 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.

3. LPRM Calibration Frequency. The change to CTS Section 3/4.3.1 does not create the possibility of a new or different kind of accident from any accident previously evaluated. The larger calibration interval, 2000 EFPH (2500 EFPH is assumed), is factored into the uncertainties used in the analyses for the cycle specific MCPR safety limit and will continue to be factored into the MCPR safety limit uncertainties for future cycle calculations. Therefore, the potential impacts on the MCPR safety limit as a result of this change are already included in the calculations. No other plant systems, structures or components are affected by this changed.

4. Recirculation Loops. The changes to the CTS Section 3/4.4.1 are administrative changes and will not create the possibility of a new or different kind of accident from any accident previously evaluated. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

5. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable has no adverse impact on safety. Therefore the possibility of a new or different kind of accident is not created.

Proposed Changes to ITS

1. 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.

2. LPRM Calibration Frequency. The change to ITS SR 3.31.1 does not create the possibility of a new or different kind of accident from any accident previously evaluated. The larger calibration interval, 2000 EFPH (2500 EFPH is assumed), is factored into the uncertainties used in the analyses for the cycle specific

MCPR safety limit and will continue to be factored into the MCPR safety limit uncertainties for future cycle calculations. Therefore, the potential impacts on the MCPR safety limit as a result of this change are already included in the calculations. No other plant systems, structures or components are affected by this changed.

3. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable and addition of NRC approved methodology has no adverse impact on safety. Therefore the possibility of a new of different kind of accident is not created.

Therefore, the proposed changes 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?

Proposed Changes to CTS

1. SDM, Control Rod Operability and Scram Insertion Time Methodology. The changes to CTS Sections 4.1.1.c, 3/4.1.3.1, 3/4.1.3.2, 3/4.1.3.3, 3/4.1.3.4, 3/4.1.3.5, 3/4.1.3.6, and 3/4.1.3.7 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 specifications 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.

2. Control Rod Scram Insertion 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.

3. LPRM Calibration Frequency. The change to CTS Section 3/4.3.1 does not reduce a margin of safety. The larger calibration interval, 2000 EFPH (2500 EFPH is assumed), is factored into the uncertainties used in the analyses for the cycle specific MCPR safety limit and will continue to be factored into the MCPR safety limit uncertainties for future cycle calculations. Including these uncertainties in the MCPR safety limit ensures that the fuel is properly protected from the consequences of accidents or transients with LPRM calibration intervals of up to 2500 EFPH. By calculating the MCPR safety limit with the correct uncertainties, which account for a 2500 EFPH calibration interval, the margin to safety for the MCPR safety limit is maintained. No other plant systems, structures or components are affected by this changed.

4. Recirculation Loops. The changes to CTS Section 3/4.4.1 are administrative changes and will not reduce a margin of safety. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

5. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable has no adverse impact on safety. Therefore no margin of safety is reduced.

Attachment C
Proposed Changes to Technical Specifications for LaSalle County Station, Units 1 and 2
INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

Proposed Changes to ITS

1. 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.

2. LPRM Calibration Frequency. The change to ITS SR 3.3.1.1 does not reduce a margin of safety. The larger calibration interval, 2000 EFPH (2500 EFPH is assumed), is factored into the uncertainties used in the analyses for the cycle specific MCPR safety limit and will continue to be factored into the MCPR safety limit uncertainties for future cycle calculations. Including these uncertainties in the MCPR safety limit ensures that the fuel is properly protected from the consequences of accidents or transients with LPRM calibration intervals of up to 2500 EFPH. By calculating the MCPR safety limit with the correct uncertainties which account for a 2500 EFPH calibration interval, the margin to safety for the MCPR safety limit is maintained. No other plant systems, structures or components are affected by this changed.

3. COLR. This is a purely administrative change. The deletion of analysis methods that are no longer applicable and addition of NRC approved methodology has no adverse impact on safety. Therefore there is no reduction in the margin of safety.

Therefore, these proposed changes 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
LaSalle County Station, Units 1 and 2
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 SR , and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment C, these proposed changes do 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 revised methodologies for determining core thermal limits and control rod scram times and various related changes that are either administrative or that do not reduce any margins of safety. These changes do not allow for an increase in the unit power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do 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
LaSalle County Station, Units 1 and 2

REVISED CURRENT TECHNICAL SPECIFICATIONS BASES PAGES

B 2-2
B 2-3 (Unit 1 only)
B 3/4 1-2
B 3/4 1-3
B3/4 1-4
Insert pages (3) for B3/4 1-2, 1-3, 1-4

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ~~ANF~~ fuel vendor's Critical Power Methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the SPC-developed ANFB critical power correlation.

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-7. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document. Reference 1

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

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

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

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

2.1.2 THERMAL POWER, High Pressure and High Flow (Continued)

9

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

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

3. "General Electric Fuel Bundle Designs," NEDE-24011-P-A, (latest approved revision).

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3/4.1.3 CONTROL RODS

~~The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) the potential effects of the rod drop accident are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.~~

~~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 for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.~~

~~The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.~~

~~The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit during the limiting power transient analyzed in Section 15.0 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the fuel cladding safety limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.~~

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line. The common drain line has two valves in series. Each header is connected to a common vent line. This common header has two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram.

3/4.1.3 CONTROL RODS (Continued)

The Design Basis Accident and transient analyses assume that all of the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100; and

b. Open on scram reset to maintain the SDV vent and drain path open such that sufficient volume is available to accept the reactor coolant discharged during a scram.

The OPERABILITY of all SDV vent and drain valves ensures that, during a scram, the SDV vent and drain valves will close to contain reactor water discharged into the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to be open to ensure that a path is available for the SDV piping to drain freely at other times.

Isolation of the SDV can also be accomplished by closure of the SDV valves under administrative control. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 and adequate core cooling is maintained.

Note * contained in Specification 3.1.3.1 allows Action Statements d and e to be entered separately for each affected SDV vent and drain line, and Completion Times to be tracked on a per line basis. For instance, when a vent valve is declared inoperable, Action d is entered for the vent line and its Completion Time starts. If a drain valve is subsequently declared inoperable, Action d is entered again for the drain line and a separate Completion Time starts and is tracked for the drain line. The same is true for both valves inoperable in one line in accordance with Action e, provided the original Completion Time (if any) affecting that line is not exceeded. Also, one line can be in Action d, while the other line is in Action e, provided the applicable Completion Times are met for each line.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 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 reactors.

3/4.1.3 CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for ~~normal scram action.~~

Control rod ~~coupling integrity is~~ required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3.65 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Volume 1 and Supplements 1 and 2, March 1983.

Insert #1

3/4.1.3. Control Rods

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.

These Specifications 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.

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 $\geq$ 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 3.1.3-1 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 3.1.3-1 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 3.1.3-1 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) (i.e., one pair of control rods for the reactor) may occupy adjacent locations (face or diagonal). For reactor steam dome pressures < 800 psig, scram times are specified in the Administrative Technical Requirements.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed. 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 3.1.3-1, 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 3.1.3-1 with the reactor steam dome pressure $\leq$ 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.

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.

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 the LCO's on Control Rod OPERABILITY maximum scram times, 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.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

fuel vendor's

The Safety Limit MCPR is determined using the ~~ANF~~ critical power methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of ~~boiling transition is determined using the SPC-developed~~ ANFB critical power correlation.

3. "General Electric Fuel Bundle Design" NEDE-24011-P-A (latest approved revision)

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-7. The uncertainties used in the analyses are provided in ~~the cycle-specific transient analysis~~ parameters document. Reference 3.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A), Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

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

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

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

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

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

Insert attached #1

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) the potential effects of the rod drop accident are limited. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

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 for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit during the limiting power transient analyzed in Section 15.0 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the fuel cladding safety limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line. The common drain line has two valves in series. Each header is connected to a common vent line. This common header has two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram.

3/4.1.3 CONTROL RODS (Continued)

The Design Basis Accident and transient analyses assume that all of the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100; and

b. Open on scram reset to maintain the SDV vent and drain path open such that sufficient volume is available to accept the reactor coolant discharged during a scram.

The OPERABILITY of all SDV vent and drain valves ensures that, during a scram, the SDV vent and drain valves will close to contain reactor water discharged into the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to be open to ensure that a path is available for the SDV piping to drain freely at other times.

Isolation of the SDV can also be accomplished by closure of the SDV valves under administrative control. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 and adequate core cooling is maintained.

Note # contained in Specification 3.1.3.1 allows Action Statements d and e to be entered separately for each affected SDV vent and drain line, and Completion Times to be tracked on a per line basis. For instance, when a vent valve is declared inoperable, Action d is entered for the vent line and its Completion Time starts. If a drain valve is subsequently declared inoperable, Action d is entered again for the drain line and a separate Completion Time starts and is tracked for the drain line. The same is true for both valves inoperable in one line in accordance with Action e, provided the original Completion Time (if any) affecting that line is not exceeded. Also, one line can be in Action d, while the other line is in Action e, provided the applicable Completion Times are met for each line.

Control rods with inoperable accumulators are declared inoperable ~~and~~ Specification 3.1.3.1 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~~ reactors.

3/4.1.3 CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3.65 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Volume 1 and Supplements 1 and 2, March 1983."

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3/4.1.3. Control Rods

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.

These Specifications 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.

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 $\geq$ 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 3.1.3-1 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 3.1.3-1 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 3.1.3-1 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) (i.e., one pair of control rods for the reactor) may occupy adjacent locations (face or diagonal). For reactor steam dome pressures < 800 psig, scram times are specified in the Administrative Technical Requirements.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed. 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 3.1.3-1, 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 3.1.3-1 with the reactor steam dome pressure $\leq$ 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.

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.

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 the LCO's on Control Rod OPERABILITY maximum scram times, 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.

<u>Attachment E-2</u>
Proposed Changes to Technical Specifications for
LaSalle County Station, Units 1 and 2

REVISED IMPROVED TECHNICAL SPECIFICATIONS BASES PAGES

<u>REVISED MARKED-UP PAGES</u>

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 3.2.2-2
B 3.2.2-4
B 3.3.1.1-29
B 3.3.1.1-30

<u>REVISED TYPED PAGES</u>

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 3.2.2-2
B 3.2.2-4
B 3.2.2-5
B 3.3.1.1-29
B 3.3.1.1-30

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, application 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 in the fuel vendor's ~~ANFB~~ critical power correlation. References 2, 3, 4, ~~and~~ 5, and 6 describe the methodology used in determining the MCPR SL.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

fuel vendor's
The ~~ANFB~~ 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 e 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. e 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 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.

(continued)

BASES (continued)

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~~ 7). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence (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).

insert ref 4 (see next page)

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).

(continued)

BASES

REFERENCES (continued)

6 ~~5~~. EMF-1125(P)(A), Supplement 1 Appendix C, ANFB Critical Power Correlation Application for Coresident Fuel, Siemens Power Corporation (as specified in Technical Specification 5.6.5).

7 ~~8~~. 10 CFR 100.

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

(i.e., one pair of control rods in the core)

BASES

LCO (continued)

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 (face or diagonal) locations.

Table 3.1.4-1 is modified by two Notes, which state 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 experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in the 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 the 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 control by applying appropriate flow dependent MCPR operating limits. The MCPR operating limit for a given power/flow state is the greater of the rated conditions MCPR operating limit or the power 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.

(if necessary)

Power dependent MCPR limits ($MCPR_p$) are determined on a cycle-specific basis. These limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.

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. MCPR operating limits which include the effects of analyzed equipment out-of-service are also included in the COLR. The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

(continued)

BASES

ACTIONS

B.1 (continued)

must be 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 3.2.2.1

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 reaches ≥ 25% RTP is acceptable given the 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 may take credit for conservatism in the control rod 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 either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The scram time dependent MCPR limits are contained in the COLR. This determination of the actual scram speed distribution for SPC methodology and of the parameter τ for GE methodology must be performed within 72 hours after each set of control rod 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 control rod scram speed distribution expected during the fuel cycle.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.6 and SR 3.3.1.1.7 (continued)

indication. This is required prior to fully withdrawing SRMs since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (initiate a rod block) if adequate overlap is not maintained. The IRM/APRM and SRM/IRM overlap are acceptable if a ½ decade overlap exists.

As noted, SR 3.3.1.1.7 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channel(s) that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.8

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The ~~1000~~ effective full power hours (EFPH) Frequency is based on operating experience with LPRM sensitivity changes.

(continued)

2000

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.9 and SR 3.3.1.1.12

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.1.1.9 is based on the reliability analysis of Reference 10.

The 24 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.10, SR 3.3.1.1.11, and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop, including associated trip unit, and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

Note 1 of SR 3.3.1.1.11 and SR 3.3.1.1.13 states that neutron detectors are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the ~~1000~~ 2000 EFPH LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note to SR 3.3.1.1.11 and SR 3.3.1.1.13 is provided that requires the APRM and IRM SRs to be performed within 24 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2.

(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 > 0.1×10^6 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 28×10^3 lb/hr (approximately a mass velocity of 0.25×10^6 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 > 28×10^3 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, application 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

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APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

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, 5, and 6 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 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

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BASES

APPLICABLE SAFETY ANALYSES	2.1.1.3 Reactor Vessel Water Level (continued) 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. 7). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10. 2. ANF-524(P)(A), Revision 2, Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence (as specified in Technical Specification 5.6.5).

(continued)

BASES

REFERENCES (continued)

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. EMF-1125(P)(A), Supplement 1 Appendix C, ANFB Critical Power Correlation Application for Coresident Fuel, Siemens Power Corporation (as specified in Technical Specification 5.6.5).

7. 10 CFR 100.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in the 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 power/flow state is the greater of the rated conditions MCPR operating limit or the power 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.

Power dependent MCPR limits ($MCPR_p$) are determined on a cycle-specific basis. These limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.

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. MCPR operating limits which include the effects of analyzed equipment out-of-service are also included in the COLR. The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

(continued)

BASES

ACTIONS

B.1 (continued)

must be 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 3.2.2.1

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 reaches ≥ 25% RTP is acceptable given the inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analyses may take credit for conservatism in the control rod 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 either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The scram time dependent MCPR limits 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 time) analyses. This determination of the actual scram speed distribution for SPC methodology and of the

(continued)

BASES

LCO (continued)

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 (face or diagonal) locations.

Table 3.1.4-1 is modified by two Notes, which state 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 experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.2 (continued)

parameter τ for GE methodology must be performed within 72 hours after each set of control rod 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 control rod scram speed distribution expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (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) , Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995.

7. NFSR-0091, 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-Neutronic 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).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.6 and SR 3.3.1.1.7 (continued)

indication. This is required prior to fully withdrawing SRMs since indication is being transitioned from the SRMs to the IRMs.

The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (initiate a rod block) if adequate overlap is not maintained. The IRM/APRM and SRM/IRM overlap are acceptable if a ½ decade overlap exists.

As noted, SR 3.3.1.1.7 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channel(s) that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.8

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 2000 effective full power hours (EFPH) Frequency is based on operating experience with LPRM sensitivity changes.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.9 and SR 3.3.1.1.12

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.1.1.9 is based on the reliability analysis of Reference 10.

The 24 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.10, SR 3.3.1.1.11, and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop, including associated trip unit, and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

Note 1 of SR 3.3.1.1.11 and SR 3.3.1.1.13 states that neutron detectors are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 2000 EFPH LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note to SR 3.3.1.1.11 and SR 3.3.1.1.13 is provided that requires the APRM and IRM SRs to be performed within 24 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2.

(continued)

Attachment F
Proposed Changes to Technical Specifications for LaSalle County Station, Units 1 and 2
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 TS Section is noted on the top right corner of each CTS page, identifying the TS Section 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.