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10 CFR 50.90

March 18, 2020

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

SUBJECT: Limerick Units 1 and 2 License Amendment Request for Proposed Changes to Technical Specifications 3.10.8 “Inservice Leak and Hydrostatic Testing”

Pursuant to 10 CFR 50.90, “Application for amendment of license or construction permit,” Exelon Generation Company, LLC (Exelon), proposes changes to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively.

The proposed amendment would revise LGS Limiting Condition for Operation (LCO) 3.10.8, and the associated Bases, to expand its scope to include provisions for temperature excursions greater than 212°F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in Operational Condition 4. This change is consistent with NRC approved Revision 0 to Technical Specification Task Force (TSTF) Traveler, TSTF-484, "Use of TS 3.10.1 for Scram Time Testing Activities." The availability of TSTF-484, Revision 0, was announced in the Federal Register on October 27, 2006 (71 FR 63050), as part of the consolidated line item improvement process (CLIIP). Any deviations from the approved TSFT-484 due to LGS not being an Improved Standard Technical Specification nuclear power plant are explained in Attachment 1 of this document.

Evaluation of the proposed changes is provided in Attachment 1. Markups of the proposed TS changes are provided in Attachment 2. Markups of the proposed TS Bases changes are provided in Attachment 3 for information only.

Exelon has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92.

This license amendment request contains no regulatory commitments.

Exelon requests approval of the proposed amendment by March 18, 2021. Upon NRC approval, the amendment shall be implemented within 60 days of issuance.

The proposed changes have been reviewed and recommended for approval by the Plant Operations Review Committee in accordance with the Exelon Quality Assurance Program.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the Commonwealth of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Steve Flickinger at 267-533-1437.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of March 2020.

Respectfully,

D. L. Helker

David Helker
Sr. Manager, Licensing
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Proposed Technical Specifications Markup Pages
3. Proposed Technical Specification Bases Markup Pages

cc: USNRC Region I, Regional Administrator w/ attachments
USNRC Project Manager, LGS "
USNRC Senior Resident Inspector, LGS "
Director, Bureau of Radiation Protection - Pennsylvania Department of Environmental Protection "

ATTACHMENT 1

License Amendment Request

Limerick Generating Station, Units 1 and 2

Docket Nos. 50-352 and 50-353

EVALUATION OF PROPOSED CHANGES

Subject: Limerick Units 1 and 2 License Amendment Request for Proposed Changes to Technical Specifications 3.10.8 "Inservice Leak and Hydrostatic Testing"

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, “Application for amendment of license or construction permit,” Exelon Generation Company, LLC (Exelon), proposes changes to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively.

The proposed amendment would revise LGS Limiting Condition for Operation (LCO) 3.10.8, and the associated Bases, to expand its scope to include provisions for temperature excursions greater than 212°F as a consequence of inservice leak and hydrostatic testing, and as a consequence of scram time testing initiated in conjunction with an inservice leak or hydrostatic test, while considering operational conditions to be in Mode 4 [LGS Operational Condition 4]. This change is consistent with NRC approved Revision 0 to Technical Specification Task Force (TSTF) Traveler, TSTF-484, "Use of TS 3.10.1 for Scram Time Testing Activities." The availability of TSTF-484, Revision 0, was announced in the Federal Register on October 27, 2006 (71 FR 63050), as part of the consolidated line item improvement process (CLIIP).

Evaluation of the proposed changes, including any deviations from the approved TSFT-484 due to LGS not being an improved Standard Technical Specification (STS) nuclear power plant, are explained in this attachment. Markups of the proposed TS changes are provided in Attachment 2. Markups of the proposed TS Bases changes are provided in Attachment 3 for information only.

2.0 DETAILED DESCRIPTION

The current LGS TS 3.10.8 states the following:

When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to 212°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of an inservice leak or hydrostatic test provided the following OPERATIONAL CONDITION 3 Specifications are met:

a. 3.3.2 ISOLATION ACTUATION INSTRUMENTION, Functions 7.a, 7.c.1, 7.c.2 and 7.d of Table 3.3.2-1;
b. 3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY;
c. 3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY;
d. 3.6.5.2.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES;
e. 3.6.5.2.2 REFUELING AREA SECONDARY CONTAIMENT AUTOMATIC ISOLATION VALVES; and
f. 3.6.5.3 STANDBY GAS TREATMENT SYSTEM.

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature greater than 200°F and less than or equal to 212°F.

The following change is proposed to TS 3.10.8 as noted below:

When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to greater than 200°F, and operation considered not to be in OPERATIONAL CONDITION 3:

- For performance of an inservice leak or hydrostatic test,
- As a consequence of maintaining adequate pressure for an inservice leak of hydrostatic test, or
- As a consequence of maintaining adequate pressure for control rod scram time testing initiated in conjunction with an inservice leak or hydrostatic test,

provided the following OPERATIONAL CONDITION 3 Specifications are met:

a. 3.3.2 ISOLATION ACTUATION INSTRUMENTION, Functions 7.a, 7.c.1, 7.c.2 and 7.d of Table 3.3.2-1;
b. 3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY;
c. 3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY;
d. 3.6.5.2.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES;
e. 3.6.5.2.2 REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES; and
f. 3.6.5.3 STANDBY GAS TREATMENT SYSTEM.

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature greater than 200°F.

The proposed change eliminates the 212°F ceiling on average reactor coolant temperature during the performance of inservice leak and hydrostatic testing and allows for average reactor coolant temperature to exceed 200°F, consistent with TSTF-484 and its Technical Analysis. The proposed change includes three bulletized circumstances under which TS 3.10.8 is applicable, providing clear guidance on when OPERATIONAL CONDITION 4 Reactor Coolant System (RCS) temperature limits may exceed 200°F, which aligns more uniformly with TSTF-484.

3.0 TECHNICAL EVALUATION

3.1 Applicability of Published Safety Evaluation

Consistent with the NRC approved Revision 0 of TSTF-484, the proposed TS changes include a revised TS 3.10.8, "Inservice Leak and Hydrostatic Testing." Proposed revisions to the TS Bases are also included in this application for information only. Adoption of the TS Bases associated with TSTF-484, Revision 0 is an integral part of implementing this TS amendment. The changes to the affected TS Bases pages will be incorporated in accordance with the LGS TS Bases Control Program. This application is being made in accordance with the CLIIP.

The proposed change will extend the allowance to include operations where temperature exceeds 212°F as a consequence of maintaining adequate pressure for inservice leak and hydrostatic testing, when initial test conditions commenced below 200°F. As such, no new OPERATIONAL CONDITIONS beyond those currently allowed by LCO 3.10.8 are introduced.

The extended allowances would result from operations that commence at reduced temperatures but approach the normal OPERATIONAL CONDITION 4 limit of 200°F prior to completion of the inspections or testing. The flexibility will allow continued inspection and testing activities without imposing the potential for interruption to steady state test pressure while reactor coolant temperatures are reduced to maintain < 200°F conditions.

Additionally, the proposed changes will extend the allowance to include operations where temperature exceeds 212°F as a consequence of maintaining pressure for continued scram time testing that was initiated in conjunction with an inservice leak or hydrostatic test. Currently, if scram time testing is not completed during the normal inservice leak or hydrostatic test conditions, and temperatures approach 212°F, completion of scram time testing is suspended and resumed during reactor startup - typically between 25% and 40% rated thermal power (RTP). By extending the provisions of LCO 3.10.8 to scram time testing, more efficient test scheduling can be realized. Performance of scram time testing during the outage can represent a significant critical path reduction in returning to full power operations and can avoid the undesired extended operation in the 25% to 40% power range.

Exelon has reviewed the model safety evaluation dated October 27, 2006, as part of the Federal Register Notice of Availability. This review included a review of the NRC's evaluation, as well as the information provided in TSTF-484, Revision 0. Exelon has concluded that the justifications presented in the TSTF-484, Revision 0 proposal and the model safety evaluation prepared by the NRC are applicable to LGS, Units 1 and 2, and justify this amendment for incorporation of the changes to the plant TS.

3.2 Deviations from the Published TSTF-484

Deviations associated with the adoption of TSTF-484 are nomenclature and formatting of the TS. TSTF-484 references improved STS LCO 3.10.1, “Inservice Leak and Hydrostatic Testing Operation,” where this proposed change is to LGS LCO 3.10.8 for performing the same action. LGS is licensed to STS and not to Improved STS. Additionally, the formatting deviates from TSTF-484 as well. However, the three bulletized actions in TSTF-484 are being inserted into LGS LCO 3.10.8 to specify the actions and evolutions applicable to exceeding the 200°F Reactor Coolant System average temperature limit.

RIS-2000-06, “Consolidated Line Item Improvement Process For Adopting Standard Technical Specifications Changes For Power Reactors,” provides guidance to Licensees who are not licensed to Improved STSs and want to adopt TS changes under the CLIIP process. The RIS states:

> The CLIIP would allow efficient adoption of the TSTF changes by licensees that have converted to the STS, as well as by licensees that have not converted to the STS but have determined that the TSTF changes are applicable. This process would streamline the documentation process for both the NRC and the licenses.

Therefore, deviations associated with this proposal are not technical deviations and the analysis within TSTF-484 applies to LGS.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

A description of the proposed TS change and its relationship to applicable regulatory requirements was provided in the NRC Notice of Availability published on October 27, 2006 (71 FR 63050), which states:

The RCS serves as a pressure boundary and also serves to provide a flow path for the circulation of coolant past the fuel. In order to maintain RCS integrity, Section XI of the American Society of Mechanical Engineers (ASME) Pressure Vessel Code requires periodic hydrostatic and leakage testing. Hydrostatic tests are required to be performed once every ten years and leakage tests are required to be performed each refueling outage. Appendix G to 10 CFR Part 50 states that pressure tests and leak tests of the reactor vessel that are required by Section XI of the American Society of Mechanical Engineers (ASME) Pressure Vessel Code must be completed before the core is critical.

NUREG–1433, General Electric Plants, BWR/4, Revision 4, Standard Technical Specifications (STS) and NUREG–1434, General Electric Plants, BWR/6, Revision 4, STS both currently contain LCO 3.10.1, ‘‘Inservice Leak and Hydrostatic Testing Operation.’’ LCO 3.10.1 was created to allow for hydrostatic and leakage testing to be conducted while in Mode 4 [LGS OPERATIONAL CONDITION 4] with average reactor coolant temperature greater than [200] °F provided certain secondary containment LCOs are met.

TSTF–484, Revision 0, “Use of TS 3.10.1 for Scram Time Testing Activities,” modifies LCO 3.10.1 to allow a licensee to implement LCO 3.10.1, while hydrostatic and leakage testing is being conducted, should average reactor coolant temperature exceed [200] °F during testing. This modification does not alter current requirements for hydrostatic and leakage testing as required by Appendix G to 10 CFR Part 50.

Control rods function to control reactor power level and to provide adequate excess negative reactivity to shut down the reactor from any normal operating or accident condition at any time during core life. The control rods are scrammed by using hydraulic pressure exerted by the control rod drive (CRD) system. Criterion 10 of Appendix A to 10 CFR part 50 states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The scram reactivity used in design basis accidents (DBA) and transient analyses is based on an assumed control rod scram time.

NUREG–1433, General Electric Plants, BWR/4, Revision 4, STS and NUREG– 434, General Electric Plants, BWR/6, Revision 4, STS both currently contain surveillance requirements (SR) to conduct scram time testing when certain conditions are met in order to ensure that Criterion 10 of Appendix A to 10 CFR Part 50 is satisfied. SR 3.1.4.1 [LGS SR 4.1.3.2a] requires scram time testing to be conducted following a shutdown greater than 120 days while SR 3.1.4.4 [LGS SR 4.1.3.2b] requires scram time testing to be conducted following work on the CRD system or following fuel movement within the affected core cell. Both SRs must be performed at reactor steam dome pressure greater than or equal to 800 [LGS 950] psig and prior to exceeding 40 percent rated thermal power (RTP).

TSTF–484, Revision 0, “Use of TS 3.10.1 for Scram Time Testing Activities,” would modify LCO 3.10.1 [LGS LCO 3.10.8] to allow SR 3.1.4.1 [LGS SR 4.1.3.2a] and SR 3.1.4.4 [LGS SR 4.1.3.2b] to be conducted in Mode 4 [LGS OPERATIONAL CONDITION 4] with average reactor coolant temperature greater than [200] °F.’ This modification to LCO 3.10.1 [LGS LCO 3.10.8] does not alter the means of compliance with Criterion 10 of Appendix A to 10 CFR Part 50.

4.2 Precedent

The proposed changes to TS LCO 3.10.8 are similar to changes previously approved by the NRC for a number of plants for Improved STS LCO 3.10.1:

1. Letter from Marshall J. David (U.S. Nuclear Regulatory Commission) to Nine Mile Point Nuclear Station, LLC, "Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Technical Specification Change for Scram Time Testing Activities, using the Consolidated Line Item Improvement Process (TAC NO. MD6903)," dated February 7, 2008.

2. Letter from Alan B. Wang (U.S. Nuclear Regulatory Commission) to Entergy Operations, Inc., "River Bend Station, Unit 1 – Issuance of Amendment Re: Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-484, "Use of TS 3.10.1 for Scram Time Testing Activities" (TAC NO. ME4431)," dated January 5, 2011.

3. Letter from Michael Mahoney (U.S. Nuclear Regulatory Commission) to FirstEnergy Nuclear Operating Company, "Perry Nuclear Power Plant, Unit No. 1 – Issuance of Amendment Re: Revise Technical Specification 3.10.1 "Inservice Leak and Hydrostatic Testing Operation" (TAC NO. ME8048)," dated April 18, 2013.

4. Letter from Terry A. Beltz (U.S. Nuclear Regulatory Commission) to Northern States Power Company – Minnesota, "Monticello Nuclear Generating Plant – Issuance of Amendment No. 174 to Adopt Technical Specifications Task Force (TSTF) Traveler TSTF-484, Revision 0, "Use of TS 3.10.1 for Scram Time Testing Activities (TAC NO. MF0362," dated August 9, 2013.

5. Letter from Blake Purnell (U.S. Nuclear Regulatory Commission) to Exelon Generation Company, LLC, "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 – Issuance of Amendments to Add Technical Specification 3.10.8, Inservice Leak and Hydrostatic Testing Operations" (CAC NOS. MF5471-MF5476)," dated December 17, 2015

4.3 No Significant Hazards Consideration

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The probability of an accident is not significantly increased because the proposed

changes will not alter the method by which inservice leakage and hydrostatic testing is performed or significantly change the temperatures and pressures achieved to perform the test. The consequences of previously evaluated accidents are not significantly increased since the RPV hydro test pressurizes the reactor coolant system (RCS) by balancing liquid water input (CRD) against liquid water rejection (RWCU) in a near water solid condition and not a saturated condition. Because pressure is maintained due to a water solid flow balanced condition, a postulated loss of water inventory (due to a loss of coolant accident) would cause a rapid RCS pressure decrease, with water level far above the top of active fuel. The required operable low pressure emergency injection systems in conjunction with an operable Secondary Containment, Standby Gas Treatment System, and their associated initiation instrumentation will be sufficient to mitigate the consequences of a postulated accident during the RPV hydro test.

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

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Technical Specifications currently allow for operation at > 200°F while imposing OPERATIONAL CONDITION 4 requirements in addition to the secondary containment requirements required to be met. No new operational conditions beyond those currently allowed by LCO 3.10.1 [LGS 3.10.8] are introduced. The extended allowances would result from operations that commence at reduced temperatures but approach the normal OPERATIONAL CONDITION 4 limit of 200°F prior to completion of the inspections or testing. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

Technical Specifications currently allow for operation at > 200°F while imposing OPERATIONAL CONDITION 4 requirements in addition to the secondary containment requirements required to be met. Extending the activities that can apply this allowance will not adversely impact any margin of safety. Allowing completion of inspections and testing and supporting completion of scram time testing initiated in conjunction with an inservice leak or hydrostatic test prior to power operation, results in enhanced safe operations by eliminating unnecessary maneuvers to control reactor temperature and pressure.

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

4.4 Conclusions

Based on the above, Exelon concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed change would not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and would not change an inspection or surveillance requirement. Furthermore, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

1. Federal Register Notice, Notice of Availability published on October 27, 2006 (71 FR 63050).

2. Federal Register Notice, Notice for Comment published on August 21, 2006 (71 FR 48561).

3. TSTF-484, Revision 0, "Use of TS 3.10.1 for Scram Time Testing Activities," dated October 27, 2006.

4. RIS-2000-06, “Consolidated Line Item Improvement Process For Adopting Standard Technical Specifications Changes For Power Reactors.”

ATTACHMENT 2

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Proposed Technical Specifications Markup Pages

Unit 1 TS Pages

3/4 10-8

Unit 2 TS Pages

3/4 10-9

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

LIMITING CONDITIONS FOR OPERATION

Insert

3.10.8 ~~When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to 212°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of an inservice leak or hydrostatic test provided the following OPERATIONAL CONDITION 3 Specifications are met:~~

a. 3.3.2 ISOLATION ACTUATION INSTRUMENTATION, Functions 7.a, 7.c.1, 7.c.2 and 7.d of Table 3.3.2-1;
b. 3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY;
c. 3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY;
d. 3.6.5.2.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES;
e. 3.6.5.2.2 REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES; and
f. 3.6.5.3 STANDBY GAS TREATMENT SYSTEM.

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature greater than 200°F ~~and less than or equal to 212°F~~.

ACTION:

With the requirements of the above Specifications not satisfied:

1. Immediately enter the applicable (OPERATIONAL CONDITION 3) action for the affected Specification; or

2. Immediately suspend activities that could increase the average reactor coolant temperature or pressure and reduce the average reactor coolant temperature to 200°F or less within 24 hours.

SURVEILLANCE REQUIREMENTS

4.10.8 Verify applicable OPERATIONAL CONDITION 3 surveillances for the Specifications listed in 3.10.8 are met.

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

LIMITING CONDITION FOR OPERATION

Insert

3.10.8 ~~When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to 212°F, and operation considered not to be in OPERATIONAL CONDITION 3, to allow performance of an inservice leak or hydrostatic test provided the following OPERATIONAL CONDITION 3 Specifications are met:~~

a. 3.3.2 ISOLATION ACTUATION INSTRUMENTATION, Functions 7.a, 7.c.1, 7.c.2 and 7.d of Table 3.3.2-1;
b. 3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY;
c. 3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY;
d. 3.6.5.2.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES;
e. 3.6.5.2.2 REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES; and
f. 3.6.5.3 STANDBY GAS TREATMENT SYSTEM.

APPLICABILITY: OPERATIONAL CONDITION 4, with average reactor coolant temperature greater than 200°F ~~and less than or equal to 212°F~~.

ACTION:

With the requirements of the above Specifications not satisfied:

1. Immediately enter the applicable (OPERATIONAL CONDITION 3) action for the affected Specification; or

2. Immediately suspend activities that could increase the average reactor coolant temperature or pressure and reduce the average reactor coolant temperature to 200°F or less within 24 hours.

SURVEILLANCE REQUIREMENTS

4.10.8 Verify applicable OPERATIONAL CONDITION 3 surveillances for the Specifications listed in 3.10.8 are met.

TS 3.10.8 Insert:

When conducting inservice leak or hydrostatic testing, the average reactor coolant temperature specified in Table 1.2 for OPERATIONAL CONDITION 4 may be increased to greater than 200°F, and operation considered not to be in OPERATIONAL CONDITION 3:

- For performance of an inservice leak or hydrostatic test,
- As a consequence of maintaining adequate pressure for an inservice leak of hydrostatic test, or
- As a consequence of maintaining adequate pressure for control rod scram time testing initiated in conjunction with an inservice leak or hydrostatic test,

provided the following OPERATIONAL CONDITION 3 Specifications are met:

ATTACHMENT 3

License Amendment Request

Limerick Generating Station, Units 1 and 2
Docket Nos. 50-352 and 50-353

Proposed Technical Specifications Bases Markup Pages

Unit 1 TS Bases Pages

B 3/4 10-2

Unit 2 TS Bases Pages

B 3/4 10-2

BASES

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

This special test exception permits certain reactor coolant pressure tests to be performed in OPERATIONAL CONDITION 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) or plant temperature control capabilities during these tests require the pressure testing at temperatures greater than 200°F ~~and less than or equal to 212°F~~ (normally corresponding to OPERATIONAL CONDITION 3). The additionally imposed OPERATIONAL CONDITION 3 requirements for SECONDARY CONTAINMENT INTEGRITY provide conservatism in response to an operational event.

Invoking the requirement for Refueling Area Secondary Containment Integrity along with the requirement for Reactor Enclosure Secondary Containment Integrity applies the requirements for Reactor Enclosure Secondary Containment Integrity to an extended area encompassing Zones 1 and 3. Core alterations and fuel handling are prohibited in this secondary containment configuration. Drawdown and inleakage testing performed for the combined zone system alignment shall be considered adequate to demonstrate integrity of the combined zones.

Inservice hydrostatic testing and inservice leak pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to the reactor going critical after a refueling outage. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.6, Reactor Coolant System Pressure/Temperature Limits. These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased reactor fluence over time, the minimum allowable vessel temperature increases at a given pressure.

BASES

3/4.10.8 INSERVICE LEAK AND HYDROSTATIC TESTING

This special test exception permits certain reactor coolant pressure tests to be performed in OPERATIONAL CONDITION 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) or plant temperature control capabilities during these tests require the pressure testing at temperatures greater than 200°F ~~and less than or equal to 212°F~~ (normally corresponding to OPERATIONAL CONDITION 3). The additionally imposed OPERATIONAL CONDITION 3 requirements for SECONDARY CONTAINMENT INTEGRITY provide conservatism in response to an operational event.

Invoking the requirement for Refueling Area Secondary Containment Integrity along with the requirement for Reactor Enclosure Secondary Containment Integrity applies the requirements for Reactor Enclosure Secondary Containment Integrity to an extended area encompassing Zones 2 and 3. Core alterations and fuel handling are prohibited in this secondary containment configuration. Drawdown and inleakage testing performed for the combined zone system alignment shall be considered adequate to demonstrate integrity of the combined zones.

Inservice hydrostatic testing and inservice leak pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to the reactor going critical after a refueling outage. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.6, Reactor Coolant System Pressure/Temperature Limits. These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased reactor fluence over time, the minimum allowable vessel temperature increases at a given pressure.