Station Support Department

10 CFR 50.90

PECO ENERGY

PECO Energy Company
Nuclear Group Headquarters
965 Chesterbrook Boulevard
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August 22, 1994

Docket Nos. 50-352
50-353
License Nos. NPF-39
NPF-85

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

Subject: Limerick Generating Station, Units 1 and 2
Technical Specifications Change Request Nos.
94-33-0, 94-34-0, 94-37-0, 94-39-0, and 94-43-0

Gentlemen:

PECO Energy Company is submitting Technical Specifications (TS) Change Request Nos. 94-33-0, 94-34-0, 94-37-0, 94-39-0, and 94-43-0 in accordance with 10 CFR 50.90, requesting amendments to the TS (i.e., Appendix A) of Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station, Units 1 and 2, respectively. The proposed TS Changes, which are consistent with the Improved Standard Technical Specifications (NUREG-1433), involve the following:

94-33-0	"Control Rod Block Instrumentation"
94-34-0	"Standby Liquid Control System Operability in Mode 5"
94-37-0	"Scram Discharge Volume Valve Testing"
94-39-0	"Optional Method of Scram Timing"
94-43-0	"Definition of Core Alteration"

Information supporting these TS Change Requests is contained in Attachment 1 to this letter, and the proposed replacement pages for the LGS, Units 1 and 2, TS are contained in Attachment 2. This information is being submitted under affirmation, and the required affidavit is enclosed.

We request that, if approved, the amendments to the LGS, Units 1 and 2, TS be issued prior to January 28, 1995 and become effective immediately upon issuance.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

G. A. Helker / FOR

G. A. Hunger, Jr., Director
Licensing Section

Attachments

Enclosure

cc: T. T. Martin, Administrator, Region I, USNRC (w/ attachments, enclosure)
N. S. Perry, USNRC Senior Resident Inspector, LGS (w/ attachments, enclosure)
R. R. Janati, Director, PA Bureau of Radiation Protection, (w/ attachments, enclosure)

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COMMONWEALTH OF PENNSYLVANIA :

:

COUNTY OF CHESTER :

W. H. Smith, III, being first duly sworn, deposes and says:

That he is Vice President of PECO Energy Company; the Applicant herein; that he has read the foregoing Technical Specifications Change Request Nos. 94-33-0, 94-34-0, 94-37-0, 94-39-0, and 94-43-0 for Limerick Generating Station, Units 1 and 2, involving Control Rod Block Instrumentation, Standby Liquid Control System Operability in Mode 5, Scram Discharge Volume Valve Testing, Optional Method of Scram Timing, and Definition of Core Alteration, respectively, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

William H Smith III

Vice President

Subscribed and sworn to

before me this 23rd day

of August 1994.

Erica A. Santori

Notary Public

Notarial Seal
Erica A. Santori, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires July 10, 1995

ATTACHMENT 1

LIMERICK GENERATING STATION

UNITS 1 AND 2

Docket Nos.	50-352 50-353
License Nos.	NPF-39 NPF-85

Section 1: Technical Specifications Change Request No. 94-33-0
"Control Rod Block Instrumentation"

Section 2: Technical Specifications Change Request No. 94-34-0
"Standby Liquid Control System Operability in Mode 5"

Section 3: Technical Specifications Change Request No. 94-37-0
"Scram Discharge Volume Valve Testing"

Section 4: Technical Specifications Change Request No. 94-39-0
"Optional Method of Scram Timing"

Section 5: Technical Specifications Change Request No. 94-43-0
"Definition of Core Alteration"

Supporting Information for Changes - 21 pages

SECTION 1: "CONTROL ROD BLOCK INSTRUMENTATION" (TSCR 94-33-0)

PECO Energy Company, Licensee under Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station, Units 1 and 2, respectively, requests that the Technical Specifications (TS) contained in Appendix A to the Operating Licenses be amended as proposed herein, to revise TS Surveillance Requirements (SR) 4.3.7.6 and 4.9.2 and Table 4.3.6-1 involving Control Rod Block and Source Range Monitor instrumentation. The proposed changes to the TS are indicated by a vertical bar in the margin of TS pages 3/4 3-61, 3/4 3-62, 3/4 3-88, and 3/4 9-4 for Units 1 and 2. The TS pages showing the proposed changes are contained in Attachment 2.

We request that, if approved, the TS changes proposed herein be issued by January 28, 1995 and become effective immediately upon issuance of the amendments.

This TS Change Request provides a discussion and description of the proposed TS changes, a safety assessment of the proposed TS changes, information supporting a finding of No Significant Hazards Consideration, and Information Supporting an Environmental Assessment.

Discussion and Description of the Proposed Changes

This section involves proposed changes to the Limerick Generating Station, Units 1 and 2, Technical Specifications in the areas of Control Rod Block and Source Range Monitor Instrumentation. It is the intent of the proposed changes to eliminate unnecessary testing by aligning Limerick Generating Station Technical Specifications surveillance test frequencies with their counterparts in the Improved Standard Technical Specifications (ISTS), NUREG-1433, issued September 28, 1992. This will provide the station with the flexibility to maintain the instrumentation in surveillance at frequencies already deemed acceptable, while minimizing delays in plant startup due to requirements that are more demanding than those imposed during normal operation. The proposed changes also provide clarity in describing allowable surveillance testing exemptions, based on plant conditions and operating modes. These changes are also consistent with the requirements of ISTS, NUREG-1433.

A listing of the Technical Specifications changes being proposed is as follows:

1. Rod Block Monitor (RBM): Delete the requirement to perform the Functional Test specifically for (prior to) startup.

2. Average Power Range Monitor (APRM): Delete the requirement to perform the Functional Test specifically for (prior to) startup.

3. Source Range Monitor (SRM):

   a. Delete the requirement to perform the Functional Test specifically for (prior to) startup.

   b. Change the required frequency of the Functional Test from Weekly (W) to Monthly (M) when in Operating Condition Two (2).

   c. Add a note of explanation to the Functional Test frequency granting an exemption from the provisions of Technical Specifications section 4.0.4 provided testing is performed within twelve (12) hours after the Intermediate Range Monitors are on range two (2) or below during a shutdown.

   d. Change the frequency required for the performance of the Calibration Test from Semi-annual (SA) to each Refueling (R).

4. Intermediate Range Monitor (IRM):

   a. Delete the requirement to perform the Functional Test specifically for (prior to) startup.

   b. Change the frequency required for the performance of the Calibration Test from Semi-annual (SA) to each Refueling (R).

5. Reactor Coolant System Recirculation Flow: Delete the requirement to perform the Functional Test specifically for (prior to) startup.

6. Reactor Mode Switch Shutdown Position: Add a note of explanation to the Functional Test frequency granting an exemption from the provisions of Technical Specifications section 4.0.4 provided testing is performed within one (1) hour after the mode switch is placed in the shutdown position.

Safety Assessment

All of the proposed changes have been made in accordance with the philosophies and requirements of the Improved Standard Technical Specifications (ISTS), NUREG-1433, issued September 28, 1992, and do not alter equipment configuration or operation in any way.

The frequencies specified in the ISTS surveillance requirements are determined from a reliability analysis based on the same document that was used for the existing Technical Specifications. An increase in the specified intervals does not increase the chance of a malfunction between test performances. Additionally, the deletion of the special pre-startup ("S/U") performance requirements does nothing more than leave the equipment with the same surveillance operability considerations for startup as it normally is subjected to during prolonged operation in the required Operating Condition. There is no need to impose extra, more restrictive, surveillance testing on the instrumentation prior to a startup. If work is performed on any of these instruments during shutdown, our procedures require that the affected instruments be tested to assure their operability before reliance on them. Further, Limerick equipment reliability compares favorably with other plants (greater than 99% reliability based on review of surveillance tests).

Because the proposed changes do not alter equipment configuration or operation, and because equipment reliability under the proposed surveillance frequencies is unaffected, the instrumentation will continue to operate in the same manner as that currently experienced.

Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed changes to the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) involving Control Rod Block and Source Range Monitor Instrumentation do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. <u>The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

The proposed TS changes can be divided into two general categories, the deletion of the "S/U" requirements, and the change in frequency of the SRM and IRM Calibration and Functional Tests. In each case in which the "S/U" requirement has been deleted, the normal surveillance frequency specified for the required Operating Condition remains. The equipment's associated probability of failure remains unchanged. In the case of the surveillance frequency changes proposed for the SRMs and IRMs, the probability of an accident evaluated in the SAR occurring does not increase since there is no credit taken in the SAR for those Control Rod Block functions with respect to an accident. As such, the proposed changes will not result in an increase in the probability of occurrence of an accident previously evaluated in the SAR. The proposed TS changes do not alter the method of operation or performance of the equipment in carrying out associated Control Rod Block functions. Thus, the consequences of an accident previously evaluated in the SAR are not increased.

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

2. <u>The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

The proposed TS changes do not alter the configuration of the plant or the way that the plant is operated. The equipment can perform no other function than it is presently capable of, or cause or permit any other accident than is now possible. Thus, the possibility of an accident of a different type than previously evaluated in the SAR cannot be created.

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

3. <u>The proposed TS changes do not involve a significant reduction in a margin of safety.</u>

Since the proposed TS changes affect only the surveillance frequency intervals and do not change the plant configuration or associated instrument setpoints, there is no quantitative or qualitative reduction in the margin of safety. Thus, the margin of safety as defined in the bases of any Technical Specification is not reduced.

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

Information Supporting an Environmental Assessment

An Environmental Assessment is not required for the changes proposed by this TS Change Request because the requested changes to the LGS, Units 1 and 2, TS conform to the criteria for "actions eligible for categorical exclusion," as specified in 10 CFR 51.22(c)(9). The requested changes will have no impact on the environment. The proposed changes do not involve a Significant Hazards Consideration as discussed in the preceding section. The proposed changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed these proposed changes to the LGS, Units 1 and 2, TS and have concluded that they do not involve an unreviewed safety question, and will not endanger the health and safety of the public.

SECTION 2: "STANDBY LIQUID CONTROL SYSTEM OPERABILITY IN MODE 5" (TSCR 94-34-0)

PECO Energy Company, Licensee under Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station, Units 1 and 2, respectively, requests that the Technical Specifications (TS) contained in Appendix A to the Operating Licenses be amended as proposed herein, to revise TS Section 3/4.1.5, "Standby Liquid Control System," (SLCS), to remove the operability requirement for the SLCS in Operational Condition (OPCON) 5, Refueling, with any control rod withdrawn. The proposed change to the TS is indicated by a vertical bar in the margin of TS page 3/4 1-19 for Units 1 and 2. The TS pages showing the proposed change are contained in Attachment 2.

We request that, if approved, the TS change proposed herein be issued by January 28, 1995 and become effective immediately upon issuance of the amendments.

This TS Change Request provides a discussion and description of the proposed TS change, a safety assessment of the proposed TS change, information supporting a finding of No Significant Hazards Consideration, and Information Supporting an Environmental Assessment.

Discussion and Description of the Proposed Change

The proposed Technical Specifications (TS) Change Request will remove the operability requirement for the Standby Liquid Control System (SLCS) in OPERATIONAL CONDITION (OPCON) 5, Refueling, with any control rod withdrawn. The purpose of the SLCS, which is a reactivity control system, is to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive, xenon-free state without taking credit for control rod movement. This is accomplished by injecting sodium pentaborate into the reactor core. The amount of boron contained in the SLCS is designed to achieve shutdown assuming the water inventory in the reactor vessel is at normal power operating levels. During refueling operations, the reactor vessel head is removed and the refueling cavity is flooded. With the refueling cavity flooded, the amount of boron in the SLCS may not be sufficient to halt an inadvertent criticality event.

According to General Design Criterion (GDC) 26, two independent reactivity control systems of different design principles shall be provided. However, according to the GDC, only one of the two reactivity control systems must be capable of holding the reactor subcritical under cold conditions. Cold conditions are defined as OPCON 4, Cold shutdown or OPCON 5, Refueling. The Control Rod System is capable of holding the reactor core subcritical under cold conditions regardless of the water level in the reactor. The SLCS provides additional backup capability for reactivity control (i.e., achieving cold shutdown), independent of the normal reactivity control system. Thus it satisfies the requirements of GDC 26.

10CFR50.62(c)(4) requires that each Boiling Water Reactor have a SLCS to reduce the consequences of Anticipated Transients Without Scram (ATWS) events, which can only occur during power operations. The SLCS at LGS Units 1 and 2 satisfies this requirement.

In OPCON 5 the reactor is already shutdown with control rods fully inserted in any cell that has fuel in it. During core shuffling, subcriticality is determined for each step with sufficient levels of conservatism. This analytical SDM, in conjunction with TS requirements and procedural controls, assures that an inadvertent criticality will not occur. During core offloading, the amount of reactivity present in the core will be constantly reduced, in accordance with TS and procedural controls. This means that the Shutdown Margin (SDM) of the core is the same or greater than its initial value during the entire core offload process.

SDM is analytically determined prior to the fuel being loaded into the vessel. The calculated SDM is the acceptance criteria used in TS Surveillance Requirement 4.1.1. This analytical SDM, in conjunction with TS requirements and procedural controls, assures that an inadvertent criticality will not occur during core alterations.

If a control rod is withdrawn in OPCON 5 and SDM has not been demonstrated (i.e., during reload), additional restrictions are placed on the plant by TS Sections 3/4.9.2 and 3/4.10.3. In the extremely unlikely event that an inadvertent criticality occurs during this time, these additional restrictions assure the Control Rod System will be automatically actuated by the Reactor Protection System (RPS). Both the Control Rod System and the RPS are highly reliable systems.

Based on the above discussion it is concluded that the SLC System is not required to be OPERABLE during OPCON 5, provided that SDM, either demonstrated or analytically determined, is maintained and all required TS actions and procedural controls are followed.

Therefore, we propose that TS Section 3/4.1.5 (i.e., page 3/4 1-19) be revised to reflect the removal of the operability requirement for the SLCS in OPCON 5. The Unit 1 TS page reflects the ARTS/MELLLA changes which have been already implemented at LGS Unit 1 during 1R05. These changes are scheduled to be implemented at LGS Unit 2 during 2R03. Our Unit 2 TS amendment request to address these changes has not been issued by the NRC yet; therefore, the Unit 2 TS page looks different. Operation with ARTS/MELLLA does not apply to OPCON 5; therefore, the proposed TS change applies to both LGS Units 1 and 2.

This proposed TS change is consistent with the requirements of the Improved Standard Technical Specifications, NUREG-1433, issued September 28, 1992, and has been approved by the NRC for implementation at other plants (i.e., Nine Mile Point Unit 2, Brunswick Units 1 and 2).

Safety Assessment

The purpose of the SLCS is to provide the capability of shutting down the reactor from a full power condition, and maintaining it subcritical until the cold shutdown condition is achieved without control rod movement. The SLCS injects sodium pentaborate solution into the reactor core upon initiation. In OPERATIONAL CONDITION (OPCON) 5, the reactor is already shut down with control rods fully inserted in any core cells that have fuel assemblies in them.

The one-rod-out interlock associated with the Refuel position of the reactor mode switch provides protection against inadvertent criticality while the reactor is in OPCON 5. Specifically, the reactor mode switch will be in the Refuel position (and locked) and this initiates the Refuel position one-rod-out interlock which prevents the selection of a second control rod for movement when any other control rod is not fully inserted. The core is designed such that adequate shutdown margin (SDM) is maintained with one control rod fully withdrawn.

During core shuffling, subcriticality is determined for each step with sufficient levels of conservatism. This analytical SDM, in conjunction with TS requirements and procedural controls, assures that an inadvertent criticality will not occur. Additional protection against inadvertent criticality is also achieved in OPCON 5 because in accordance with TS and procedural controls, the amount of reactivity present in the core will be constantly reduced during core offloading. This means that the SDM of the core is the same or greater than its initial value during the entire core offloading process. SDM is analytically determined prior to fuel being reloaded into the reactor vessel. The calculated SDM is the acceptance criterion used in TS Surveillance Requirement 4.1.1. If a control rod is withdrawn in OPCON 5 and SDM has not been demonstrated (i.e., during reload), additional restrictions are placed on the plant by TS

Sections 3/4.9.2 and 3/4.10.3. In the extremely unlikely event that an inadvertent criticality occurs during this time, these additional restrictions assure that the control rod system will be automatically actuated by the RPS.

Therefore, the SLCS should not be required to be operable in OPCON 5 when any control rod is withdrawn since adequate SDM in conjunction with TS requirements for operability of the Refuel position one-rod-out interlock will assure that an inadvertent criticality event will not occur during refueling operations.

Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed change to the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) which removes the operability requirement for the Standby Liquid Control System (SLCS) in Operational Condition (OPCON) 5, Refueling, with any control rod withdrawn does not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed Technical Specifications (TS) change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed TS change will remove the SLCS operability requirement in OPCON 5. The purpose of the SLCS is to bring the reactor to and maintain it in a cold shutdown condition from normal power operations following failure to scram during power operations. Initiation of the SLCS is not a precursor to any accident. Therefore, inoperability of the SLCS in OPCON 5 cannot increase the probability of an accident previously evaluated.

   The proposed TS change does not involve a physical change in any system's configuration and no new modes of operation are introduced. The SLCS has no analyzed function in OPCON 5. The probability of fuel failure will not be increased by this change. Shutdown margin, in conjunction with TS requirements and procedural controls, will assure that an inadvertent criticality event will not occur during refueling. In addition, the Reactor Protection System (RPS) and Control Rod System will provide protection in the unlikely event that an inadvertent criticality should occur.

   Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed TS change does not involve a physical change in any system's configuration and no new modes of operation are introduced. The SLCS's only purpose is to mitigate the consequences of a failure to scram during power operation. In OPCON 5, the SLCS has no analyzed function; therefore, the proposed TS change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed TS change does not involve a significant reduction in a margin of safety.

The purpose of the SLCS is to bring the reactor to and maintain it in a cold shutdown condition from normal power operations following a failure to scram during power operations. The SLCS is not designed to terminate an inadvertent criticality during OPCON 5. Shutdown margin, either demonstrated or analytically determined, in conjunction with Technical Specifications and procedural controls, will assure that an inadvertent criticality event will not occur during refueling operations. In addition, the RPS and Control Rod System, which are extremely reliable, will provide protection in the unlikely event that an inadvertent criticality does occur. Therefore, the proposed TS change does not involve a reduction in a margin of safety.

Information Supporting an Environmental Assessment

An Environmental Assessment is not required for the change proposed by this TS Change Request because the requested change to the LGS, Units 1 and 2, TS conforms to the criteria for "actions eligible for categorical exclusion," as specified in 10 CFR 51.22(c)(9). The requested change will have no impact on the environment. The proposed change does not involve a Significant Hazards Consideration as discussed in the preceding section. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed the proposed change to the LGS, Units 1 and 2, TS and have concluded that it does not involve an unreviewed safety question, and will not endanger the health and safety of the public.

SECTION 3: "SCRAM DISCHARGE VOLUME VALVE TESTING" (TSCR 94-37-0)

PECO Energy Company, Licensee under Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively, requests that the Technical Specifications (TS) contained in Appendix ', to the Operating Licenses be amended as proposed herein to revise TS Surveillance Requirement (SR) 4.1.3.1.4 to delete the requirement that the Scram Discharge Volume (SDV) be determined operable by testing the SDV vent and drain valves from a configuration of less than or equal to 50% rod density. The proposed change to TS SR is indicated by a vertical bar in the margin of TS pages 3/4 1-5 for Units 1 and 2. The TS pages showing the proposed change are contained in Attachment 2.

We request that, if approved, the TS change proposed herein be issued by January 28, 1995, and become effective immediately upon issuance of the amendments.

This TS Change Request provides a discussion and description of the proposed TS changes, a safety assessment of the proposed TS changes, information supporting a finding of No Significant Hazards Consideration, and information supporting an Environmental Assessment.

Discussion and Description of the Proposed Changes

Currently, Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) Surveillance Requirement (SR) 4.1.3.1.4a requires that the Scram Discnarge Volume (SDV) be determined OPERABLE by testing the SDV vent and drain valves when the control rods are scram tested from a normal control rod configuration of less than or equal to 50% rod density at least once per 24 months. Operability of the SDV is determined by the SDV vent and drain valves closing within 30 seconds after a signal for the control rods to scram, and by re-opening when the scram signal is reset.

This proposed TS change involves revising TS SR 4.1.3.1.4a to delete the requirement for testing SDV vent and drain valves from a configuration of less than or equal to 50% rod density. The operability of these valves can be satisfactorily demonstrated during scram testing from shutdown conditions. The proposed TS change would eliminate the potential need for an additional startup and shutdown cycle solely to comply with the existing TS requirement, in the event a unit were to trip off-line shortly before an outage when the surveillance was scheduled to be performed. The proposed TS change will permit the surveillance to be performed during shutdown conditions without re-entering Power Operation (i.e., OPCON 1) or Startup (i.e., OPCON 2), and then scramming the reactor simply to achieve a control rod density of less than or equal to 50% rod density. Furthermore, this proposed TS change will reduce the potential for unnecessary challenges to plant systems and equipment.

In addition, this proposed TS change is consistent with criteria delineated in the Improved Standard TS (i.e., NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," dated September 28, 1992). Similar changes have been requested, some of which have been approved by the NRC for implementation, for other nuclear power plants (e.g., Susquehanna, Hope Creek, Browns Ferry, Perry, LaSalle, and River Bend).

Safety Assessment

The Scram Discharge Volume (SDV), which is part of the Control Rod Drive (CRD) system, consists of header piping which connects to each Hydraulic Control Unit (HCU) and drains into an instrument volume. The primary safety of the SDV vent and drain valves is to prevent an uncontrolled release of reactor coolant following a scram.

The design of the SDV minimizes the potential for a common mode failure of the scram function.

During normal plant operation, the SDV is empty and vented to the atmosphere through its open vent and drain valves. When a scram occurs, upon a signal from the Reactor Protection System (RPS), these vent and drain valves close to prevent an uncontrolled release of reactor coolant following a scram. The position of the SDV vent and drain valves is continuously monitored from the Main Control Room (MCR). Redundant scram discharge system vent and drain valves are provided to ensure that no single failure can result in an uncontrolled loss of reactor coolant. Redundant solenoid-operated pilot valves control the vent and drain valves. In addition, the solenoid-operated pilot valves are fail-safe (i.e., the SDV isolates) on loss of electric or pneumatic power. The SDV drain discharges to the equipment drain collection tank, and the vent line discharges to dirty radwaste. The vent lines are protected by vacuum breakers. The redundant isolation valves on the vent and drain lines are normally open during power operation and would not prevent draining of those lines in the nonscrammed condition.

Currently, Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) Surveillance Requirement (SR) 4.1.3.1.4a requires that the SDV be determined OPERABLE by testing the SDV vent and drain valves when the control rods are scram tested from a normal control rod configuration of less than or equal to 50% rod density at least once per 24 months. Operability is determined by the valves closing within 30 seconds after receipt of a signal for the control rods to scram, and by re-opening when the scram signal is reset. The primary safety function of the SDV vent and drain valves is to prevent an uncontrolled release of reactor coolant following a scram. The surveillance requirement to test the SDV vent and drain valve stroke times provides an integrated test of the SDV. This test verifies that the valves close within 30 seconds to limit the amount of leakage from the reactor. The 30 seconds requirement provides assurance that the leakage remains less than that assumed in the bounding analysis for coolant leakage in 10CFR100. The re-opening of the valves after the scram signal is reset verifies the ability of the SDV to drain to accept another scram.

This proposed TS change involves deleting the requirement of testing the SDV vent and drain valves from a configuration of less than or equal to 50% rod density. The operability of the valves can be satisfactorily demonstrated during a scram from shutdown conditions. Reactor pressure and CRD discharge flow conditions do not influence the SDV vent and drain valve closure rates since the SDV is of sufficient volume and initially vented such that peak pressure prior to complete isolation of the SDV will not be substantial. Performing the test, as currently required (i.e., from power operation), will result in higher backpressure conditions following the scram, as opposed to performing the test from shutdown conditions. However, the ability of the valves to re-open against rated reactor pressure is demonstrated after each reactor scram during power operation. The lower coolant temperatures expected during testing at shutdown conditions will also have a negligible impact on the performance of the surveillance. Since the initial conditions of pressure, temperature, and CRD discharge flowrate will not have an appreciable effect on the vent and drain valve performance, conducting this surveillance during shutdown conditions, as proposed, will not affect the validity of the surveillance results. The proposed change to the TS will continue to ensure the safety functions of the SDV vent and drain valves are maintained, and therefore, does not constitute a reduction in safety.

This proposed change to the TS would eliminate the potential need for an additional startup and shutdown cycle solely to comply with this TS requirement, in the event a unit were to trip off-line shortly before an outage when the surveillance was scheduled to be performed. The proposed TS change will allow the surveillance to be performed during shutdown conditions without re-entering Power Operation (i.e., OPCON 1) or Startup (i.e., OPCON 2), and then scramming the unit solely to achieve a control rod density of less than or equal to 50%. Furthermore, every reactor scram is a serious plant transient and a potential challenge to safety-related systems and equipment. Therefore, the potential decrease in future scrams which could result from this proposed TS change will represent an increase in safety.

This proposed TS change is consistent with the criteria specified in the Improved Standard Technical Specifications (i.e., NUREG-1433, dated September 28, 1992), and similar changes have been approved by the NRC for implementation at other power plants (e.g., Susquehanna, Hope Creek, Browns Ferry, Perry, LaSalle, and River Bend).

Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed change to Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) Surveillance Requirement (SR) 4.1.3.1.4a to delete the requirement of determining the OPERABILITY of the Scram Discharge Volume (SDV) by testing SDV vent and drain valves from a configuration of less than or equal to 50% rod density, does not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed Technical Specifications (TS) change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The Scram Discharge Volume (SDV) is not an accident initiator. Deletion of the requirement that the SDV be determined OPERABLE by testing the SDV vent and drain valves when control rods are scram tested from a normal control rod configuration of less than or equal to 50% rod density at least once per 24 months, as proposed, will have no effect on the probability or consequences of an accident previously evaluated.

   This proposed TS will have a negligible impact on the conditions experienced by the vent and drain valves as they stroke closed, since the SDV is initially vented to the atmosphere, and the valves close before the SDV becomes pressurized, even during a scram at full reactor power. Reactor pressure and Control Rod Drive (CRD) discharge flow conditions do not influence the SDV vent and drain closure rates, since the SDV is of sufficient volume and initially vented such that peak pressure prior to the SDV complete isolation will not be substantial. In addition, lower coolant temperatures expected during testing at shutdown conditions will also have a negligible impact on the performance of the test. Although, there could be some variation in the performance the SDV vent and drain valves to re-open when performing the test during shutdown conditions, as opposed to conducting the test during power operation, the ability of the valves to re-open is demonstrated after each reactor scram during power operation.

   In the event a SDV vent or drain valve failed to open, increasing SDV level during reactor operation would cause 1) an alarm in the Main Control Room (MCR), 2) a control rod block, and finally a reactor scram initiated by the Reactor Protection System (RPS) if action is not taken to drain the SDV. Therefore, the ability to shut down the reactor is not impaired. If a SDV vent or drain valve fails to close, the redundant valve's closure would provide the required function. If both valves failed to close, a loss of reactor coolant in the form of water discharged from the CRD system would occur. The amount of water discharged will be relatively small, and is more of a concern from the standpoint of contamination in the Secondary Containment rather than a loss of reactor water inventory. A structural failure of the SDV, which bounds this case of an open SDV vent or drain line, has been previously evaluated in NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping." In this evaluation, the NRC concluded that, for a bounding leakage case corresponding to a rupture of the SDV, the offsite doses would be

well within the limits of 10CFR100, and that adequate core cooling would be maintained.

Deletion of the requirement that the SDV be determined OPERABLE by testing the SDV vent and drain valves, as proposed in this TS Change Request, will have an insignificant effect on the probability of occurrence of malfunction of any plant equipment. The conditions in the SDV at the time of vent and drain valve closure are not appreciably different whether a scram is initiated from power operation or during shutdown conditions. In addition, this proposed TS change eliminates the potential need for an additional startup and shutdown cycle, along with the associated challenges to all systems and components, that would be required to satisfy the current TS requirements in the event a unit were to trip off-line shortly before a planned outage when the surveillance was scheduled to be performed. Furthermore, this proposed TS change does not affect the testing frequency for the valves.

This proposed TS change will not result in appreciably different conditions experienced by the valves as they close, and their ability to re-open is confirmed following each reactor scram from power conditions. The consequences resulting from a failed closed or failed open SDV vent or drain line have been evaluated and determined not to result in offsite doses that would exceed the limits specified in 10CFR100, or jeopardize adequate reactor core cooling capability. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated is not increased.

Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

2. <u>The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

The SDV is not an accident initiator. Deletion of the requirement that the SDV be determined OPERABLE by testing the SDV vent and drain valves from a configuration of less than or equal to 50% rod density, as proposed, will not create the possibility of a different type accident than any previously evaluated.

No plant equipment is added or deleted as a result of this proposed change. Since the initial conditions of pressure, temperature, and CRD system discharge flowrate have no appreciable effect on the SDV vent and drain valve performance, no different type of malfunction of any equipment important to safety is created.

Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. <u>The proposed TS change does not involve a significant reduction in a margin of safety.</u>

Since the initial test conditions of pressure, temperature, and CRD discharge flowrate will have no appreciable effect on the SDV vent and drain valve performance, conducting the surveillance test during shutdown conditions, as specified in this proposed TS change, will not affect the validity of the surveillance results with respect to the operability of the SDV to perform its intended safety function. Furthermore, every reactor scram is a serious plant

transient and a potential challenge to safety-related systems and equipment. The potential decrease in future scrams which could result from this proposed TS change will represent an improvement in overall safety.

Therefore, the proposed TS change does not involve a reduction in a margin of safety.

Information Supporting an Environmental Assessment

An Environmental Assessment is not required for the change proposed by this Change Request because the requested change to the LGS, Units 1 and 2, TS conforms to the criteria for "actions eligible for categorical exclusion," as specified in 10 CFR 51.22(c)(9). The requested change will have no impact on the environment. The proposed change does not involve a significant hazards consideration as discussed in the preceding section. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite. In addition, the proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed the proposed change to the LGS, Units 1 and 2, TS and have concluded that it does not involve an unreviewed safety question, and will not endanger the health and safety of the public.

SECTION 4: "OPTIONAL METHOD OF SCRAM TIMING" (TSCR 94-39-0)

PECO Energy Company, Licensee under Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station, Units 1 and 2, respectively, requests that the Technical Specifications (TS) contained in Appendix A to the Operating Licenses be amended as proposed herein, to revise TS Surveillance Requirements (SR) 4.1.3.2 and 4.2.3 to provide an optional method of verifying scram insertion times. The proposed changes to the TS are indicated by a vertical bar in the margin of TS pages 3/4 1-6 and 3/4 2-9, and TS Bases page B 3/4 1-2, for Units 1 and 2. The TS pages and Bases page showing the proposed changes are contained in Attachment 2.

We request that, if approved, the TS changes proposed herein be issued by January 28, 1995 and become effective immediately upon issuance of the amendments.

This TS Change Request provides a discussion and description of the proposed TS changes, a safety assessment of the proposed TS changes, information supporting a finding of No Significant Hazards Consideration, and Information Supporting an Environmental Assessment.

Discussion and Description of the Proposed Changes

Surveillance Requirement (SR) 4.1.3.2 of the LGS Technical Specifications requires that control rods which may have scram insertion times affected by system maintenance should be scram tested at greater than 950 psig reactor pressure. The proposed changes to the Technical Specifications will add an optional method to verify the scram insertion times of affected control rods. The changes to the Technical Specifications will allow control rods to be scram tested at zero reactor pressure and then again at greater than 950 psig but prior to achieving 40% rated reactor power. The zero reactor pressure scram test will verify the scram capability of the drives, while the 950 psig scram test will identify the effect of reactor pressure on the scram insertion times.

The optional method to verify scram insertion times for re-worked Control Rod Drives (CRDs) at zero and 950 psig reactor pressure will require an additional scram test but will allow operational flexibility during a maintenance outage. For example, CRD maintenance can be performed during a mid-cycle outage and then the scram insertion times can be verified without performing a vessel hydrostatic test.

The 950 psig pressure test should be performed as soon as practical during the plant startup but consistent with plant operating constraints, e.g. after the rod worth minimizer is no longer enforcing and when permitted by control rod patterns. The required test must be performed at greater than or equal to 950 psig reactor coolant pressure prior to exceeding 40% rated thermal power.

The original method of verifying scram insertion times for control rods whose scram times may have been affected by maintenance or modification activities has been developed as part of the BWR Owners Group Improved Technical Specifications program. The method is discussed for PECO Energy use in General Electric Company Report GENE-770-08-1290 "Control Rod Drive Scram Time Surveillance Testing Requirements" dated December, 1990.

Safety Assessment

The zero reactor pressure scram limit was established to ensure that control rods that satisfy the zero reactor pressure limit will also satisfy the design basis scram limit during reactor startup and up to 40% core thermal power. The zero reactor pressure scram limit of 2.0 seconds was developed by performing an evaluation of scram insertion times as a function of reactor pressure for various degraded control rod drive conditions. The degraded control rod conditions included the effects of increased scram inlet/outlet line friction, increased frictional forces on the collet fingers, variations in accumulator precharge pressure, and increased control rod friction. Combinations of these conditions were evaluated to determine the expected scram insertion times as a function of reactor pressure. Based on the results of this evaluation, the 2.0 second scram criterion was developed such that, if satisfied at zero reactor pressure, the scram insertion time limits would also be satisfied over the entire reactor pressure range. For these evaluations, it was determined that the 90% insertion scram limit provided the best measure of the control rod scram performance. Therefore, only the 90% control rod insertion limit of 2.0 seconds (notch 05) is specified.

Operability of control rods from zero reactor pressure up to 40% rated core thermal power can be assured by the following:

- Development of the 2.0 second scram time limit at zero reactor pressure. The 2.0 second time limit was established to account for the variability of scram insertion times as a function of reactor pressure. The 2.0 second time limit was developed to require the same level of scram performance for the re-worked control rods relative to a test performed during a hydrostatic test at 950 psig reactor pressure.

- Scram time testing at zero reactor pressure. This verifies proper operation of individual control rod system components required for a rod scram, e.g., scram inlet/outlet valves, electrical solenoids on scram valves, hydraulic piping, control rod seals, proper HCU operation, scram discharge volume system, and hydraulic scram accumulator.

- Basic design of the CRD system. UFSAR Section 4.6.3.1.1.5 states that if a rod can be moved by hydraulic drive pressure, then if may be expected to scram since a scram condition results in increased pressure below the control rod.

- Operation at low reactor power. There are significant thermal limit margins at low reactor power, e.g., power levels less than 40% rated.

The UFSAR for LGS evaluates the consequences of accidents related to the operability of the control rod drive system. The proposed changes to the Technical Specifications will not affect the UFSAR transient analyses. The UFSAR transient analyses which evaluated the CRD system performance were reviewed for the following two basic plant operating considerations:

- From zero reactor pressure to 950 psig reactor pressure, and

- From 950 psig reactor pressure to 40% rated core thermal power.

The limiting transient analysis for the reactor startup condition is the control rod drop accident (CRDA) which is evaluated in UFSAR Section 15.4.9. The CRDA analysis for LGS is based on a generic accident analysis which assumes a 5.0 second scram insertion time as input. The proposed Technical Specifications changes will not effect the CRDA because the zero reactor pressure scram time limit would be met during plant startup up to 950 psig reactor pressure.

Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed changes to the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) to revise Technical Specifications Surveillance Requirements 4.1.3.2 and 4.2.3 to provide an optional method of verifying scram insertion times do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Scram testing control rods at zero reactor coolant pressure will not increase the probability of any control rod related transient or accident discussed in the UFSAR. UFSAR Sections 15.4.1.1 and 15.4.1.2 discuss the consequences of inadvertent reactivity insertion errors due to the withdrawal of one or more control rods. The probability of one of these events occurring is a function of operator error and equipment malfunction and is not related to scram insertion times.

An inadvertent reactivity insertion error is prevented by existing system hardware interlocks and procedural controls that are not affected by scram time testing, e.g. core design, control rod design, one-rod-out interlocks, refueling interlocks, control rod sequence designations, and neutron monitoring systems.

UFSAR Section 15.4.9 discusses the control rod drop accident (CRDA). The CRDA assumes that a control rod suddenly drops out of the core due to equipment malfunction. The probability of occurrence of this accident is based on an equipment malfunction and is not affected by scram testing.

Engineering analysis and control rod scram test data demonstrate that a control rod drive that will meet the 2.0 second, scram insertion time, test criteria at zero reactor coolant pressure will also meet all scram insertion criteria during reactor startup and up to 40% rated thermal power.

The 2.0 second criterion was chosen to conservatively envelope scram time criteria and reactivity insertion criteria during reactor startup and up to 40% rated power conditions. Therefore, scram testing affected control rods at zero reactor pressure will not increase the consequences of an accident previously evaluated.

UFSAR Sections 15.4.1.1 and 15.4.1.2 evaluate reactivity insertion transients at low power conditions due to inadvertent control rod withdrawal errors. The UFSAR concludes that rod withdrawal errors at low power are adequately precluded by refueling interlocks, rod worth minimizer, operating procedures, core design, and control rod hardware design. However, should operator errors followed by equipment malfunctions result in an inadvertent criticality event, the IRMs would provide the necessary rod blocks or reactor scram to preclude the operational transient. Scram insertion time limits for the continuous rod withdrawal error during startup is 5.0 seconds. This scram time criterion will be met by a control rod that scrams within 2.0 seconds at zero reactor pressure. The 2.0 second scram criterion was established to ensure that affected control rods will meet scram requirements from zero reactor pressure up to 40% core thermal power.

Also, during low power operation (UFSAR Subsection 15.4.1.2) the rod worth minimizer (RWM) prevents the operator from selecting and withdrawing an out-of-sequence control rod. During reactor operation in the power range (UFSAR subsection 15.4.2) the rod block monitor (RBM) prevents a rod withdrawal error by inhibiting inadvertent control rod withdrawal. The RWM and RBM do not rely on a scram function to mitigate the consequences of a rod withdrawal error, and therefore the consequences of an accident evaluated in the UFSAR will not be affected by the proposed changes to the Technical Specifications.

The consequences of a control rod drop accident (UFSAR Section 15.4.9) would not be affected by scram testing a control rod at zero reactor pressure. The design basis accident of the rod drop accident assumes that control rods scram within 5.0 seconds. This 5.0 second scram test requirement will be met by control rods that meet the 2.0 second criterion at zero reactor pressure.

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

2. <u>The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

The changes to the Technical Specifications will allow control rods to be scram tested at zero reactor pressure and then again at rated reactor pressure prior to achieving 40% rated reactor power. No new types of accidents will be introduced since control rods that meet the 2.0 second scram criterion at zero reactor pressure will also meet all scram test criteria during reactor startup and at rated reactor pressure.

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

3. <u>The proposed TS changes do not involve a significant reduction in a margin of safety.</u>

The basis for shutdown margin (TS Bases 3/4.1.1) states that the reactor shall be made subcritical by a certain margin in all operating and shutdown conditions. The proposed changes to the Technical Specifications will not affect the shutdown margin requirements. Adequate shutdown margin is assured by core design, the one-rod-out interlock, and administrative controls.

The basis for the control rod insertion times (TS Bases 3/4.1.3) states that the scram times are to be consistent with those used in the transient and accident analysis. The proposed Technical Specifications changes will add an additional scram test verification for affected control rods at zero reactor pressure. The zero reactor pressure scram limit (2.0 seconds) was designed to ensure that the scram times assumed in the transient analysis will remain bounding from zero reactor pressure up to 40% rated core thermal power.

The basis for the control rod drop accident (TS Bases 3/4.1.3) states that the potential effects of a CRDA are limited. The proposed Technical Specifications changes will not effect the control rod drop results as the changes do not affect the reactivity of the rod or the rod drop velocity. The CRDA analysis is based on a 5.0 second scram insertion time criterion. The 2.0 second time criterion was established to ensure that the 5.0 second scram time criterion was valid from zero reactor pressure to 950 psig reactor pressure.

The basis for MCPR limits (TS Bases 3/4.1.3 and 2.3) states that the CRD system must bring the reactor subcritical at a rate fast enough to prevent MCPR from becoming less than the fuel cladding safety limit during the limiting power transient analyzed in the UFSAR. The proposed changes to the Technical Specifications will not affect the scram insertion rates that are used as input to the transient analysis. The zero reactor pressure scram limit of 2.0 seconds was developed to ensure that the control rods would meet their design scram insertion times from zero reactor pressure up to 40% rated power.

The proposed changes to the Technical Specifications will not increase the probability of inadvertent criticality because the changes do not affect the reactivity worth of control rods.

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

Information Supporting an Environmental Assessment

An Environmental Assessment is not required for the changes proposed by this TS Change Request because the requested changes to the LGS, Units 1 and 2, TS conform to the criteria for "actions eligible for categorical exclusion," as specified in 10 CFR 51.22(c)(9). The requested changes will have no impact on the environment. The proposed changes do not involve a Significant Hazards Consideration as discussed in the preceding section. The proposed changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed these proposed changes to the LGS, Units 1 and 2, TS and have concluded that they do not involve an unreviewed safety question, and will not endanger the health and safety of the public.

SECTION 5: "DEFINITION OF CORE ALTERATION" (TSCR 94-43-0)

PECO Energy Company, licensee under Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, requests that the Technical Specifications (TS) contained in Appendix A to the Operating License be amended, as proposed herein, to change the definition of CORE ALTERATION. The proposed change to the TS is indicated by a vertical bar in the margin of TS page 1-2 for Unit 1 and 1-2 and 1-3 for Unit 2. We request that if approved, that the TS change proposed herein be issued by January 28, 1995 and become effective immediately upon issuance of the amendments.

Discussion and Description of the Proposed Change

This TS Change Request would change the TS definition of CORE ALTERATION to exclude control rod movement in a control cell that contains no fuel and adopt the most recent language conventions approved by the NRC in conjunction with NUREG - 1433 "Standard Technical Specifications." The change will eliminate the requirement to have a Senior Reactor Operator (SRO) or Limited (to fuel handling) Senior Reactor Operator (LSRO) supervise control rod withdrawal of an empty cell, thereby reducing the required number of operators involved with control rod/blade refueling outage activities, pertaining to an off-loaded cell.

Safety Assessment

The change will revise the TS definition of CORE ALTERATION to exclude control rod movement in a control cell that contains no fuel assemblies and adopt the most recent language conventions approved by the NRC in conjunction with the BWR Standard Technical Specifications (NUREG - 1433). Currently the LGS Updated Final Safety Analysis Report (UFSAR) permits the withdrawal and or the removal of a control rod, provided there are no fuel assemblies in the associated cell. The definition change will eliminate the requirement to have a SRO or LSRO supervise control rod withdrawal in the off-loaded cell. Reactor refueling mode control rod interlocks, physical design of the control blade and administrative TS requirements which preclude inadvertent criticality will remain unchanged.

Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed change to the LGS, Unit 1 and Unit 2, TS, which will re-define CORE ALTERATION to exclude control rod movement under certain conditions, does not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards, set forth in 10 CFR 50.92 is provided below.

1. The proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed definition change removes the requirement to have a SRO or LSRO supervise control rod withdrawal in an off-loaded cell (i.e. no fuel assemblies). The evaluated accident potentially affected by this change is a control rod movement error during refueling resulting in inadvertent criticality. The supervision by a SRO or LSRO does not solely preclude inadvertent criticality and was not relied upon in the accident analysis contained in Section 15.4 of the LGS Updated Final Safety Analysis Report (UFSAR). The LGS reactor core is designed to have adequate shutdown margin with the highest-reactivity-worth control rod withdrawn. The withdrawal of a second rod with fuel assemblies loaded in the associated cell is prevented by a combination of the refueling, one-rod out interlock, and the Limiting Conditions for Operation (LCO) requirement of TS 3.9.10.2. The LCO requirements ensure adequate shutdown margin is present prior to control rod withdrawal. This is accomplished by testing during startup following a refueling outage or by analytical calculations during refueling. The refueling interlock will provide a rod block upon an attempt to withdraw a second control rod and

is required to be operable in accordance with TS 3.9.10.2 except for rods which have no fuel assemblies in the associated cell. The removal of the fuel assemblies from a cell eliminates the need for the reactivity control function of the associated rod. The physical removal of a control blade from the core by means of the refueling floor, first requires the removal of the four associated fuel assemblies in the cell. This design inherently prevents inadvertent criticality. Finally, this change is consistent with NUREG - 1433 "Standard Technical Specifications." Since current analysis permits the withdrawal of a control rod blade, provided the associated cell is unloaded, and refueling mode interlocks, administrative TS requirements and the physical design of the control blade and fuel cell, which preclude inadvertent criticality, will remain unchanged, this proposed change to the TS definition of CORE ALTERATION will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The LGS UFSAR currently permits control rod withdrawal and or removal, provided there are no fuel assemblies in the associated fuel cell. The definition change removes the requirement to have a SRO or LSRO supervise rod withdrawal in an off-loaded cell. The change potentially effects a control rod movement error during refueling resulting in inadvertent criticality which has been previously evaluated. In addition, the proposed change will make no physical changes to equipment or remove administrative controls which solely preclude inadvertent criticality. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS change does not involve a significant reduction in a margin of safety.

   The LGS TS bases address reactivity concerns, radiological releases, control rods, and monitoring of the facility related to this change. With the four fuel assemblies removed from a cell, the control rod/blade in the associated cell has no reactivity function. The reactivity issues addressed by TS are therefore unaffected. The rod/blade coupling integrity is maintained by the requirement to perform a coupling check following maintenance. Section 15.4 of the UFSAR states that there are no radiological releases in association with a rod withdrawal error during refueling. This conclusion is maintained by the administrative requirements of TS 3.9.10.2, the refueling interlocks for one-rod-out, and the physical design of the blade and cell. Lastly, the TS requirements for Emergency Core Cooling, Plant System, Containment, and Electrical Power Distribution System, which provide the systems necessary to mitigate the effects of a radiological release during control rod movement in an unloaded cell were reviewed and were found not to be adversely effected by the proposed change. Therefore, this change will not involve a significant reduction in a margin of safety.

Information Supporting an Environmental Assessment

An Environmental Assessment is not required for the Technical Specifications change proposed by this Change Request because the requested changes to the Limerick Generating Station, Units 1 and 2, TS conform to the criteria for "actions eligible for categorical exclusion," as specified in 10CFR51.22(c)(9). The requested change will have no impact on the environment. The proposed TS change does not involve a Significant Hazards Consideration as discussed in the preceding safety assessment section. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite. In addition, the proposed TS change does not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed this proposed change to the Limerick Generating Station, Units 1 and 2, Technical Specifications, and have concluded that it does not involve an unreviewed safety question.

ATTACHMENT 2
LIMERICK GENERATING STATION

UNITS 1 AND 2

Docket Nos.	50-352
	50-353
License Nos.	NPF-39
	NPF-85

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TSCR 94-33-0 "Control Rod Block Instrumentation"

List of Affected Pages

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3/4 3-62	3/4 3-62
3/4 3-88	3/4 3-88
3/4 9-4	3/4 9-4

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TSCR 94-34-0 "Standby Liquid Control System Operability in Mode 5"

List of Affected Pages

Unit 1	Unit 2
3/4 1-19	3/4 1-19

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TSCR 94-37-0 "Scram Discharge Volume Valve Testing"

List of Affected Pages

Unit 1	Unit 2
3/4 1-5	3/4 1-5

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TSCR 94-39-0 "Optional Method of Scram Timing"

List of Affected Pages

Unit 1	Unit 2
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3/4 2-9	3/4 2-9
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TSCR 94-43-0 "Definition of Core Alteration"

List of Affected Pages

Unit 1	Unit 2
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	1-3