evaluated. The proposed revision to our previous Technical Specification (TS) change request dated June 18, 1992, would help assure the availability of the block valves for accident mitigation. The availability of the block valves for accident mitigation has been found to outweigh any negative safety consequences associated with full cycle testing of a block valve isolating a pressurizer power-operated relief valves (PORV) with 'excessive" seat leakage. There would be no significant increase in the probability or consequences of an accident previously evaluated since this event is fully bounded by the failing open of a single pressurizer code safety relief valve event which is analyzed in Chapter 15 of the Updated Final Safety Analysis Report. Accordingly, the requested revision will not involve a significant increase in the probability or consequences of an accident previously evaluated.

- 2. The requested revision to our previous TS change request does not create the possibility of a new or different kind of accident from any accident previously evaluated. Periodic testing of the block valves in accordance with the requested revision is only intended to assure the functioning and capability of the block valves. The requested revision will only clarify the conditions when block valve surveillance testing is required. The performance of this testing is intended to improve block valve availability and thereby assure the capability of certain accident mitigation strategies identified within Abnormal and Emergency Operating Procedures. Therefore, the requested revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.
- 3. The requested revision to our previous TS change request does not involve a significant reduction in the margin of safety. The requested revision is intended to help assure block valve availability to support certain accident mitigation strategies. This additional assurance of block valve availability and functioning increases the margin of safety. Accordingly, the requested revision will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

*NRC Project Director:* William H. Bateman.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: December 14, 1994.

Description of amendment request: The proposed amendments would revise technical specifications related to allowed outage times (AOT) and surveillance test intervals (STI) for certain actuation instrumentation in the reactor protection system (RPS), primary containment isolation system (PCIS), emergency core cooling system (ECCS), recirculation pump trip, reactor core isolation cooling (RCIC), control rod withdrawal block, monitoring, and feedwater/main turbine trip systems. These changes are generally consistent with General Electric topical reports which have been reviewed and approved by the NRC. The changes also include revising the Feedwater/Main Turbine Trip LCO 3.3.8 action statement to achieve consistency with existing instrumentation LCOs; deleting the surveillance of the APRM Neutron Flux—High, Setdown functional unit in Operational Condition 1; revising the applicability of the provisions of Specification 4.0.4 to several Reactor Protection System and Control Rod Withdrawal Block Instrumentation surveillance requirements; adding the requirement to perform shiftly channel checks for applicable RPS, PCIS, ECCS, and RCIC instrumentation channels equipped with master trip units; and other changes to correct typographical errors and to delete cycle specific footnotes which are no longer applicable.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

It has been determined that the changes do not constitute a Significant Hazards Consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Units 1 and 2 in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

a. The proposed changes increase the STI and AOT for actuation instrumentation supporting RPS, ECCS, Isolation, CRBF, RCIC, ATWS-RPT, EOC-RPT, Monitoring, and Feedwater/Main Turbine Trip System Actuation functions. There are no changes in instrumentation configuration and function, and no instrumentation setpoints are changed. Because of this there is no change

in the probability of occurrence of an accident or the consequences of an accident or the consequences of malfunction of equipment. With respect to the probability of equipment malfunction, topical reports prepared by GE demonstrate that there is a reduction in scram frequency for the RPS, but in the case of the ECCS there is a small increase in the unavailability of the water injection function. This increase in unavailability was judged acceptable by GE. The NRC concurred with this conclusion in its review and approval of the topical reports. The proposed changes are consistent with the Safety Evaluation Reports issued for the topical reports.

b. The changes proposed for the Feedwater/Main Turbine Trip LCO action statements provide actions which are consistent with presently existing instrumentation LCOs. The design and function of the feedwater/main turbine trip instrumentation to trip the feedwater pumps and the main turbine upon detection of a Level 8 event is not altered. The probability and/or consequences of this moderate frequency transient are not increased.

c. The APRM Neutron Flux—High, Setdown scram setting provides adequate thermal margin between the setpoint and the safety limits for operation at low pressure and low flow during a plant startup. This function remains in effect until the mode switch is placed in the Run (Operational Condition 1) position, at which time it is bypassed. Deleting the requirement for the surveillance of the APRM Neutron Flux-High, Setdown functional unit in Operational Condition 1 is appropriate since its function is not applicable in this mode. This deletion serves to achieve consistency between Technical Specification Tables and the Bases section.

d. The changes associated with Specification 4.0.4 are administrative in nature and are intended to provide the plant operators with better guidance for its application. In cases where complete surveillances cannot be achieved, such as during a plant shutdown, then the required surveillances will be performed within 24 hours of entering the Mode or condition in which the surveillance is required. The stabilization of the plant will be of primary consideration. This change does not affect the evaluation for any accident presented in Chapter 15 of the UFSAR. The APRM Fixed Neutron Flux—High quarterly functional tests most of the APRM channel equipment associated with the APRM Neutron Flux-High, Setdown scram.

Additionally, the expected result of the functional tests associated with the SRMs, IRMs, and APRMs is to demonstrate the operability of the instrumentation. Therefore, 24 hours is a reasonable time to permit the surveillances to be performed upon entering the mode or condition in which the surveillance is required.

e. The proposal to include the performance of channel checks as requirements of technical specifications is administrative in nature. Presently, channel checks performed for the applicable analog instrumentation in reactor vessel water level applications is controlled solely by procedure. Adding this