

The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. The changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. The proposed changes to the TS are consistent with the intent of the flexibility currently provided in NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 2.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the updated final safety analysis report (UFSAR) because the accident analysis assumptions and initial conditions will continue to be maintained.

#### Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated.

Response: No.

The proposed change does not involve any physical alteration of plant systems, structures, or components. The requirements for the P/T limit curves and LTOP setpoints remain in place. The fundamental approach follows approved ASME and Westinghouse report methodology. The proposed curves and change to the enable temperature for LTOP system reflect changes in material properties acknowledged and managed by regulation and an upgrade in technology, which has been approved by ASME.

The proposed changes to allow operation with two pumps capable of injecting into the RCS and utilization of the RHR suction relief valves has been evaluated. The evaluation has shown that both the PORVs (power-operated relief valves) and RHR suction relief valves provide adequate relief protection of the RCS from the worst case pressure transient and provide equivalent protection to that already allowed by the current TS (technical specification).

The proposed changes do not introduce new failure mechanisms for system structures, or components not already considered in the UFSAR. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created because no new failure mechanisms or initiating events have been introduced.

#### Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety.

Response: No.

The proposed changes are developed utilizing the methodology of ASME XI, 10 CFR (part) 50 Appendix G, in conjunction with Code Case N-640 and Code Case N-641 methodology. Usage of these methodologies provides compliance with the underlying intent of 10 CFR (part) 50 Appendix G and provides operational limits that ensure failure of the reactor vessel will not occur. Although the Code Cases constitute

relaxation from the current requirements of 10 CFR (part) 50 Appendix G, the alternative methodology allowed by the Code is based on industry experience gained since the inception of the 10 CFR (part) 50 Appendix G requirements for which some of the requirements have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code Cases maintain a margin of safety that is consistent with the intent of 10 CFR (part) 50 Appendix G, *i.e.*, with regard to the margin originally contemplated by 10 CFR (part) 50 Appendix G for determination of RCS P/T limits.

The analyses completed for this proposed TS amendment demonstrate that established acceptance criteria continue to be met. Specifically, the P/T limit curves, LTOP setpoints, allowances for operating two pumps, utilization of RHR suction relief valves and LTOP  $T_{enable}$  values provide acceptable margin to vessel fracture under both normal operation and LTOPs design basis (mass addition and heat addition) accident conditions. The proposed changes to the TS are consistent with the intent of the flexibility currently provided in NUREG-1431, Standard Technical Specifications for Westinghouse Plants, Revision 2. Therefore, there will be no significant reduction in a margin of safety as a result of the proposed changes.

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.

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*Exelon Generation Company, LLC, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois; Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois*

*Date of application for amendment request:* March 28, 2003, as supplemented by letter dated October 23, 2003.

*Description of amendment request:* The proposed amendments would revise the technical specifications to reduce the main steam line low pressure primary containment isolation allowable value.

*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:

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

Response: No.

Current licensing bases events remain bounding for ATWS, transient, and accident analyses. For the bounding events, a reduction in the allowable value for the MSL LPIS produces no significant change in the limiting results with respect to the acceptance criteria. The proposed change does not alter the response of plant equipment to transient conditions, nor does it introduce any new equipment, modes of system operation or failure mechanisms. The proposed change does not adversely impact structures, systems, or components.

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

#### ECCS-LOCA Performance

In the analyses used to evaluate the ECCS-LOCA performance, the MSIVs are assumed to close at the start of the accident for all break locations. Therefore, the low pressure isolation trip is not used in the LOCA analyses and the LOCA analysis results are not affected by the reduction in the LPIS.

For large breaks in the MSL (both inside and outside containment), the MSIV closure is initiated by a high steam line flow signal at the beginning of the event, well before the LPIS is reached. For these cases, the ECCS performance is not affected by the reduction in the LPIS.

If the steam line break is too small to result in a high flow isolation signal, MSIV closure may be initiated by another signal (*e.g.*, high steam line tunnel temperature or low reactor water level) or it may occur due to the LPIS trip. In either case, steam line breaks of any size are not the limiting events with respect to ECCS performance, and a 40 psi reduction in the LPIS will not affect compliance with the acceptance criteria of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

Based on the above discussions, the reduction of the MSIV LPIS has no adverse impact on the plant response to a LOCA or on compliance with the acceptance criteria of 10 CFR 50.46.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of a previously evaluated ECCS-LOCA accident.

#### Containment System Response

In evaluating containment response to pipe breaks inside containment, the MSIVs are assumed to close at the start of the accident for all break locations in the containment system response analyses. Therefore, the low pressure isolation trip is not assumed and the analysis results are not affected by the reduction in the LPIS.

In the event that MSIV closure does not occur at the beginning of the accident, MSL isolation is effectively achieved as the pressure regulator closes the turbine control and bypass valves in an attempt to maintain turbine throttle pressure at the regulator setpoint of approximately 925 psig. Thus, for events other than breaks in the main steam