3/4.4.10 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation or inspection to determine the effects of the out-of-limit condition on the fracture toughness of the Reactor Pressure Vessel; determine that the Reactor Pressure Vessel remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

TABLE 4.4-5

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Values of ΔRT_{ndt} determined in this manner may be used until the next results from the material surveillance program, evaluated according to ASTM E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in FSAR Section 5.4. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the surveillance capsule exceeds the calculated ΔRT_{ndt} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{ndt}, is used and this includes the radiation induced shift, ΔRT_{ndt}, corresponding to the end of the period for which heatup and cooldown curves are generated.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation or inspection to determine the effects of the out-of-limit condition on the fracture toughness of the Reactor Pressure Vessel; determine that the Reactor Pressure Vessel remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

TABLE 4.4-5

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Values of ΔRT_{ndt} determined in this manner may be used until the next results from the material surveillance program, evaluated according to ASTM E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in FSAR Section 5.4. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the next surveillance capsule exceeds the calculated ΔRT_{ndt} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{ndt}, is used and this includes the radiation induced shift, ΔRT_{ndt}, corresponding to the end of the period for which heatup and cooldown curves are generated.

ATTACHMENT 2

Significant Hazards Evaluation Pursuant to 10 CFR 50.92 for the Deletion of the Reactor Vessel Material Surveillance Specimen Withdrawal Schedule from the Technical Specifications

Proposed Change

The purpose of this proposed change is to delete the reactor vessel surveillance specimen withdrawal schedule from the Technical Specifications. This change involves the deletion of Surveillance Requirement 4.4.10.1.2 and Table 4.4-5. In addition, the corresponding Bases is revised to eliminate the reference to Table 4.4-5 and indicate that the information previously provided in the table will be added to the FSAR.

Background

The Farley Nuclear Plant Unit 1 and 2 program for surveillance of reactor vessel materials is governed by 10 CFR 50 Appendix H and has been reviewed and approved by the Office of Nuclear Reactor Regulation. The schedule for removal of reactor vessel surveillance specimens is contained in Technical Specification Table 4.4-5 and complies with the guidance of ASTM E 185 as directed by 10 CFR 50 Appendix H. Periodically the need arises to update the information contained in Table 4.4-5. For example, since surveillance specimens are removed at the refueling outage nearest the scheduled removal exposure, the actual exposure at removal will likely vary from that indicated in the schedule. Following removal of each specimen, the schedule for withdrawal of remaining specimens is reviewed to ensure that the requirements of 10 CFR Appendix H are satisfied. Updating the surveillance specimen withdrawal schedule to reflect the actual time of specimen removal currently requires a license amendment.

Deletion of Table 4.4-5 from the Technical Specifications will allow future adjustments to the withdrawal schedule, including the lead factors, to be made without submittal of a license amendment request. It is anticipated that future changes to the surveillance specimen withdrawal schedule will only be necessary as a result of the analysis of surveillance specimens. Since the Code of Federal Regulations requires that the results of each surveillance specimen analysis be submitted to the NRC, the reactor vessel material surveillance program information will continue to be provided to the NRC. It should be noted that the Technical Specification Bases will retain the description of the reactor vessel material surveillance program including the reference of 10 CFR 50 Appendix H and ASTM E 185-82. The information currently included in Table 4.4-5 will be added to the FSAR. Removal of this information from the Technical Specifications will obviate the unnecessary use of licensee and NRC resources to process future license amendments. In addition, deletion of this material will enhance the useability of the Technical Specifications by plant operators resulting in an incremental benefit to plant safety.

Surveillance Requirement 4.4.10.1.2 requires that surveillance specimens be removed in accordance with the schedule in Table 4.4-5, examined in accordance with 10 CFR 50 Appendix H and that the results of the capsule examinations be used to update the reactor coolant system (RCS) heatup and cooldown limitation curves in Technical Specifications (Figures 3.4-2 and 3.4-3). All of the conditions of this Surveillance Requirement are inherent in the Code of Federal Regulations. The schedular requirements for withdrawal of specimens are included in ASTM E 185 which is referenced in Appendix H. Rules for the application of the results of material examinations used in the determination of heatup and cooldown limitations are found in 10 CFR 50 Appendix G which is also referenced by 10 CFR 50 Appendix H. Since Appendix G specifies the pressure and temperature limits for the reactor vessel based on the material properties, the Technical Specification heatup and cooldown curves must continue to be reviewed as results from the material surveillance program are obtained. Thus, the conditions of Surveillance Requirement 4.4.10.1.2 are redundant to the Code of Federal Regulations.

It is anticipated that NRC approval of this requested Technical Specification change will occur subsequent to Revision 6 of the Farley Nuclear Plant FSAR Update (July 1988). Revision 6 will add the information currently included in Technical Specification Table 4.4-5 to Section 5.4 of the FSAR. Accordingly, the proposed change to Technical Specification Bases 3/4.4.10 indicates that the schedule for withdrawal of surveillance specimens is shown in FSAR Section 5.4. The Bases will retain the reference to 10 CFR 50 Appendix H and ASTM E 185-82.

It should be noted that two minor editorial changes are being made on B 3/4 4-8. Specifically, the word "next" is being added as the last word on the first line of Unit 1 page B 3/4 4-8. The first sentence of Unit 2 page B 3/4 4-8 is being revised to indicate that the applicable version of ASTM E 185 is the 1982 edition. These changes are strictly editorial and are requested to restore the similarity of the Unit 1 and Unit 2 Technical Specifications.

Analysis

Alabama Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to this proposed Technical Specification change and considers the proposed change not to involve a significant hazards consideration. In support of this conclusion the following analysis is provided:

1) The proposed change does not significantly increase the probability or consequences of an accident previously evaluated because the reactor vessel material surveillance program is not affected by this proposed change. Implementation of the proposed change will delete a license requirement that is redundant to the Code of Federal Regulations. Thus, this proposed Technical Specification is considered to be administrative in nature.

2) The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated because implementation of this change will not alter plant configuration or mode of operation. Compliance with existing regulations will ensure continued confidence in reactor vessel material properties.

3) The proposed change will not involve a significant reduction in the margin of safety because the evaluation of reactor vessel material embrittlement is not altered by this change. Additionally, Surveillance Requirement 4.4.10.1.2 and Table 4.4-5 are not beneficial to the primary user of the Technical Specifications (i.e., the reactor operator). Thus, deletion of this material will actually enhance the useability of the Technical Specifications by plant operators resulting in an incremental benefit to plant safety.

Conclusion

Based upon the analysis provided herewith, Alabama Power Company has determined that the proposed Technical Specification change will not significantly increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, Alabama Power Company has determined that the proposed change meets the requirements of 10 CFR 50.92 and does not involve a significant hazards consideration.