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January 28, 1986
ST-HL-AE-1600
File No.: G9.17

Mr. Vincent S. Noonan, Project Director
PWR Project Directorate #5
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
DSER Items;
Pressurized Thermal Shock

Reference: 1: Letter ST-HL-AE-1515 dated 11/8/85; M. R. Wisenburg to G. W. Knighton

2. Letter ST-HL-AE-1580 dated 1//17/86; M. R. Wisenburg to G. W. Knighton

Dear Mr. Noonan:

This letter is provided to address Draft Safety Evaluation Report (DSER) Item # 110 (see Reference 1) regarding pressurized thermal shock. The contents are to be considered as a supplement to information provided in Reference 2 (the attachment replaces page 7 of the Reference 2 Attachment).

If you should have any questions on this matter, please contact Mr. M. E. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

JSP/yd

Attachment: DSER Item #110; Additional Information

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PDR ADOCK 05000498
E PDR

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Revised 12/2/85

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(Section 5.3.3.6, Page 5.3-16)

In addition, the reactor vessel is further qualified to ensure against unstable crack growth under faulted conditions. Actuation of the emergency core cooling system (ECCS) following a loss-of-coolant accident produces relatively high thermal stresses in regions on the reactor vessel which come into contact with ECCS water. Primary consideration is given to these areas to ensure the integrity of the reactor vessel under this severe postulated transient.

For the beltline region, significant developments have recently occurred in order to address Pressurized Thermal Shock (PTS) events. On the basis of recent deterministic and probabilistic studies, taking U.S. PWR operating experience into account, the NRC staff concluded that conservatively calculated screening criterion values of RT_{NDT} less than 270^{o} for plate material and axial welds, and less than 300^{o} for circumferential welds, present an acceptably low risk of vessel failure from PTS events. These values were chosen as the screening criterion in the PTS Rule for 10CFR50.34 (new plants) and 10CFR50.61 (operating plants) (Reference 5.3-9). The conservative methods chosen by the NRC Staff for the calculation of RT_{PTS} for the purpose of comparison with the screening criterion is presented in paragraph (b) (2) of 10CFR50.61. Details of the analysis method and the basis for the PTS Rule can be found in SECY-82-465. (Reference 5.3-10).

The reactor vessel beltline materials are specified in section 5.2.3. The design basis fluence of 2.76 x 10^{19} n/cm^2 which is the design basis fluence at the vessel inner radius, at 32 EFPY, at the peak azemuthal location, was used for calculating the RT_{PTS} values. RT_{PTS} is RT_{NDT}, the reference nil-ductillity transition temperature as calculated by the method chosen by the NRC Staff as presented in paragraph (b) (2) of 10CFR50.61, the "PTS Rule". The PTS Rule states that this method of calculating RT_{PTS} should be used in reporting values used to be compared to the above Screening Criterion set in the PTS Rule. The screening criteria will not be exceeded using the method

of calculation prescribed by the PTS Rule for the vessel design lifetime. The material properties, initial RT_{NDT}, and end-of-life RT_{PTS} values are in Tables 5.3-7 and 5.3-8. The materials identified in Tables 5.3-7 and 5.3-8 are those materials that are exposed to high fluence levels at the beltline region of the reactor vessel and are, therefore, the subject of the PTS Rule. The material properties in Tables 5.3-7 and 5.3-8 are those of the materials used in these reactor vessels and were obtained from certified test reports supplied by the vessel fabricator. The certified test reports were taken from samples of the actual beltline material. Amounts of residual elements in the beltline plate were measured from samples removed from the actual beltline plate. Weld samples were prepared using the same heat, lot and flux as the materials used in the vessel welds.

The principles and procedures of linear elastic fracture mechanics (LEFM) are used to evaluate thermal effects in the regions of interest. The LEFM approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the site of defect necessary to cause failure.