GULF STATES UTILITIES COMPANY

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July 25, 1985
RBG - 21682
File Nos. G9.5,G9.8.6.2

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

River Bend Station - Unit 1
Docket No. 50-458

Enclosed is Gulf States Utilities' (GSU) revised response to the NRC Staff's Request for Additional Information 210.110 delineated in Mr. A. Schwencer's letter of April 30, 1985 (RBC-31,879). Pursuant to our discussions with your Mechanical Engineering Branch, the enclosed will be incorporated into a future Final Safety Analysis Report amendment.

Sincerely,

J. E. Booker

J. E. Booker
Manager-Engineering
Nuclear Fuels & Licensing
River Bend Nuclear Group

JEB/ERG/je

Enclosures

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QUESTION 210.82 (3.6.2)

Provide the basis for assuring that the feedwater isolation check valves can perform their function following a postulated pipe break of the feedwater line outside containment.

RESPONSE

11 The response to this request is provided in revised Appendix 3C, Section 3C.2.2. See also Question and Response 210.110. The details of this analysis have been provided under separate cover (RBG-21487 dated July 8, 1985).

QUESTION 210.110

Your response to Q 210.82 regarding the basis for assuring that the feedwater check valves can perform their intended function following a feedwater line break outside containment is not totally acceptable. In your letter from J. E. Booker to H. Denton dated December 17, 1984, you provided a table which summarized the results of your feedwater check valve analysis. The staff finds that additional information is needed to justify the allowable values used for the disk shear load and strain limits.

In Note (1) to Enclosure 2 of your letter you state:

> "Strain due to absorption of energy must be less than 0.7 times the ultimate elongation (based on Table CB-3700-1 of the ASME Code, Section III, Division II, 1983 Edition)."

The staff does not find the use of Table CB-3700-1 to be acceptable for Division 1 components including valves. Table CB-3700-1 is applicable to Division 2 components (i.e, concrete reactor vessels and containment).

Furthermore, in Note (1) you state:

> "It is assured that strain rates are sufficiently high to warrant doubling the material allowables (Ref. Juvinal, R. C., Stress, Strain, and Strength - McGraw-Hill, 1967, pg-168)."

The staff finds that the figure provided in the referenced text book shows the effect of strain rate on tensile properties of typical mild steel at room temperature. However, the text book figure provides only general information not suitable for specific design application without more details. The use of the figure in the text book is not appropriate for the specific valve design because it does not represent the yield strength of the disk material (i.e., SA-216 WCC) at temperature. Furthermore, the text book states that because it is difficult to predict the conditions of impact strain rate, it is common to design parts for impact based on empirically determined stress impact factors together with the static strength of the materials. It appears that the shear load on the disk for check valve 1B21*AOVF032A, B is near the allowable value. (Factor of Safety equal to 1.05). Thus, the staff is concerned that the acceptability of doubling the material allowables at temperature is not well-justified, and because of the small design margin and the uncertainties involved, the shear load on the disk might be unacceptable. Additionally, it appears the calculated values were established assuming strain hardening effects. Provide the basis for this assumption if used. Furthermore, it has been shown for mild steel at elevated temperatures that a negative slope occurs in the stress-strain rate curve at the higher strains which is generally associated with strain aging.

The staff requests that you provide a detailed discussion of the valve parts which do not satisfy linear stress limits per the ASME Code and the basis for concluding that the leakage of the reactor coolant pressure boundary through the feedwater check valves following the postulated break in the feedwater line outside containment is acceptable using your calculated strain values.

RESPONSE

The response to this request is provided as the revised response to Question 210.82.

qualified for the post-LOCA drywell environment as discussed in Section 3.11.

Inside the Steam Tunnel

All feedwater piping from inboard of the first moment-limiting (zero gap) restraint in the drywell to
15 outboard of the jet impingement wall meets the criteria for no postulated breaks as discussed in Section 3.6A.

In the Auxiliary Building

The reactor vessel water is protected from blowdown, following a postulated rupture of the feedwater piping outside the containment, by check valves 1B21*F010A, B inside the containment and by testable check valves 1B21*AOVF32A, B outside the containment. Breaks are not postulated in the piping between the valves because that region is classified as a break exclusion area. Analyses were performed to demonstrate that the feedwater isolation check valves can perform their function following a postulated pipe break of the feedwater line outside the containment.

The reverse flow caused by the sudden pressure reduction at the break rapidly closes both valves. A dynamic analysis
16 was performed to obtain the forcing function for use in the valve stress analysis. First, a flow transient analyses was performed for the feedwater system to simulate the pipe break condition. The reverse flow condition at the check valve location was determined using the SWEC computer program WATHAM (Section 3A.21). Hydrodynamic torque exerted on the valve disk by the reverse flow was applied to determine the valve closing time and the disk impact speed on its seat.

A stress analysis was conducted to determine the ability of these isolation valves to withstand impact of the disk on the seat, at the speeds obtained from that dynamic analysis. The acceptance criterion is that gross leak rates do not occur because of disk rupture, serious fracture of the seat/disk interface, or misalignment of the disk.

Loads on the critical elements, i.e., the disk, tail link, rockshaft and seat, were computed by simulation of the impact dynamics using the STARDYNE and GT-STRUDL computer program (Section 3A.5). Both uniform and point impact were considered to determine the worst case kickback loads generated by the point of impact not being at the center of

~~percussion. Seismic, hydrodynamic, and dead loads were not considered because of their insignificant magnitude compared to impact loads.~~

~~In most cases, linear stresses were below their allowables. If not, a nonlinear time history strain analysis was conducted to demonstrate integrity.~~ 16

~~It is concluded that both valves will remain intact and perform their function following rupture of the feedwater piping outside the containment.~~

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An inelastic analysis was performed in accordance with Appendix F of the ASME III Code (1977) for Class 1 service, using the ANSYS computer program (Appendix 3A, Section 3A.25). The non-linear stress/strain relationship was conservatively approximated by a bilinear curve with the strain at ultimate stress equal to 2/3 the elongation at temperature as provided in ASME II, adjusted for strain rate and temperature effects. This analysis has verified that structural integrity of the feedwater check valves is maintained. Any long term leakage through the check valves is controlled by redundant motor-operated valves which are closed and sealed by the penetration valve leakage control system at approximately 25 min. after the accident. Note that, as discussed in FSAR Section 15.6.6, a feedwater line break outside containment is less limiting than other postulated LOCA's.