FRACTURE MECHANICS EVALUATION OF THE REACTOR VESSEL OF THE SMUD UNIT AT RANCHO SECO FOR THE UNANTICIPATED TRANSIENT ON 3/20/78

A fracture mechanics analysis of the reactor vessel has been performed using the conservative approach outlined in the ASME Code, Section III, Appendix G. Specific details of the analytical approach are documented in B&W Topical Report BAW-10046A, Rev. 1.

Analyses were performed on the two most critical areas of the reactor vessel - the beltline region and the outlet nozzle region. At the lowest temperature during the transient the material is still in the upper shelf region (ductile behavior). Due to the low level of radiation, degradation to beltline region materials is not significant enough to produce a shift in the transition reference temperature. Consequently, the beltline region materials are also at the upper shelf toughness region.

Factors-of-safety for the beltline region and outlet nozzles have been calculated as follows:

$$\text{F. of S.} = \frac{K_{IR}}{K_{Im} + K_{It}}$$

where K_{IR} is the reference stress intensity factor

K_{Im} is the stress intensity factor due to pressure

K_{It} is the stress intensity factor due to the thermal gradient through the thickness

The factor-of-safety for the beltline region is 1.8.
The factor-of-safety for the outlet nozzles is 1.2.

Detailed calculations are provided on sheet 3 of 3.

It should be recognized that this is a very conservative analysis. A postulated flaw size of 1/4t is assumed for the beltline region and the 3.0" nozzle corner flaw is assumed for the outlet nozzle. The material fracture toughness (reference stress intensity value) used is 200 ksi$\sqrt{\text{in}}$. Finally, the calculation of the stresses associated with the transient is conservative.

Charles E. Harris

Charles E. Harris, PE
March 23, 1978

Henrik S Palme

Henrik S. Palme
March 23, 1978

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ADJUSTED REFERENCE TEMPERATURE OF BELTLINE REGIO

WF-70 in the longitudinal seam of the lower shell is the governing weld in the beltline region. Approximately 75% of this weld from the I.D. is WF-70, therefore, the 1/4 T and 3/4 T RT_{NDT} will be determined for WF-70.

Number of EFPD ACCUMULATED
AT TIME OF TRANSIENT 3/20/78 = 567 = 1.55 EFPY

This corresponds to a conservative max fluence at the vessel inner surface $= 1.1 \times 10^{18}\ n/cm^2$ ‡

$$F_{1/4T} = 6.1 \times 10^{17}\ n/cm^2$$

$$F_{3/4T} = 8.1 \times 10^{16}\ n/cm^2$$

For WF-70 — $\%Cu = 0.27$, $\%P = 0.014$ — $F_{1/4}T = 4.7 \times 10^{17}\ n/cm^2$, $F_{3/4}T = 6.3 \times 10^{16}\ n/cm^2$

Note that the longitudinal seam is not the same azimuthal location as the maximum fluence.

From Reg. 6.1.99 $\quad \Delta RT_{NDT})_{1/4T} = 56°F$

$$\Delta RT_{NDT})_{3/4T} = 21°F$$

$RT_{NDT} = 20°F$ unirradiated

The adjusted reference temperatures are:

$$\boxed{\begin{array}{ll} 1/4 T & Adj\ RT_{NDT} = 76°F \\ 3/4 T & Adj \cdot RT_{NDT} = 41°F \end{array}}$$

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‡ ...ign curve is from the report for ...se Ib of the User's Group ...ram for the evaluation of Reactor ...sel material properties.

Charles E. Harris, PE
March 22, 1978

E*

Docket No. 50-346

Toledo Edison Company
ATTN: Mr. Lowell E. Roe
Vice President, Facilities
Development
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Gentlemen:

SUBJECT: NATURAL CIRCULATION TEST - DAVIS BESSE, UNIT NO. 1

We have evaluated your request, as specified in your letter of February 13, 1978, that Davis Bess Unit 1 be relieved from the requirements of conducting a natural circulation test.

We will still require that a natural circulation test be conducted in accordance with your commitment and test plan described on page 14-103, Revision No. 27 of the FSAR to meet our position sent to you in our letter dated January 24, 1977. Our January 24, 1977 letter described the reasons supporting the staff position.

exemption?

However, we find your other request for proceeding to 100% power and conducting the natural circulation test at the first opportunity after the current coal crisis to be acceptable, but in any event, the test should be conducted within at least 120 days from the date of this letter.

Our basis for allowing you to proceed as stated above is based upon the completed Oconee No. 1 tests which have confirmed acceptable natural circulation, and your calculational results presented to the NRC staff in Bethesda, Maryland on February 14, 1978, which demonstrated that the Davis Besse, Unit 1 177 FA raised loop design should have natural circulation flow at least comparable to the completed results of the Oconee No. 1 tests.

Sincerely,

POOR ORIGINAL

By what authority are such actions allowed?

Roger S. Boyd, Director
Divison of Project Management
Office of Nuclear Reactor Regulation

for comment

cc: See Page 2

OFFICE	LWR 1	LWR	DSS	DSS	DPM	DPM
SURNAME	LEngle/red JStolz	DVassallo	GMazetis	RMattson	RCDeYoung	RSBoyd
DATE	2/16/78	2/ /78	2/16/78	2/ /78	2/ /78	2/ /78