CEN - 189
APPENDIX I

50-382

EVALUATION OF PRESSURIZED THERMAL SHOCK EFFECTS
DUE TO
SMALL BREAK LOCA'S WITH LOSS OF FEEDWATER
FOR
THE WATERFORD REACTOR VESSEL

Prepared for

THE LOUISIANA POWER AND LIGHT COMPANY

NUCLEAR POWER SYSTEMS DIVISION
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POWER
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COMBUSTION ENGINEERING, INC

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ABSTRACT

This Appendix to CEN-189 provides the plant-specific evaluation of pressurized thermal shock effects due to small break LOCA's with extended loss of feedwater for the Waterford reactor vessel. It is concluded that crack initiation would not occur for the transients considered for more than 32 effective full power years, which is assumed to represent full plant life.

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I1.0 PURPOSE

This Appendix provides the plant-specific evaluation of pressurized thermal shock effects of the SB LOCA + LOFW transients presented in the main body of the CEN-189 report for the Waterford reactor vessel.

I2.0 SCOPE

The scope of this Appendix is limited to the evaluation of the SB LOCA + LOFW transients presented in CEN-189, as applied to the Waterford reactor vessel.

Other C-E NSSS reactor vessels are reported in separate Appendices.

I3.0 INTRODUCTION

This Appendix to CEN-189 was prepared by C-E for Louisiana Power and Light for their use in responding to Item II.K. 2.13 of NUREG-0737 for the Waterford reactor vessel.

This Appendix is intended to be a companion to the CEN-189 report. The transients evaluated in this Appendix are those reported in Chapter 4.0 of the main report. Chapter I5 of this Appendix reports the plant-specific fluence distributions developed as described in Chapter 5.0 of the main report. Chapter I6 reports the plant-specific material properties and change of properties due to irradiation, based on the methods of Chapter 6.0 of the report. Chapter I7 reports the results of comparing the fracture mechanics results of Chapter 7.0 of the report, to a set of material properties which are conservative with respect to the plant sepcific properties reported in Chapter I6. This additional conservatism was not removed because of the favorable results.

I4.0 THERMAL HYDRAULIC ANALYSES

The pressure-temperature transients used to perform the plant-specific vessel evaluation reported in this Appendix are those reported in Chapter 4.0 of CEN-189. As discussed in the body of the report, there are several plant parameter conservatisms included in the analyses to develop these transients due to the reference plant approach used which could be eliminated by performing more detailed plant-specific thermal-hydraulic system analyses. Removal of these available conservatisms by additional analyses was not performed due to the favorable conclusion achieved.

I.5.0 FLUENCE DISTRIBUTION

The Waterford Station is not yet in operation and has not yet completed a surveillance capsule evaluation. Since the vessel beltline materials are low copper content, detailed fluence profiles were not necessary for demonstration of acceptable PTS capability. Accordingly, the FSAR end of life peak fluence prediction was used to estimate end of life material properties. Also, in order to evaluate the sensitivity of the fluence prediction value, material properties were also determined assuming an end of life fluence twice the FSAR prediction value.

APPENDIX I WATERFORD UNIT #3

I.6 MATERIAL PROPERTIES

The chemistry and initial (pre-irradiation) toughness properties of the Waterford Unit #3 reactor vessel beltline materials are summarized in Table I6-1. The most controlling material in terms of residual chemistry (copper and phosphorus) and initial RT_{NDT} is plate M-1004-2 from the lower shell course. The predicted RT_{NDT} shift based on the maximum design fluence, $3.68 \times 10^{19} n/cm^2$ (E>1MeV) at the inside **surface of the** reactor vessel, is 77F using Regulatory Guide 1.99. This will result in an adjusted RT_{NDT} at end-of-life (32 effective full power years) of 99F at the vessel inside surface. If the design fluence was increased by a factor of two to $7.36 \times 10^{19} n/cm^2$, the RT_{NDT} shift is predicted to be 109F for an adjusted RT_{NDT} of 131F.

TABLE I6-1

WATERFORD UNIT #3
REACTOR VESSEL MATERIALS

Product Form	Material Identification	Drop Weight NDTT (°F)	Initial[d] RT_{NDT} (°F)	Chemical Content (%) Nickel	Copper	Phosphorus
Plate	M-1003-1	-30	-30	0.71	0.02	0.004
Plate	M-1003-2	-50	-50	0.67	0.02	0.006
Plate	M-1003-3	-50	-50	0.70	0.02	0.007
Plate	M-1004-1	-50	-16	0.62	0.03	0.006
Plate	M-1004-2	-20	22	0.58	0.03	0.005
Plate	M-1004-3	-30	16	0.62	0.03	0.007
Weld	101-124 A,B,&C[a]	-60	-60	0.96	0.02	0.010
Weld	101-142 A,B,&C[b]	-80	-80	<0.20	0.03	0.007
Weld	101-171[c]	-70 to -30	-70 to -80	0.16	0.05	0.008

a Intermediate shell course longitudinal seam weld

b Lower shell course longitudinal seam weld

c Intermediate-lower shell girth weld

d Plate RT_{NDT} determined using Branch Technical Position MTEB 5-2; weld RT_{NDT} determined in accordance with ASME Code, Section III, NB-2300

I.7.0 Waterford 3 Vessel Integrity

The fracture mechanics analysis is performed uing upper bound data for fluence and material properties in the Waterford 3 vessel. The peak vessel fluence is considered to occur at the point of maximum RT_{NDT}. The material toughness properties K_{IC} and K_{Ia} are determined from the calculated temperatures for the SBLOCA + LOFW transients using the method described in Section 7.3.3 and predicted RT_{NDT} values through the depth of the wall. Critical crack depth diagrams are constructed from the applied K_I vs crack depth curves at the mid-core level of the vessel and the calculated material toughness curves. In this manner the integrity of the Waterford 3 vessel is evaluated for the SBLOCA + LOFW transients.

I.7.1 Summary of Physics and Material Data Input to Fracture Mechanics Analysis

A nominal design fluence value of 3.68×10^{19} n/cm^2 (E > 1 MeV) was used to approximate the end-of-life fluence for the Waterford 3 vessel, as well as a conservative upper bound of 7.36×10^{19} n/cm^2 or double the predicted end-of-life value. The peak fluence is considered to be uniform around the vessel. A conservative radial fluence attenuation was used such that:

$$\frac{F}{F_o} = \exp\ (-8.625 \text{ in. } \times .33 \text{ in.}^{-1}) \cdot (a/w)$$

$$= \exp\ (-2.85) \cdot (a/w)$$

where F = point fluence in wall
F_o = peak fluence at surface
a/w = fractional wall depth

Upper bound materials data were used to conservatively envelope all plate and weld materials, which are as follows:

PCT.	Cu	=	.10
PCT.	P	=	.010
Initial RT_{NDT}		=	+40^{o}F

The shift in the value of the RT_{NDT} was determined using the method of Reg. Guide 1.99. This produces an end-of-life prediction for the surface RT_{NDT} of 174°F using the nominal design fluence. A predicted surface RT_{NDT} value of 230°F is determined for a fluence double the nominal design fluence.

I.7.2 Results of Fracture Mechanics Analysis for SBLOCA + LOFW ⟶ Restoration of Feedwater (Case 5)

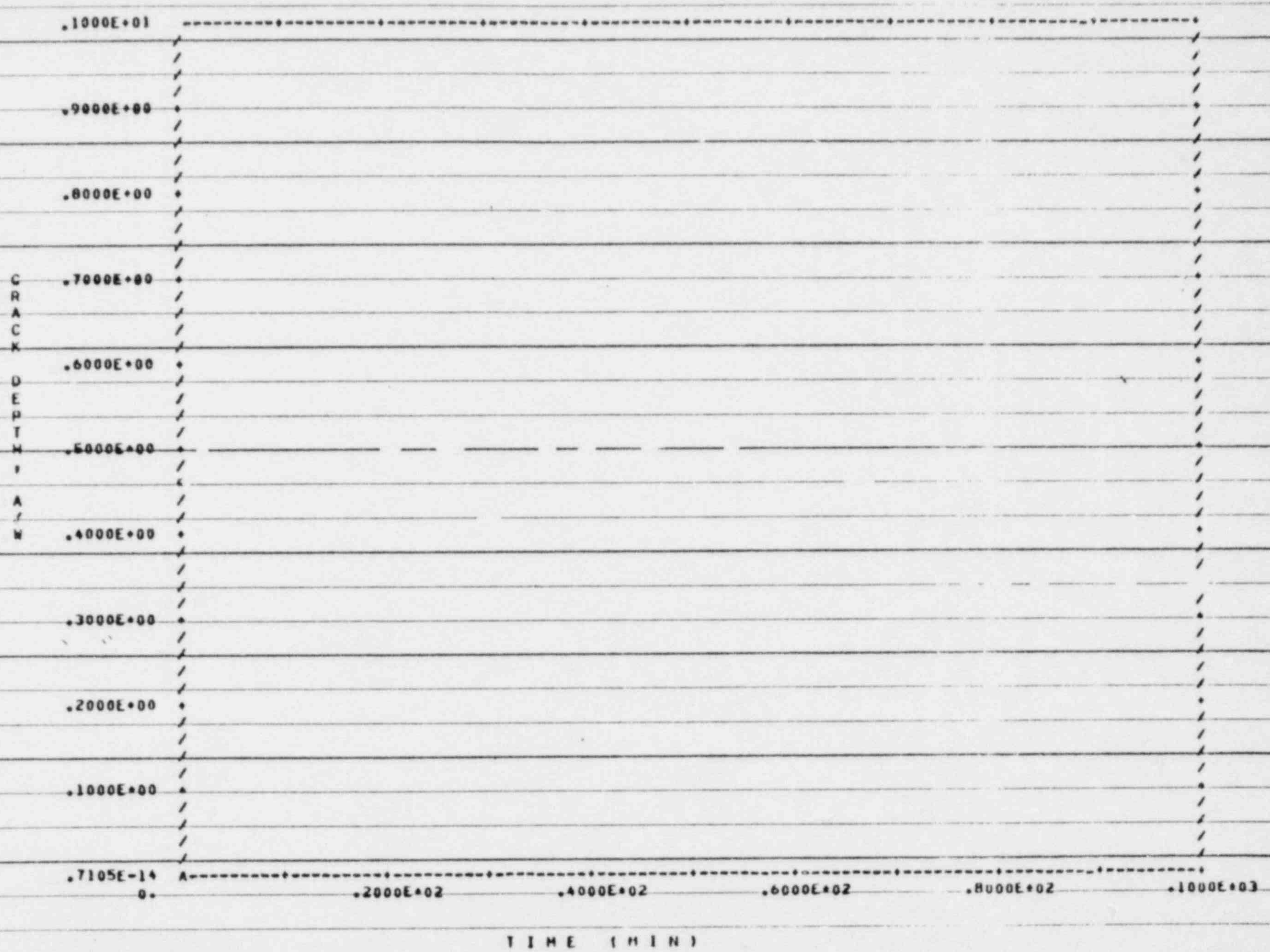
The stress analysis for this case is presented in Section 7.8.2 of the report. The fracture mechanics analyses were performed using upper bound properties for the Waterford 3 vessel and conservative end-of-life fluence levels. The critical crack depth diagram is constructed using the stresses in the transient at the mid-core level coincident with the peak fluence and material properties. Figure I.7-1 shows the critical crack depth diagram for a nominal design fluence of 3.68×10^{19} n/cm^2. The calculated shifts in RT_{NDT} are relatively low, and for this transient loading condition the initiation toughness level is not exceeded. Therefore, no crack initiation would occur for this combination of loading, fluence, and material properties.

Figure I.7-2 shows the critical crack depth diagram for the same transient loading and upper bound material properties, but twice the nominal design fluence. From the figure it is apparent that no crack initiation would occur for this transient even with fluence levels greatly exceeding the nominal design fluence.

I.7.3 Conclusion

These results demonstrate that the integrity of the Waterford 3 vessel would be maintained throughout the assumed life of the plant for the SBLOCA + LOFW transient with recovery of feedwater.

CRITICAL CRACK DEPTH VS. TIME



FLUENCE = .37E+20 N/SQ. CM
INIT RTNDT = 40. DEG. F
PCT. CU = .10
PCT. P = .010
RTNDTS = 174. DEG. F

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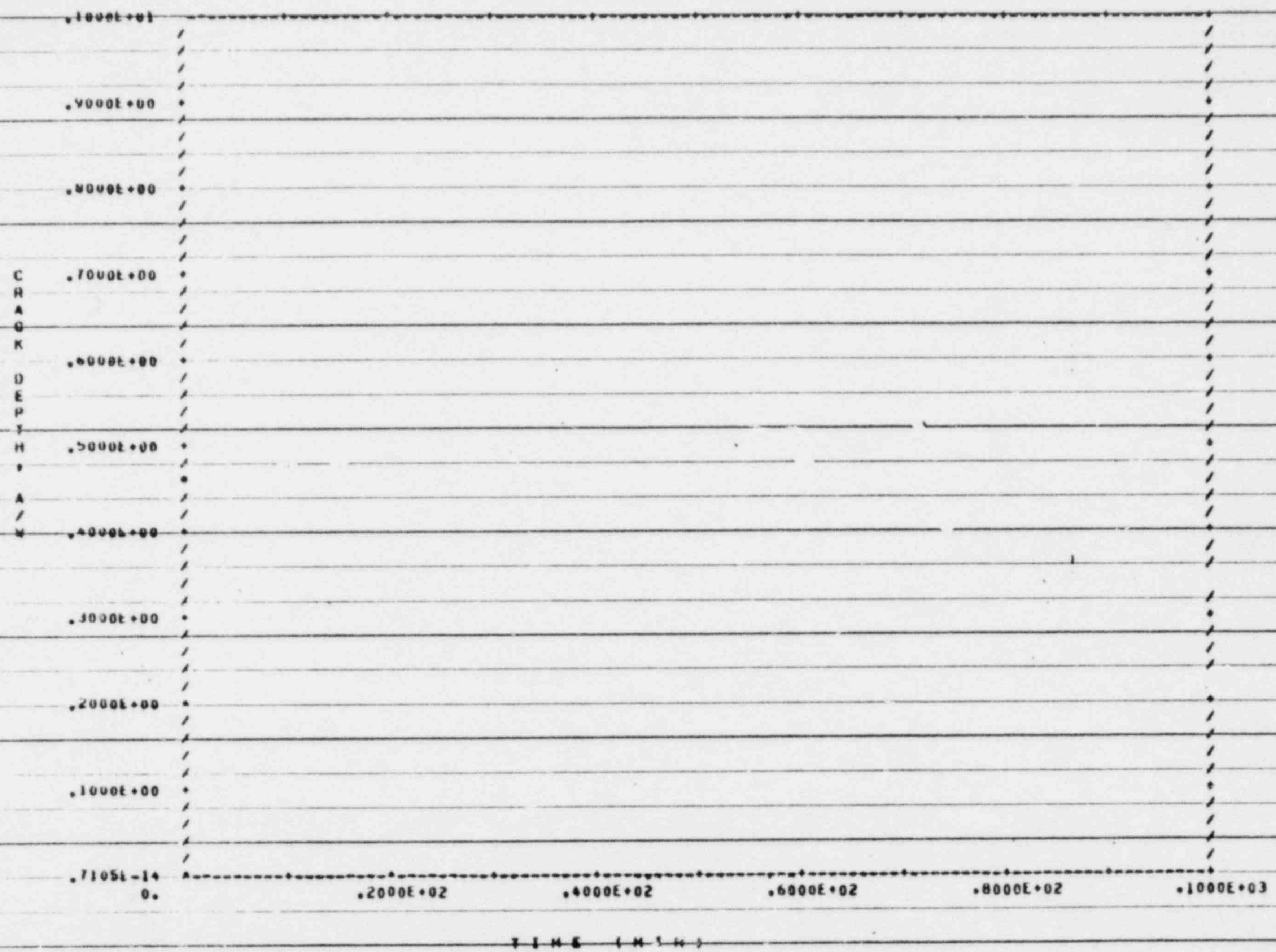
Figure I.7-1

CRITICAL CRACK DEPTH VS. TIME



FLUENCE = .74E+20 N/SQ. CM
INIT RTNDT = 40. DEG. F
PCT. CU = .10
PCT. P = .010
RTNDTS = 230. DEG. F

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Figure I.7.2

I8.0 CONCLUSIONS

This Appendix to CEN-189 provides the results of analytical evaluations of pressurized thermal shock effects on the Waterford reactor vessel for cases of a SBLOCA + LOFW, in response to the requirements of Item II.K.2.13 of NUREG-0737. Two different scenarios were chosen for evaluation based on remedial actions to prevent inadequate core cooling:

1. SBLOCA + LOFW + PORV's opened after 10 minutes
2. SBLOCA + LOFW + Aux. FW reinstated after 30 minutes

Thermal-hydraulic system transient calculations were performed on a reference-plant basis, as reported in CEN-189 with the parameter variations over the range representing all operating plants. Four different cases were analyzed for each of the two different scenarios defined above, for a total of eight cases. The most challenging of the two different scenarios was analyzed using linear elastic fracture mechanics methods to determine the critical crack tip stress intensity values for comparison to plant specific materials properties at various times in plant life. The effect of the warm prestress phenomenon is identified where applicable for each transient, and credited where appropriate.

In this Appendix, the results of plant specific peak neutron fluence predictions are superimposed on plant specific material properties to define vessel capability versus plant life. The results of the generic LEFM analyses were evaluated using the plant specific material properties. It is concluded that crack initiation would not occur due to the SBLOCA + LOFW transients considered, for more than 32 effective full power years of operation, which is assumed to represent full plant life.

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