GE Nuclear Energy

NEDC-32015
GE-NE-523-59-0591
Class II
September 1994
Revision 1

NINE MILE POINT 2
FATIGUE EVALUATION
POWER UPRATE OPERATING CONDITIONS

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REPORT CERTIFICATION

This design certification, with the documents listed below, constitutes the reconciliation to the Nine Mile Point 2 (NMP2) reactor pressure vessel Code Stress analysis for a power uprate program. I certify, to the best of my knowledge and belief, that the stress report, listed below is correct, complete, and complies with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plants Components, - 1971 Edition with Addenda to and including Winter 1972. I also hereby certify that I am a duly Registered Engineer under the laws of the State of California.

SUPPORTING DOCUMENTS

Document	Revision	Type of	Title
25A5000	0	Design Specification	Reactor Vessel - Power Uprate
NEDC-32015	1	Stress Report	Nine Mile Point 2 Fatigue Evaluation Power Uprate Operating Conditions

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NOMENCLATURE

σ	- Stress (psi or ksi)
α	- Influence Factor (psi/Δ°F)
S_m	- Allowable Stress Intensity
S_n	- Maximum Stress Intensity Range (P+Q stress)
S_p	- Peak Stress Intensity Range
S_{alt}	- Alternating Stress Intensity
E_a	- Actual Elastic Modulus at 552°F (26 x 10^6 psi)
E_c	- Fatigue Curve Elastic Modulus Value (30 x 10^6 psi)
K_e	- Strain Concentration Factor
ΔU	- Incremental Fatigue Usage Factor
$\Sigma(\Delta U)$	- Sum of Differences between Original and Power Uprate Incremental Usages
U_{CUM}	- Cumulative Fatigue Usage Factor
N_{allow}	- Number of Allowable Fatigue Cycles
N_{actual}	- Number of Actual Fatigue Cycles
M+B	- Membrane + Bending Stresses
M+B+P	- Membrane + Bending + Peak Stresses
P+Q	- Primary + Secondary Stresses
O, PU	- "Original" and "Power Uprate" Subscripts

1.0 INTRODUCTION

A fatigue evaluation of the Nine Mile Point 2 (NMP-2) Nuclear Power Plant has been performed for the new power uprate operating conditions. The original analysis is still bounding with respect to the new maximum vessel temperature, pressure, flow, and feedwater (FW) pressure for NMP-2. Only FW temperature and steam outlet flow increased beyond the upper bounds of the original analyses.

An increase of approximately 6% in steam outlet flow resulted in a 4.7% increase in heat transfer coefficients for the steam outlet nozzle. However, it was found that an increase of this order has a negligible effect on temperatures. Since no other operating parameters increased for the steam outlet nozzle, fatigue usage was not re-evaluated for this component.

The new operating parameters for FW temperature and flow are given in the Power Uprate Design Specification (Reference 1). The upper limit of the new operating pressure is 1055 psia (Reference 2) and is still within the operating conditions defined in the current thermal cycle diagram (Reference 3). The governing stress report (Reference 4) includes a generic analysis which is bounding for all replaceable sparger type FW nozzles. The thermal cycle diagrams contained in the referenced report were the basis for the original analysis. Those cycle diagrams remain applicable as the design basis and will be modified as necessary to recalculate thermal stresses.

Originally, a stainless steel (SS) clad location (element #374) was reported to have the most severe fatigue usage of 0.9503 (Reference 4). However, per Appendix 10 of the Reactor Vessel Purchase Specification (Reference 6), an additional environmental fatigue analysis (Reference 5) resulted in an incremental usage factor of 0.157 to account for the effects of stress corrosion in carbon steels. As a result of this analysis, another critical location was identified in the carbon steel section of the nozzle (element #228). Following a subsequent new loads assessment to account for new pool hydrodynamic loads (Reference 7), a cumulative usage of 0.968 was reported for the carbon steel location (element #228).

The analysis presented in this report will address the effects of power uprate on the two most critical locations identified above (element #374 and element #228).

2.0 ANALYSIS AND RESULTS

The following analysis methods are consistent with ASME Code Section III methodology and requirements.

2.1 General Analysis and Procedure

Only stress cycle combinations with incremental usage factors of U>0.01 were re-evaluated. Combinations with U<0.01 were neglected. The remaining dynamic cycles are not influenced by power uprate, and rapid cycles actually decrease as a result of power uprate. Therefore, the usage for these cycles was not changed. Since only thermal stresses increase due to power uprate, mechanical and dynamic stresses need not be considered. As such, the overall stresses were increased by a "delta" change in thermal stress.

The allowable stress intensity, S_m, and actual elastic moduli, E_a, remain unchanged for power uprate conditions, since they were initially evaluated using a vessel temperature of 552°F which remains bounding.

Note that according to the special stress rules for stress corrosion mitigation, environmental fatigue is considered for only carbon steel and not for SS clad.

2.2 Thermal Stress Calculation Methodology

Each stress cycle combination generally includes a cool-down and warm-up transient. The resulting thermal stresses are due to both the initial temperature and temperature range of each transient. Transient thermal stresses have essentially two components:

$$\sigma_{thermal} = \sigma_{SS} + \sigma_{Shock}(\Delta T) \qquad [1]$$

where,

σ_{SS} = Initial steady state thermal stress (prior to a ramp in temperature)

$\sigma_{Shock}(\Delta T)$ = Incremental thermal shock stress (relative to SS stress)

The initial steady-state (SS) component is independent of the transient temperature change and, if not defined, can be easily calculated from any other specified steady-state condition. For NMP-2, the Zeroload condition (i.e. zero thermal stress) has been specified relative to 70°F. Alternately, the increment thermal shock stress is a function of the transient temperature range, regardless of the initial condition. Thus, since each stress component is proportional to a temperature difference, the stresses can be normalized by some characteristic *influence factor*, α, as follows,

$$\sigma_{SS} = \alpha_{SS}(\Delta T_{SS}) = \alpha_{SS}(T_{SS} - 70) \qquad [2]$$

$$\sigma_{Shock} = \alpha_{Shock}(\Delta T_{ramp}) = \alpha_{SS}(T_{final} - T_{SS}) \qquad [3]$$

where,

T_{SS} = initial SS temperature of transient

T_{final} = final temperature of transient

Thus,

$$\alpha_{SS} = [\sigma_{SS} / \Delta T_{SS}] = \text{constant SS influence factor} \qquad [4]$$

$$\alpha_{Shock} = [\sigma_{Shock} / \Delta T_{ramp}] = \text{thermal shock influence factor} \qquad [5]$$

For any specific nozzle location and time, the steady state influence factor is constant regardless of nozzle temperature; whereas, the thermal shock influence factor is dependent on the transient temperature profile and transient time. Separate influence factors should be calculated for membrane and bending (M+B) stresses and membrane, bending and peak (M+B+P) stresses.

Influence factors were calculated based on the original stress report (Reference 4) transient descriptions using equations [4] and [5]. Power uprate thermal stresses may then be easily calculated using the influence factors and revised power uprate ΔT_{SS} and ΔT_{ramp} values using equations [2] and [3] above.

2.3 Alternating Stress Calculation

Stress ranges were originally computed based upon the worst permutation or combinations of minimum and maximum mechanical and dynamic load types. However, since mechanical and dynamic loads are not affected significantly by power uprate, a simplified approach will be used here to recalculate stress ranges, peak stress ranges, and alternating stress intensities. Only incremental increases in alternating stress intensities due to changes in thermal stress as a result of power uprate will be computed and added to the previous alternating stress intensity, S_{alt}, as follows:

$$S_{alt,\,PU} = S_{alt,O} + \Delta S_{alt,\,PU} \qquad [6]$$

For each transient, a "delta" thermal stress is computed between original and uprate conditions. "Delta" peak thermal stresses, ΔS_P , and "delta" alternating stresses, ΔS_{alt}, are then calculated:

$$\Delta S_{alt} = 1/2\ (E_c\ /\ E_a)\ K_e\ \Delta S_P \qquad [7]$$

For the transient cases considered here, the strain concentration factors, K_e, were assumed to be equal to 1.0. This was a reasonable assumption since in most cases the S_{alt} values were on the order of $3S_m$ or less; the S_n values were expected to be less than S_{alt}. However, for larger S_{alt} values ($>3S_m$), S_n values and K_e factors were recalculated per the elastic-plastic methods of NB-3228.3 of the ASME Code (Reference 7). Per NB 3228.3 of the ASME Code (Reference 7), S_n may exceed $3S_m$ only if the P+Q stress range (i.e. S_n) minus thermal bending is $\leq 3S_m$.

2.4 Fatigue Usage Factor Calculation

The allowable number of cycles, N_{allow}, was computed per Section III of the ASME Code (Reference 8) using the fatigue table values and interpolation formula provided. The incremental usage for each stress cycle combination was then computed,

$$\Delta U_{PU} = N_{actual} / N_{allow} \qquad [8]$$

The incremental usage for the selected combinations is summed for both U_O and U_{PU}. The net difference between the original and power uprate incremental usages considered in this analysis is represented as $\Sigma(\Delta U)$. This value was added to the previous (pre-uprate) reported usage to derive the new cumulative usage factor for power uprate conditions.

2.5 Power Uprate Maximum P+Q Stress Intensity Range, S_n

The maximum stress intensity range (with thermal bending removed) reported after new loads evaluations were considered was 53.1 ksi (corresponding to S0.0+Dyn/Zeroload cycle combination). Thermal membrane stresses were generally small and the increase in membrane stress due to power uprate only resulted in an increase of 0.067 ksi in the P+Q minus thermal bending stress range. Thus the new maximum P+Q minus thermal bending stress range is 53.2 ksi $< 3S_m = 54.3$ ksi, and thus satisfies ASME Code criteria (Reference 8).

2.6 Thermal Stress Ratchet Requirements

Additional requirements for thermal stress ratcheting (NB-3222.5, Reference 7) in the carbon steel base metal were also met since the peak pressure, yield strength, and maximum P+Q stress of the previous analysis (Reference 4) remain bounding. Per Reference 4, the allowable thermal stress range is 74.5 ksi (unaffected by power uprate), which is still well above the power uprate maximum S_n value of 60.8 ksi.

2.7 Results

For the critical SS clad location, the cumulative usage actually decreases from 0.9503 to 0.916 (Table 1). The usage decreases because there was some conservatism included in estimating N_{allow} in the original analysis. As a result of providing more accurate values for allowable stress cycles, the overall usage decreased from the original value which more than offsets the small increase in the alternating stresses.

The total fatigue usage for the carbon steel location also decreased from 0.968 to 0.965 (Table 2). Once again, minor increases in alternating stresses are more than offset by using more accurate values for N_{allow}.

The P+Q (i.e. S_n) stresses with thermal bending removed were scaled in a similar manner based upon the information given for the Zeroload/S.0+Dyn transient. None of the P+Q stresses with thermal bending removed exceeded the $3S_m$ limit. Furthermore, all thermal stress ratcheting requirements for the base metal were satisfied.

Table 1

FW Nozzle Safe End, Thermal Sleeve, Primary Seal, Element #374 (Reference 4)						
Transient	$S_{alt,O}$	$S_{alt,PU}$	$N_{allow,PU}$	N_{actual}	ΔU_O	ΔU_{PU}
HS 4.70 LOFP 2.02	145.1	145.1	412	56	0.1490	0.1359
FHB 33.0 HS 0.45	71.5	72.4	6095	330	0.0750	0.0541
					0.224	0.190
				$\Sigma(\Delta U)$ =	-0.034	
				$U_{CUM,O}$ =	0.9503	
				$U_{CUM,PU}$ =	**0.916**	

Table 2

FW Nozzle, Downstream of Thermal Sleeve, Element #228 (Reference 4)						
Transient	$S_{alt,O}$	$S_{alt,PU}$	$N_{allow, PU}$	N_{actual}	ΔU_O	ΔU_{PU}
TR 0 275 FHB 33.0	59.73	60.23	2427	260	0.1028	0.1071
FHB 33.0 LOFP 2.02	59.54	60.04	2451	56	0.0220	0.0229
Zeroload WR 50.0	45.38	45.82	5695	497	0.0940	0.0873
WR 50.0 DESHYDRO	40.51	40.95	7941	820	0.1055	0.1033
FHB 13.8 S 1.0	40.46	41.00	7982	228	0.0292	0.0286
S 0.0+Dyn Zeroload	73.5	73.85	1365	10	0.0072	0.0073
S 0.25 +Dyn Zeroload	48.54	48.83	4734	220	0.0456	0.0465
S 0.0 - Dyn Zeroload	34.98	35.26	12902	10	0.0008	0.0008
					0.4071	0.4038
				$\Sigma(\Delta U)$ =	-0.0033	
				$U_{CUM,O}$ =	0.968	
				$U_{CUM,PU}$ =	0.965	

3.0 SUMMARY AND CONCLUSIONS

The limiting location for the FW nozzle is still the low carbon steel location (element #228). The cumulative fatigue usage, including environmental fatigue damage and additional hydrodynamic pool loads, is 0.965 and is below the allowable value of 1.0. The cumulative fatigue usage for the SS clad location is 0.916, which is also below the allowable limit of 1.0. Therefore, the FW nozzle satisfies fatigue design requirements for the new power uprate operating conditions.

Power uprate has no significant effect on fatigue usage of the steam outlet nozzle, and the original usage factor of 0.54 (Reference 9) remains applicable.

4.0 REFERENCES

1. GE Document #25A5000, Revision 0, Reactor Vessel-Power Uprate, Certified Design Specification, NMP-2, GE-NE, San Jose, Ca., June 1991.

2. GE Document # 23A6942, Revision 0, NMP-2 Heat Balances for Power Uprate- 100% Power, GE-NE, San Jose.

3. GE Document #762E673, Revision 2, NMP-2 Reactor Vessel Thermal Cycles, GE-NE, San Jose Ca.

4. VPF #6038-81-1, "Special Analysis for New Loads Replaceable Sparger Type Feedwater Nozzle", CBI Nuclear Co., February 1981.

5. F.E. Cooke, "Evaluation of Carbon Steel Environmental Design Rules", GE-NE, San Jose, Ca., June 1981 (DRF B13-00985).

6. GE Document #21A477AZ, Revision 0, Purchase Specification Data Sheet, GE-NE, San Jose.

7. NEDC-31145, NMP-2 New Loads Assessment, Lattin & Patel, GE San Jose, 1986.

8. ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, with Addenda to and including Winter 1972.

9. VPF 3516-51-2, "251 BWR Vessel Thermal Analysis, Steam Outlet Nozzle", CBI Nuclear Co., May 5, 1976.