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Waterford 3

W3F1-2001-0119
A4.05
PR

December 11, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

SUBJECT: Waterford Steam Electric Station, Unit 3
Docket No. 50-382
Supplement to License Basis Request
License Basis Change Regarding GL-96-06 Over-Pressurization of Containment Penetrations Request for Additional Information

Gentlemen:

In accordance with 10CFR50.90, Entergy Operations, Inc. (Entergy) submitted by letter W3F1-2001-0061 dated July 23, 2001, a request for a deviation to the Waterford 3 licensing basis commitment to comply with ASME Section III Code, Class 2 design provisions for the containment penetration piping, as described in the Final Safety Analysis Report (FSAR). On December 4, 2001, Entergy and members of the NRC staff held a call to discuss issues related to the Probabilistic Risk Assessment (PRA) analysis contained in the submittal. As a result of the call, a formal request was made to provide a summary description of the calculation that supports the PRA discussion contained in the original submittal. Entergy's response is contained in Attachment 1.

There are no technical changes proposed. The original no significant hazards considerations included in the original submittal dated July 23, 2001 is not affected by any information contained in this supplemental letter. There are no new commitments contained in this letter.

Should you have any questions or comments concerning this response, please contact Ron Williams at (504) 739-6255.

A072

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 11, 2001.

Very truly yours,

Alan J Harris

A.J. Harris
Director, Nuclear Safety Assurance
Waterford 3

AJH/RLW/cbh

Attachments: 1. Summary of LERF Impact Due to The Postulated Overpressurization Failure of Seven Containment Piping Penetrations GL 96-06

cc: E.W. Merschoff, NRC Region IV
N. Kalyanam, NRC-NRR
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NRC Resident Inspectors Office
Louisiana DEQ/Surveillance Division
American Nuclear Insurers

Attachment 1

To

W3F1-2001-0119

Summary of LERF Impact Due to
The Postulated Overpressurization Failure of
Seven Containment Piping Penetrations
GL 96-06

SUMMARY OF LERF IMPACT DUE TO THE POSTULATED OVERPRESSURIZATION FAILURE OF SEVEN CONTAINMENT PIPING PENETRATIONS GL 96-06

The following summarizes the calculation that determined the impact on the Large Early Release Frequency (LERF) at Waterford 3 from selected penetrations due to the phenomena described in GL 96-06. The containment penetration lines evaluated were the Steam Generator Sample Lines, Steam generator Blowdown Lines, RCS Hot Leg Sample Line, Pressurizer Surge Line Sample Line and the Pressurizer Steam Space Sample Line.

Method of Analysis

The cross-sectional areas of the identified pipes were calculated to determine if a break in a particular pipe would result in a LERF. A LERF would result from hole in containment equivalent to at least a 2" diameter hole. Only the steam generator blowdown lines are greater than 2" in diameter. Therefore, only these lines would contribute to the LERF.

Three distinct probabilistic parameters were required in order to determine the sensitivity of the LERF on the identified penetration piping and failure mechanism. The first is the Core Damage Frequency (CDF) due to a Large Break LOCA, Main Steam Line Break, or a Feedwater Line Break while the plant is in MODE 4. Only these accidents provide the large energy release in containment required to heat the trapped water in the subject pipes to potentially challenge the pipe integrity. The second is the probability that the plant is in Mode 4. The total time the plant was in Mode 4, which consisted of both heat-up and cooldown, was used to conservatively bound this duration, as opposed to limiting the time to only the plant heat-up window of vulnerability. The third is the failure probability for a pipe with a diameter of at least 2" at the pressure calculated for the hypothesized scenarios.

Waterford 3 does not have a Mode 4 model. The CDF due to a Large Break LOCA, Main Steam Line Break, or a Feedwater Line Break with the plant in Mode 4 was assumed to be similar to that for Mode 1 operation. The assumption should be bounding because of 1) the lower pressures involved in all the pressurized systems in mode 4 should reduce the initiator frequencies and 2) the lower initial and decay heat terms should allow longer response times and improved success paths.

The level 1 model was quantified with all the other initiators except the Large Break LOCA, Main Steam Line Break, or a Feedwater Line Break set to 0. This quantification provided an estimate of CDF with the plant in Mode 4 due to the Large Break LOCA, Main Steam Line Break, or a Feedwater Line Break.

The failure probability for a pipe at a given pressure was estimated by a log-normal distribution. The fitting parameters were determined using the following assumptions. It was assumed that the probability of failure equals 0.001 when the hoop stress reaches the yield stress for the pipe material and the probability of failure equals 0.999 when the hoop stress reaches the ultimate stress. The limiting hoop stresses used were conservative, bounding values. The stresses for the steam generator blowdown pipe are listed below.

Steam Generator Blowdown Pipe Properties and Bounding Hoop Stress

Pipe Property	Stress, ksi
Yield Stress	31.36
Ultimate Stress	60
Calculated Bounding Hoop Stress	37.033

Calculation

CDF due to Large Break LOCA, Main Steam Line Break, or Feedwater Line Break:

CDF = 7.28×10^{-7} per year, from the Level 1 PSA.

Probability of the Plant in Mode 4:

Total Plant Time in Mode 4 Data (1992 through 2000):

Year	Total (hours)	mode 4 (hours)
1992	8784	75.4
1993	8760	0
1994	8760	79.7
1995	8760	123.1
1996	8784	23.7
1997	8760	137.7
1998	8760	159.6
1999	8760	137.8
2000	8784	111.8

Total	78912	848.8
Average	8768	94.3

Probability = Time in Mode 4 / Total Time

Probability = 848.8 hr / 78912 hr = 0.01076

Pipe Over-Pressure Failure Probabilities:

Summary of Line Failure Probabilities (lines > 2" diameter)

Line Description	Failure Probability
SG 2 Blowdown Line	6.60E-02
SG 1 Blowdown Line	6.60E-02

Reference: DRAFT NUREG/CR 5745, "Assessment of ISLOCA Risks – Methodology and Application Combustion Engineering Plant"

The reduction in LERF possible from the addition of over-pressure relief valves to the piping evaluated in this calculation (the only lines that would result in at least a 2" diameter opening are the two steam generator blowdown lines).

ΔLERF = (CDF for identified initiators) * (Probability in Mode 4) * (Pipe failure probability)
$= 7.28\text{x}10^{-7} * 0.011 * (2*0.066) = 1.0\text{x}10^{-9}$

Conclusion

The conservatively calculated impact (reduction) in LERF resulting from the mitigation of the postulated thermal expansion overpressurization was on the order of $1\text{x}10^{-9}$. This negligible impact in LERF does not justify the installation of relief valves or other preemptive measures. The Waterford 3 baseline LERF is on the order of $1.8\text{x}10^{-6}$ per year. The ΔLERF value remains well below the very small change of $1\text{x}10^{-7}$ ΔLERF given in Regulatory Guide 1.174 for a LERF of less than $1\text{x}10^{-5}$. The CDF is unchanged by the potential for overpressurization failure of the identified piping.