FRAMATOME ANP

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Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Response to RAI on BAW-2374, Rev. 1

Ref.: 1. BAW-2374 Revision 1, "Risk-Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," Framatome ANP, March 2001.

The NRC sent a draft request for information to Framatome ANP on topical report BAW-2374, Rev. 1 (see Reference), on October 17, 2001. The response to this request is enclosed and is being submitted on behalf of the B&W Owners Group. The response demonstrates that the frequency of a simultaneous reactor coolant system upper hot leg, large-bore pipe break and a secondary side failure due to a seismic event is small. Further, it is concluded that the risk from a seismically-induced large-bore pipe break that could cause a tube rupture in the once through steam generator due to thermal loads is also small.

Framatome ANP understands that the NRC plans to address other matters identified during its review of this topical report. We are especially anxious to resolve any outstanding concerns as soon as possible so this review can be successfully concluded in the near future.

Very truly yours,

Gerald S Holm

James. F. Mallay, Director
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/lmk

Enclosure

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REQUEST FOR ADDITIONAL INFORMATION

BAW-2374, REV. 1, RISK-INFORMED ASSESSMENT

OF OTSG THERMAL LOADS DURING HOT LEG LOCA

Question: The severe accident sequence of concern involves the gross failure of a hot leg pipe in the "candy cane" region (creating a large LOCA), the failure of the steam system pressure boundary in the steam generator on the same reactor coolant system loop, and success of the emergency core cooling system injection for about 15 minutes (leading to overcooling and over stressing the steam generator tubes). CDF and LERF values for this sequence were estimated in the topical report to be <8E-10/reactor year and <4E-11/reactor-year, respectively. However, the estimates were based on the assumption that the pressure boundary failures in the reactor coolant system and steam system were independent. It did not address the probability that both pressure boundary failures could be caused by a seismic event.

Please demonstrate with reasonable assurance that, when seismic initiating events are considered, the CDF and LERF values for this sequence meet the numerical guidance values in RG 1.174 for acceptably small increases.

Response: This response is based on the seismic PRA for TMI-1 since this site has the most severe seismic hazard of any of the operating B&W Owners Group plants (References 1 and 2). Therefore, use of the TMI-1 seismic PRA (which is taken from its IPEEE, Reference 3, for which an SER was issued, Reference 4) is bounding for all other B&WOG units.

The TMI-1 IPEEE uses seismic hazard curves derived from the site seismic hazard study performed by the Electric Power Research Institute (Reference 2). A sensitivity analysis was also performed using the revised seismic hazard curves from Lawrence Livermore National Lab (Reference 1). The initiating event frequencies used in the TMI-1 IPEEE correspond to discrete peak ground accelerations of 0.052g to 0.2g, 0.2g to 0.3g, 0.3g to 0.5g, and 0.5g to 1.0g. The seismic hazard curves in References 1 and 2 indicate that the upper bound ground acceleration at Three Mile Island is 1.0g.

The TMI-1 IPEEE includes seismic fragility (or failure vulnerability) analyses that were conducted by EQE International for TMI-1 structures and components (References 5 and 6). These fragilities and the associated EQE reports were included in the NRC review of the TMI-1 IPEEE (Reference 4). Component and structural vulnerabilities were examined for the indicated seismic accelerations.

The component fragility evaluation (Reference 5) performed for TMI-1 included the major reactor coolant system (RCS) and balance-of-plant (BOP) components. The fragility evaluation included screening components with high generic seismic capability using the EPRI methodology for assessing seismic margin (Reference7). All of the RCS loop piping and pressure boundary components were assessed to be seismically rugged. In addition, the OTSG and pressure boundary components of the main steam system (e.g., safety valves) were also determined to be seismically rugged. These conclusions are generally applicable to all of the B&WOG plants due to similar seismic ruggedness of major components. Other analyses

(Reference 8) also support the conclusion that seismic stresses are negligible contributors to direct breaks of RCS large-bore pipe. The same report concludes that indirect breaks (i.e., breaks due to building or support failures) are a more plausible mode of failure for large-bore piping.

The civil structure fragility evaluation (Reference 6) performed for TMI-1 indicates that the probability of a seismically induced large-bore pipe break is controlled by structural failure involving movement of the primary and secondary shield walls. RCS pressure boundary failure is assumed to occur with a one-inch uplift of these walls, which corresponds to a median peak ground acceleration of 1.2g. Due to the probability distribution associated with the structural fragility curve uncertainty, this failure mode may occur with peak ground accelerations as low as 0.58g. The probability of this failure mode occurring is negligible below 0.58g due to high confidence of low probability of failure limits. Therefore, only the highest amplitude seismic initiating event (0.5g to 1.0g) contributes to this failure mode. To obtain the conditional probability of RCS failure due to the structural fragility, the fragility curves are convoluted with the seismic hazard curves, which yields a conditional RCS failure probability of 3.98×10^{-2}. The RISKMAN computer software, which uses a Monte Carlo simulation to perform the convolution of seismic hazard curves and fragility curves, was used to generate this value. Use of this conditional large-bore pipe break probability is conservative for the scenario of large-bore pipe break induced high OTSG tube thermal loads, because the location of the seismically induced break may be in the lower hot leg rather than the upper hot leg.

Based on the TMI-1 fragility analyses, the high amplitude seismic event (0.5g-1.0g) is the only one that leads to a large-bore pipe break. The seismic initiating event frequency for the 0.5g to 1.0g seismic event is 3.74×10^{-6}/year based upon the EPRI seismic hazard curves (Reference 2). Since the conditional probability of seismically induced RCS large-bore pipe break given the high amplitude seismic event is 3.98×10^{-2}, this yields a frequency of 1.49×10^{-7}/year for a seismically induced large-bore pipe break at TMI-1.

The TMI-1 IPEEE does not address whether building failures can cause failure of secondary side piping coincident with the seismically induced RCS large-bore pipe break. This was not relevant in the TMI-1 IPEEE because the conditional core damage probability for the highest amplitude seismic initiating event is approximately 1.0 due to fragilities of other (i.e., non-pressure boundary) components. However, the frequency of a seismically induced large-bore RCS pipe break, 1.49×10^{-7}/year, bounds the frequency of a seismically induced large-bore RCS pipe break with coincident secondary-side failure. Another B&WOG PRA that includes simultaneous, seismically-induced failure of both the RCS and BOP piping is the Oconee PRA (Reference 9), which assigns it a frequency of 5.8×10^{-8}/year, and which further demonstrates that the TMI-1 frequency is bounding.

However, it is very conservative to use the seismically induced large-bore pipe break frequency for the scenario of high thermal loads on an OTSG tube. The high thermal loads discussed in BAW-2374 require successful emergency core cooling system injection following the RCS upper hot leg pipe break. A seismic event of this magnitude would probably cause other failures (e.g., borated water storage tank and certain electric power systems in the TMI-1 IPEEE) that would preclude overcooling of the OTSG tubes.

Therefore, it is concluded that the frequency of a coincident large-bore pipe break and secondary side failure due to a seismic event is very small. Furthermore, the conclusions of BAW-2374 are not changed by consideration of the frequency of seismic events.

References

1) NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," Lawrence Livermore National Laboratory, 1994.

2) EPRI NP-6395-D, "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," Electric Power Research Institute, April 1989.

3) "Three Mile Island 1 Nuclear Station Individual Plant Examination for External Events," GPU Nuclear Corporation, December 1994. Submitted to USNRC by letter dated December 29, 1994.

4) "Safety Evaluation Report of the Individual Plant Examination of External Events (IPEEE) Submittal on Three Mile Island Nuclear Station, Unit 1," attached to letter from Timothy G. Colburn (USNRC) to James W. Langenbach (GPUN), dated July 9, 1999.

5) "Development of Equipment Seismic Fragilities for TMI-1," EQE International, September 1994.

6) "Seismic Fragilities of Civil Structures at Three Mile Island Unit 1," EQE International, April 1994.

7) EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)," Electric Power Research Institute, August 1991.

8) NUREG/CR-4290, "Probability of Pipe Failure in the Reactor Coolant Loops of Babcock & Wilcox PWR Plants," Lawrence Livermore National Laboratory, May 1986.

9) Oconee Nuclear Station PRA, Revision 2, Duke Energy Corporation, December 1996.