CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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SUBJECT: Forwards questions re SG tube rupture releases,probability of power recovery,containment isolation failure & penetration seal failure.

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C A T E G O R Y 1 D O C U M E N T

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Question 6: Steam Generator Tube Rupture (SGTR) Releases

The probability of the steam generator (S/G) atmospheric relief valves (ARVs) failing to close is determined in the Level 1 analysis. There are essentially 3 cases in which the ARV on the ruptured S/G can fail to close. The first is if feedwater flow to the ruptured S/G is not isolated such that the S/G rapidly overfills and the ARV relieves water. In this case the ARV is assumed to stick open such that rapid cooldown to RHR shutoff head is required. The second case is if isolation is successful but the ARV on the intact S/G fails to open. In this case, operators are instructed to use the ARV on the ruptured S/G to cooldown the RCS. In this case the probability of the ARV on the ruptured S/G failing open is determined by the Level 1 data analysis portion of the PSA using plant specific data updated with generic industry data. The failure probability for the ARV to reclose following a steam release is 8.53E-04. The third case is one in which the operators fail to cooldown and depressurize the primary system prior to overfilling the ruptured S/G due to a failure of the PORVs to open. Again, due to a liquid release through the ARV, it is assumed to stick open.

In the Level 2 analysis, no adjustment was made to account for increased failure probability due to harsh conditions. The possibility of debris entrained in high temperature gas being transported from the core through the RCS piping, through the ruptured tube which could potentially be under water, and up through the S/G and its moisture separators (which are designed to remove droplets or particles entrained in gas) was not considered to be credible. It should be noted that following the Level 1 requantification, the contribution to CDF from SGTR sequences dropped from approximately 33% to 16% such that this issue is of significantly less consequence.

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Guy Vissing

Question 7: The Probability of Power Recovery

As stated in the January 15, 1997 response to the RAI, new power recovery curves were developed for the Level 1 resubmittal. These new curves were used to develop power non-recovery probabilities for the Level 2 analysis. The preliminary results for the four cases of interest are shown in the table below. The table shows the time to vessel failure and the time to containment failure, along with the power non-recovery probability associated with each of those times. The large and medium LOCA cases are evaluated separately from the other SBO cases because in the Level 1 analysis large and medium LOCAs coincident with a SBO were assumed to lead directly to core damage and did not transfer into the SBO event tree. The two SBO scenarios take into account whether the TDAFW pump starts and runs (i.e., the first branch in the SBO event tree). Although PDS binning is not yet complete, these non-recovery probabilities will be used in the binning process. Note that the Level 1 event tree for SBO includes the potential for power recovery prior to core damage which will affect the binning process.

	VF Time	VF N.R. Prob	CF Time	CF N.R. Prob
Large LOCA	.9 hrs.	.383	12 hrs.	.065
Medium LOCA	1.6 hrs.	.267	13 hrs.	.090
SBO (no AFW)	4 hrs.	.09	21 hrs.	.267
SBO (AFW for 6 hrs.)	13 hrs.	.024	25 hrs.	.985

Question 9: Containment Isolation Failure

The probability of containment isolation failure was determined by quantifying the containment systems event tree (CSET), which includes a heading for Containment Isolation failure. As previously stated, the five areas in NUREG-1355 are evaluated in detail. Specifically, items 1 through 4 form the basis of the containment isolation fault tree which is quantified (item 5). As in any Level 1 fault tree, the model includes the appropriate top gates (failure of the pathways determined in item 1), all supporting systems (motive force for valves and signals required as determined in items 2 and 3), and plant specific failure rate data and testing and maintenance data (item 4). Section 3.2.1.3 of the original submittal discusses this in more detail.

Preliminary requantification of the Containment System Event Tree (CSET) indicates that the current percentage of non-containment bypass core damage sequences which result in containment isolation failure is 3.0% (down from 5.2% in the original submittal). Of this 3%, approximately 1/3 is a result of the mechanical failure of AOV 371 to close during LOCA sequences where sump recirculation using the RHR system is required. Section 6.16.4.N of the 1/15/97 submittal discusses this failure path in detail (note that there is a typographical error in that paragraph; the phrase "which overflows to the Auxiliary building sump" is inadvertently repeated). Another 1/3 of the 3% is due to the failure of MOV 313 to close due to a loss of DC power on train A. This failure only leads to containment isolation failure if pressure in containment exceeds 85 psig such that the relief valve on the VCT opens creating an open path outside containment, or, if there is a failure of CVCS piping. It was conservatively assumed in the CSET quantification that if containment spray and the containment recirculation fan coolers (CRFC) fail, pressure in containment will exceed 85 psig. The remaining 1/3 of the 3% involves various other random failure combinations.

Question 10. Penetration Seal Failure

NUREG-1037, pg 2-15, states that "piping penetrations and associated piping for the six reference plants are not likely to contribute to containment leakage before reaching the capability pressures". Electical penetration assemblies (EPA) have both inboard and outboard seals which would have to fail in order to fail the penetration. NUREG-1037, pg C-4 states that "If at least one set of the EPA seals and/or sealants are at or below the design temperature, then the potential for leakage is expected to be low." Details of the piping penetrations and electrical penetration assemblies are discussed below in items e and f. Since these penetrations are similar to those described in NUREG-1037, they are not expected to leak significantly. Those penetrations having the greatest potential for leakage include:

- equipment hatch
- personnel hatch
- fuel transfer tube
- purge and vent system isolation valves

a. The 14' equipment hatch is pressure un-seating with double tongue and groove silicone rubber seals. There are 36 swing bolts which are 1 3/4" diameter and have a specified torque of 900 to 1000 ft-lb. The equipment hatch is similar to the Peach Bottom equipment hatch (12' diameter, 24 1-3/4 in. swing bolts with preload torque of 1900 ft-lb) as described in NUREG-1037, Appendix B, page 27. The NUREG calculates an upper bound leak area (assumes no gasket) for the Peach Bottom equipment hatch of 4.15 square inches at 160 psig.

b. The 116" diameter personnel hatches (2) are both pressure seating with double tongue and groove silicone seals. The personnel hatches are similar to the Zion hatch (122" diameter) shown in NUREG-1037, Appendix B, Figure 10. The NUREG calculates an upper bound leak area of 5.36 square inches at 134 psig.

c. The fuel transfer tube is a 24" pipe sealed by a double gasketed blind flange on the containment side and by a gate valve on the spent fuel pool side. The fuel transfer tube penetration is (like Zion and Surry) similar to the one shown in Fig. 12 of Appendix B of NUREG-1037. The NUREG does not include calculations of the leakage for the fuel transfer tube but does state that "for a leak to occur between the FTT and its containment penetration sleeve, the leak must penetrate a bellows on the containment side, the seal plate, and a bellows on the outside of the containment." It is assumed that this penetration would not leak significantly.

d. NUREG-1037 states that "the large-diameter butterfly valves associated with the purge and vent system are considered to have the greatest potential for containment leakage" and that "the main concern is that the non-metallic seals between the valve body and disc will become degraded when subjected to the combination of high pressures and temperatures associated with severe accident conditions." At Ginna, the 48" containment purge ducts (supply and exhaust) have double gasket flanges during normal operation. The mini-purge supply and exhaust valves are 6" XOMOX model 801 Pliaxseal butterfly valves with PEEK/metal back-up seats. This is considered a fire-safe seal in that the PEEK material is the contact point during normal operations but if they were to experience high temperatures that degraded the PEEK, there is a metal back up to maintain the seal. These valves would thus not be considered likely to result in significant leakage.

e. The Ginna piping penetrations are generally embedded sleeves except for the 3 drain lines from Sump B which are embedded pipe (2-8", 1-4"). There are 15-10" and 3-6" flanged sleeves or pipes. There are 8-6" and 13-10" flued head/bellows penetrations. There are 2-6" and 3-10" flued head penetrations. There are 2-24 1/4", 1-14 1/4" and 8-12 1/2" insulated flued head/bellows penetrations. There is 1-24 1/4" insulated flued head/bellows penetration.

f. There are 50 Ginna electrical penetrations which were manufactured by Crouse-Hinds. The critical sealing function for these penetrations is ceramic to metal which, according to NUREG/CR-3234, is an excellent seal design.

The worst case temperature scenarios for Ginna are the station blackouts without power recovery or with power recovered late. Temperatures are seen between 300-375° for 8-10 hours. Worst case pressure sequences are seen when there are no fan coolers or containment sprays available concurrent with core-concrete interaction. These again tend to be the station blackout sequences. Based on the above discussion, RG&E still believes that penetration failure is significantly less important than overpressure failure.