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See Reports

SUBJECT: Provides response to RAI & resubmits level 1 PSA of Ginna Station.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

January 15, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Generic Letter 88-20, Level 1 Probabilistic Safety Assessment (PSA)
Rochester Gas & Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing,

By Reference 1, RG&E submitted a Level 2 PSA in response to Generic Letter (GL) 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities.* Subsequent to that submittal, RG&E identified several inconsistencies and unwarranted conservatisms within the PSA models and determined that a re-assessment of the Level 2 PSA was appropriate (Ref. 2). In the midst of performing this new analysis, the NRC provided RG&E with a request for additional information (RAI) related to the March 15, 1994 submittal (Ref. 3). RG&E responded to this RAI with several letters which ultimately extended RG&E's response until January 15, 1997 (Refs. 4, 5, and 6). Therefore, the purpose of this letter is to provide a response to the RAI and to resubmit a Level 1 PSA of Ginna Station. In addition, an update to all outstanding responses to the remaining GL 88-20 supplements is being provided.

With respect to the RAI documented in Reference 3, Attachment A to this letter contains the response to all NRC questions and comments. The response to these questions is primarily based on Revision 1 to the Ginna Station PSA as contained in Attachment B. This document replaces the Level 1 PSA provided in Reference 1 in its entirety. The Level 2 and flooding portions of the PSA will be updated and re-submitted to the NRC in accordance with the schedule listed below.

GL 88-20 is comprised of multiple supplements related to severe accident vulnerabilities. A description of each supplement and RG&E's understanding of its status is provided below:

a. *Generic Letter 88-20* - This requested an individual plant examination (IPE) of each plant that was basically comprised of an evaluation of core damage frequency (Level 1 PSA), containment performance (Level 2 PSA), and a flooding evaluation. RG&E originally responded to this request in Reference 1; however, Attachment B to this letter supersedes the previous Level 1 PSA. A new Level 2 PSA will be submitted to the NRC by May 1, 1997 (Ref. 6). The new flooding analysis will be presented with the rest of the external events by September 30, 1997 (see Supplement 4 below).

b. *Supplement 1* - This announced the availability of NUREG-1335 which provided reporting guidelines with respect to the IPE as discussed in a. above. This NUREG was used in the preparation of the PSA such that no further action is required.

c. *Supplement 2* - This requested consideration of severe accident management (SAM) strategies in the IPE process. Subsequent to this document, the nuclear industry developed generic SAM guidance and committed to implement SAM capabilities by December 31, 1998. RG&E responded to the supplement in Reference 7 and stated that we would meet this deadline; however, we also identified that RG&E was "actively pursuing completion of these tasks in 1997." Due to the delays in completing the Level 2 PSA which provides an input into the plant-specific SAM guidelines, RG&E does not expect to complete these activities in 1997. However, the industry deadline of December 31, 1998 will still be met.

d. *Supplement 3* - This announced the completion of the NRC's containment performance improvement program. Relevant information will be incorporated into the Level 2 PSA and SAM guidelines such that no further action is required on this specific supplement.

e. *Supplement 4* - This requested an IPE for external events (IPEEE) and provided NUREG-1407 for reporting information. The following provides a status of each required external event evaluation:

1. Seismic events - This is being addressed as part of the closeout of GL 87-02, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46.* A final report on these topics will be submitted to the NRC by February 1, 1997 (Reference 8).

2. Internal fires - RG&E will submit a fire analysis by September 30, 1997 (Ref. 9). Per Reference 10, RG&E plans to use the guidance of NUREG-1407, Section 14.1 and the EPRI FIVE propagation and damage assessment models for screening purposes, and the FIVE walkdown procedures to address Fire Risk Scoping Study (FRSS) issues.

3. High winds and tornadoes - In Reference 11, RG&E stated Ginna Station was designed to withstand a 1E-05/yr tornado with structures capable of withstanding a 1E-06/yr tornado. This was considered to meet Section 5.2.4 of NUREG-1407. The NRC responded in Reference 12 that since RG&E was using "the screening approach described in NUREG-1407 for high winds ...", this was acceptable. Consequently, RG&E considers this external evaluation complete.

4. External floods - In Reference 11, RG&E stated that a Probable Maximum Flood with a recurrence interval of 5E-04/yr was used as the basis for the Systematic Evaluation Program (SEP) of Ginna Station. Necessary plant modifications were implemented to meet this flooding event such that Section 5.3 of NUREG-1407 was considered to have been met. The NRC responded in Reference 12 that since RG&E was using "the screening approach described in NUREG-1407 for high winds, floods ...", this was acceptable. Consequently, RG&E considers this external event evaluation complete.

5. Transportation and nearby facility accidents - In Reference 11, RG&E stated that the NRC safety evaluation for SEP Topic II-1.C concluded that the Standard Review Plan was met for these issues. Therefore, RG&E considered that Section 5.2.3 of NUREG-1407 was met. The NRC responded in Reference 12 that since RG&E was using "the screening approach described in NUREG-1407 for high winds, floods, and transportation and nearby facility accidents," this was acceptable. Consequently, RG&E considers this external event evaluation complete.

f. *Supplement 5* - This announced that seismic review requirements as outlined in NUREG-1407 could be downgraded based on NRC re-assessment of potential seismic events. RG&E responded in Reference 13 that RG&E had already downgraded our seismic reviews from those in NUREG-1407. The seismic evaluation will be provided by February 1, 1997 as described in e.1 above.

In addition to GL 88-20 and its supplements, the NRC requested that RG&E specifically evaluate two other issues in the PSA as follows:

a. During the Safety System Functional Inspection of the Residual Heat Removal system in 1989, the NRC identified a potential scenario where the discharge line for Service Water cooling to the diesel generators could crimp during a during a design basis seismic event (Ref. 14). RG&E responded to this beyond design basis concern by agreeing to evaluate the scenario in the PSA (Ref. 15). This evaluation is provided in Section 9.4.1 of Attachment B to this letter.

b. In Reference 16, the NRC identified a potential concern related to hydrogen storage adjacent to the diesel generator buildings and requested that RG&E evaluate this issue as part of Supplement 4 to GL 88-20. This evaluation will be provided by September 30, 1997 consistent with the discussion above.

Please contact George Wrobel, Manager of Nuclear Safety and Licensing at (716) 724-8070 if you have further questions.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

References:

1. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Generic Letter 88-20*, dated March 15, 1994.

2. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Level 2 Probabilistic Risk Assessment*, dated March 10, 1995.

3. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, *Ginna Probabilistic Risk Assessment Project Report of March 15, 1994, to the NRC in Response to Generic Letter 88-20; Request for Additional Information (TAC No. M74414)*, dated February 23, 1995.

4. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Generic Letter 88-20, Response to Request for Additional Information (TAC No. M74414)*, dated April 19, 1996.

5. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Generic Letter 88-20, Updated Response to Request for Additional Information (TAC No. M74414)*, dated July 31, 1996.

6. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Generic Letter 88-20, Updated Response to Request for Additional Information (TAC No. M74414)*, dated November 15, 1996.

7. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Severe Accident Management*, dated March 24, 1995.

8. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, *Generic Letter 87-02, Supplement 1 (TAC No. M69449) and Generic Letter 88-20, Supplement 4 (TAC No. M83624)*, dated November 15, 1996.

9. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, *Generic Letter 87-02, Supplement 1 (TAC No. M69449) and Generic Letter 88-20, Supplement 4 (TAC No. M83624)*, dated August 26, 1996.

10. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Individual Plant Examination of External Events (IPEEE), 180 Day Response to Generic Letter 88-20, Supplement 4*, dated December 26, 1991.

11. Letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, *Individual Plant Examinations for External Events (IPEEE)*, dated November 3, 1992.

12. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, *R.E. Ginna - Review of Response to Generic Letter 88-20, Supplement 4 - Individual Plant Examinations for External Events (TAC No. M83624)*, dated August 11, 1993.

13. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, *Response to Generic Letter 88-20, Supplement 5*, dated November 7, 1995.

14. Letter from M.W. Hodges, NRC, to R.C. Mecredy, RG&E, *NRC Safety System Functional Inspection Team Report No. 50-244/89-81*, dated May 9, 1990.

15. Letter from R.C. Mecredy, RG&E, to T.T. Martin, NRC, *Response to Inspection Report 50-244/89-81, Safety System Functional Inspection - RHR System*, dated June 8, 1990.

16. Letter from A.R. Johnson, NRC, to R.C. Mecredy, *Risk Associated With Hydrogen Storage Facility at Ginna*, June 24, 1993.

Attachments

MDF\897

xc: U.S. Nuclear Regulatory Commission
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Ginna Senior Resident Inspector

Attachment A

Response to February 23, 1996 RAI

The following pages provide a response to the questions documented in the February 23, 1996 request for additional information (RAI) letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E. This RAI was provided with respect to the Ginna Station Level 2 PSA submitted by RG&E on March 15, 1994. However, the Level 1 PSA has been revised and is being resubmitted to the NRC as part of this letter (see Attachment B). In addition, the Level 2 PSA will be revised and resubmitted by May 1, 1997. Therefore, the NRC questions and comments do not match up with the new page numbers and sections, and in some cases, may no longer be applicable. As such, RG&E's response to the RAI consists of the following:

a. The NRC question is repeated (in italics) followed by a RG&E response.

b. For the Level 1 PSA and human error questions (RAI Enclosure 1 and 2, respectively), the RG&E response consists of a reference to a specific section within the revised Level 1 PSA which addresses the subject. The revised Level 1 PSA was specifically written to include relevant discussion of those topics which the NRC raised issue with in the February 23, 1996 RAI. The only exception to this is with respect to flooding (see Enclosure 1 Question 10 and Enclosure 2 Question 14) which will be provided by September 30, 1997 when the external event PSA is submitted.

c. For the Level 2 PSA questions (RAI Enclosure 3), the RG&E response is with respect to the March 15, 1994 submittal only. This response indicates whether RG&E plans to make a change to the approach used in the new Level 2 PSA or continue to use the original technique. The results of using any new approach will be provided when the revised Level 2 PSA is submitted to the NRC (see response to Enclosure 3 Questions 5, 6, 7, 9, 10, and 17, and Enclosure 2 Question 15).

ENCLOSURE 1

LEVEL I QUESTIONS

1. *The following question concerns the Bayesian updating scheme used in the individual plant examination (IPE):*

   (a) *In the modeling of loss of offsite power (LOOP) events, your Bayesian updating scheme leads to an order of magnitude reduction in the LOOP frequency over the generic nuclear power plant LOOP frequency. The total LOOP frequency calculated is about 3.5E3/yr. Considering the relatively frequent weather phenomena (e.g., ice storms) in the Ginna area that could conceivably lead to a loss of the whole grid, this number seems very low.*

   It is also surprising that Bayesian updating could lead to such a large reduction in the initiating event frequency. The discussion in the submittal indicates that you are attempting to update a prior distribution with evidence of zero failures in 9 years. Please note that such evidence is not very strong since the prior mean of occurrence rate is 0.027/yr, or one failure in about 50 years. For cases of weak evidence, the prior distribution should dominate the behavior of the posterior distribution. However, this is not the case for the application results presented in the submittal.

   It would appear that you have replaced the Lognormal prior with an equivalent Gamma prior, which is conjugated with a Poisson likelihood under the Bayes algorithm. The method used for translating Lognormal to Gamma is the method of matching the first two moments (mean and variance). However, the Gamma distribution is not a good approximation for a Lognormal distribution when the resulting parameters of the Gamma distribution (namely α and β) fall in certain regions, specifically, if the parameter α becomes less than or equal to one. In such a case, the Gamma distribution would not have any maximum, and if used as a prior would heavily weight the low values of the occurrence rate, contrary to the Lognormal distribution. This appears to be the case for your application. It is also important to note that the parameter α is unitless and would not change if an annual occurrence rate or a 100-year occurrence rate is used (in contrast, the β parameter has dimensions of time). Use of the Gamma distribution is not recommended for cases in which the parameter α falls below one. In practice, the Gamma distribution should only be used as an approximation to the Lognormal distribution when α is greater than two. Calculations show that if a Lognormal distribution instead of a Gamma distribution is used, your prior mean of 0.027/yr and error factor of 23.7 result in a posterior mean of 0.015/yr and an error factor of 8.5, a much more modest reduction than that indicated in the

submittal.

For these reasons, and the fact that the updating was done with the relatively weak evidence of no failure in 9 years, the posterior distribution obtained appears to be in error.

Please recalculate your Bayesian update using a Lognormal prior distribution and compare the results with your submittal. Please provide the impact this adjustment in modeling has on the core damage frequency (CDF) and on the important sequences.

See Section 7.3.1.2. The new loss of offsite power frequency is 6.32E-02/ryr.

(b) *Please explain your Bayesian updating treatment of reliability data for other components where conditions similar to the ones found in LOOP treatment existed (i.e., $\alpha<1$ and sparse or zero plant-specific evidence). For example, the following component failure rates may fall into this category: 120-volt ac bus failures, auxiliary feedwater (AFW) and residual heat removal pump failure rates, and safety injection and service water (SW) demand failure rates. Please discuss how the Bayesian updating of these components was done (e.g., how many failures were experienced, details of Bayesian updating calculations, and the final posterior numbers used in the IPE).*

If an adjustment in failure data is necessary, please provide an estimate of the impact on the CDF and important sequences.

Appendix C contains a detailed discussion of the plant-specific data collection effort. Table C-3 provides a listing of the failure data used in the PSA models for which plant-specific data was collected against. The table provides plant-specific failure data ("Plant"), generic data ("Aggregated"), and final value used in the PSA ("Final"). Essentially, if no component failures were observed in a given population, generic data was used. If one or two component failures were observed, Bayesian updating was typically performed. If a significant number of failures were observed (i.e., more than two), only plant-specific data was used.

2. *The following question concerns the modeling of LOOP events:*

(a) *It is not clear how the possibility of a post-trip LOOP was modeled. The submittal states that the conservative assumption is made in that a reactor trip would lead to a LOOP as a result of a grid transient caused by a-loss of the Ginna generating capacity. Does that mean that any initiator (e.g., a loss-of-coolant accident (LOCA), a transient) will also lead to a LOOP and a demand for*

diesel generators? In that case, a loss of SW would lead first to a LOOP, and then to a station blackout (SBO), because SW is used to cool the diesels. Therefore, a loss of SW initiator should have a relatively high conditional core damage probability, which is not supported by the results. Please clarify the treatment of the post-initiator LOOP and provide data (and the bases) for the conditional probability of a LOOP following a reactor trip. If an adjustment in modeling is necessary, please provide the impact on the CDF results and the important core damage sequences.

Section 3.4.2.3 discusses the post-trip LOOP event while Section 7.3.1.2 discusses the data used for these events. In summary, a post-trip LOOP was considered for any initiator (similar to UFSAR Chapter 15 assumptions) since the loss of Ginna Station could lead to sufficient grid instability so as to result in a subsequent LOOP.

(b) *There is no discussion of LOOP and SBO sequences, and no separate event tree is provided, even though a statement is made that these events are treated separately from other transients because of their special nature. Please discuss exactly how you treated SBO and provide the SBO event tree, if available.*

A SBO event tree has been generated as shown in Section 5.2. All SBO scenarios are transferred to the SBO event tree for evaluation. The only exceptions are for medium and large LOCAs and ATWS events which are assumed to directly result in core damage.

(c) *It is not clear how the turbine-driven auxiliary feedwater (TDAFW) pump is used under SBO conditions and how it is modeled. The dependencies of this pump include SW; heating, ventilation, and air conditioning (HVAC); and dc voltage.*

It is stated that a test run of 1 hour and 45 minutes was made to show that the pump can survive a total loss of ac power (i.e., loss of SW cooling to the bearing oil coolers). However, no test was run beyond that time.

Another test showed that the temperatures will reach 145°F in the TDAFW room after a 4-hour loss of HVAC. There is no indication as to what happens beyond the 4 hours, except for a statement that there would be no damage for at least 24 hours, but that for conservatism a 10-hour power recovery is modeled.

There is no indication of the depletion time for the battery supporting TDAFW operation (other than the 1,200 amp-hr capacity given), of how this conclusion was modeled, or what battery supports TDAFW operation (e.g., is it battery IB?). The TDAFW pumps are tested to show operation over a 2-hour period.

Please describe the sequence of events for SBO in a manner similar to that for other initiators, which specifically addresses the operation and modeling (including the time assumed for operation) of the TDAFW pump.

The success criteria used for the SBO event tree (and TDAFW pump) is found in Section 4.2.2.4, SG Cooling, and Appendix B.

(d) *Certain details are not clear about the design and operation of the 125V-dc system. Can the technical support center (TSC) battery support the operation of the safeguards equipment in an SBO if battery 1A or IB failed, and how is this eventuality modeled? Can the IA or 1B battery support the operation of the other division and how is this function modeled? Why is the TSC battery the only one tested (apparently) and tested only for 2 hours? Is this period the assumed running time of the TDAFW pump in an SBO? What are the depletion times for the 1A, IB, and TSC batteries?*

The TSC battery has the capability to tie into DC safeguards trains as shown in Figure 6-8. However, due to the significant amount of effort required to utilize this capability, credit was not assumed in the PSA (or in the SBO mitigation plans per 10 CFR 50.63). With respect to cross-support among Battery A and B, there is the capability for certain loads on a DC train to be supplied by the opposite train during emergency conditions (e.g., diesel generator support). This is an automatic feature incorporated into the PSA.

(e) *It is stated that the TSC battery and/or the standby electric power TSC diesel(s) can be used as a limited backup should the main emergency diesel generators (EDGs) fail. Which loads can these diesels support? Please explain if these diesels are credited in the model, and if so, explain how they were modeled (including operator actions, unavailability, and failure data).*

Credit for the TSC diesel generator was not assumed in the PSA model.

(f) *It seems that the hardware exists for cross-connecting the emergency buses. Is this action proceduralized? If so, is credit taken for it in the IPE, and how was it modeled?*

The capability for cross-connecting the 480 V safeguards buses exists within the emergency plans. However, this action was not deemed critical for the PSA, and as such, was not credited.

(g) *Please explain why the events of January 21, 1985 (LER 85-002), and July 16, 1988 (LER 88-006), were not counted as plant-specific LOOP events in the*

Bayesian updating calculation of LOOP frequency. The former event was an incipient LOOP, in which both EDGs were started and tied to their safeguards buses of low grid frequency caused by extremely cold weather. The latter event was via loss of normal power, including power to all four safeguards buses, such that both EDGs were started and loaded onto the safeguards buses.

See Sections 3.3.4 and 3.3.9 for discussion of these events.

(h) *The EDG fuel oil transfer system is apparently modeled separately from the EDGs. However, the Common Cause Tables (3.3.4-1 and 3.3.4-2) do not seem to show these pumps, even though there has been at least one event involving a common cause failure of these pumps. The event of February 20, 1987 (LER 87-001), involved plugging of the strainers in both pumps as a result of the use of the inappropriate materials to clean the fuel oil tank. Please explain how this event was incorporated in the modeling of the common-cause failure of the EDG fuel transfer system. If the event was not accounted for, please justify the omission.*

CCFs of the diesel fuel oil pumps has been added to the model (see Table 7-3).

3. *This question concerns the treatment of the main feedwater (MFW) system:*

(a) *Please provide a description of the MFW system and how it was modeled (e.g., which components were taken into account?).*

See Section 6.14.

(b) *A statement is made that the MFW pumps cannot be used below 4 percent power. The power level will fall below this value very quickly after shutdown, yet MFW operation seems to be credited in many sequences, and there is no discussion of any timing concerns with respect to the MFW. Please clarify how and when MFW operation was credited in the analysis. If an adjustment in the modeling of MFW is necessary, please provide an estimate of the impact on the results and on the important sequences.*

See Section 6.14 for discussions of how MFW was credited. Note that the MFW system <u>can</u> be used below 4% power (and is typically used initially post-trip). However, due to the difficulties in maintaining SG levels under these low power conditions, the motor-driven AFW pumps are normally used during startup and planned shutdown activities. These difficulties do not prevent use of the MFW system during accident conditions as instructed by the EOPs.

4. *Please provide the bases for using the cutset truncation limit of 5.E-8/yr, which is*

relatively high compared to industry practice, and provide an estimate of the residual. If the residual is significant (e.g., greater than 5 percent of the CDF), please provide an estimate of the impact on the important sequences and on the results.

A new truncation limit of 1.0E-10 was used as discussed in Section 9.3.1.9.

5. *One of the small-small LOCA success paths utilizes a rapid cooldown of the reactor coolant system (RCS) to low pressure injection (LPI) conditions. This process is supported by modular accident analysis program runs, according to the submittal. Are there procedures in place to utilize this option? If not, provide the bases for crediting this action. If available, please provide an estimate of the impact on the results and important sequences if this option was not credited.*

   See Section 4.2.2.3.3, Small-Small LOCA, and Appendix F for discussion of this operator action.

6. *This question concerns the treatment of initiating events:*

   (a) *The small-small LOCA initiating event frequency in the Ginna IPE is significantly smaller than that used in NUREG/CR-4550. The IPE S3 frequency is 7.3E-4 versus the NUREG/CR-4550 frequency-of 1.3E-2. It does not seem to include such events as spurious reactor coolant pump (RCP) seal LOCAS, which are a major contributor to the S3 frequency in NUREG/CR-4550 and other studies.*

   Please provide the bases for excluding RCP seal LOCAs from the S3 frequency and, if available, provide an estimate of the impact on the results and the important sequences if a more traditional small-small LOCA frequency estimate is used.

   Spurious RCP seal LOCAs have been included within the small-small LOCA frequency as discussed in Section 7.3.1.6. The total frequency of small LOCAs due to pipe breaks and spurious RCP seal LOCAs is now 6.6E-03 (note that Ginna Station only has two RCPs).

   (b) *Please explain why the loss of a 4kV-bus is not considered as an initiating event. For example, loss of bus 12A or 12B might cause an initiating event while failing safeguards equipment on that bus.*

   See Sections 3.3.4 and 3.3.9.

   (c) *At least two events at Ginna have involved the possibility of freezing, leading to a loss of SW. This situation could be due to frazil ice buildup on the intake screens.*

In addition to the event discussed in the IPE (LER 83-006) caused by lowering of the voltage to the intake heaters, a more recent event occurred after the IPE submittal. This event involved a zebra mussel buildup on the heaters, again causing frazil ice buildup.

In calculating the frequency of the loss of SW initiator in the IPE, was the possibility of ice buildup accounted for? If so, please summarize how it was taken into consideration. If it was not, please provide an estimate of the impact on the CDF and on important sequences.

See Section 7.3.1.8 for a discussion of the frequency for loss of SW.

(d) *Extremely cold weather could conceivably cause a LOOP because of a high load on the grid (a precursor event occurred in January 1985; see question 2 (g) above), in conjunction with a loss of SW because of ice buildup as discussed in part (c). Loss of SW would, in turn, lead to failure of the diesel generators, thus leading to an SBO. Please discuss how you considered such an initiating event. If this event has not been adequately considered, please provide an estimate of the impact of this event on the CDF and on important sequences.*

The frequency of LOOP was based on industry data between 1980 and 1995 such that extremely cold weather is a consideration. A coincident failure of SW is considered very unlikely due to the fact that frazil ice occurs following rapid changes in wind direction (i.e., from the north). Therefore, the extreme temperatures would have to be followed by a rapid wind change, with a subsequent failure of all RG&E preventative measures.

(e) *Please provide the frequency estimate used for initiation of an anticipated transient without scram (ATWS) in the IPE analysis.*

See Section 7.3.1.11.

7. *This question concerns the data used in the model:*

(a) *In discussing components with relatively high failure rates compared to generic data (e.g., containment spray pumps), a statement is made that the high rate of failure is due to a limited test exposure time, and not necessarily to a plant vulnerability. However, certain failure modes (e.g., sediment buildup) may appear in components that are idle for long periods.*

Please verify that the plant experience for these components with a relatively high failure rate was retained in developing the database.

See Section 7.2.1.3.1 and Appendix C.

(b) *Please provide the final reliability data listings of Appendix E. If not included already in Appendix E, please also provide the generic data used, as well as the plant-specific experience (e.g., number of demands, or number of hours, and number of failures) for each failure mode and each component.*

See Tables 7-1 and Appendix C, Table C-3.

(c) *The common-cause analysis seems to have omitted some potentially important components, which might influence your search for vulnerabilities. The following components were apparently not considered:*

Circuit breakers
Relays (engineered safety features actuation system)
Electrical switchgear
Transmitters
Ventilation fans
Air compressors
Inverters

In addition, common-cause failures within the AFW and standby auxiliary feedwater (SAFW) systems, also involving the turbine driven pump (i.e., driver independent failures), could be postulated.

Please provide the bases for omitting these potential common-cause failures and discuss how you ensured that no vulnerabilities were missed as a result of these omissions.

See Section 7.2.2 and Table 7-3 for detailed discussions related to common cause failures.

(d) *Please show how the possibility of freezing, along with any plant specific data involving freezing, has been accounted for in failure data and common-cause failure data for components that are vulnerable to this phenomenon. Apparently, this phenomenon includes the SAFW system, the EDGs, and the SW system.*

See Sections 3.3.5 and 6.11.

8. *It is not clear from the submittal if plant changes as a result of the SBO were credited in the analysis. Please provide the following information:*

(a) *Discuss whether plant changes (e.g., procedures for load shedding, ac power) made in response to the SBO were credited in the IPE and which plant-specific plant was credited.*

The Ginna Station design with respect to meeting 10CFR50.63 was used in the Ginna Station PSA.

(b) *If available, provide the total impact of these plant changes to the total plant CDF and to the CDF contribution from SBO (i.e., the reduction in total plant CDF and SBO CDF).*

Since the revised PSA began with the current Ginna Station design (i.e., post implementation of any SBO related plant changes) no Δ can be provided.

(c) *If available, provide the impact of each individual plant change on the total plant CDF and the SBO CDF (i.e., the reduction in total plant CDF and SBO CDF).*

Since the revised PSA began with the current Ginna Station design (i.e., post implementation of any SBO related plant changes) no Δ can be provided without significant modeling changes.

(d) *Discuss any other changes to the plant that are separate from those made strictly in response to the SBO rule that nonetheless may reduce the SBO CDF. In addition:*

(i) *Describe whether these changes are implemented or planned.*
(ii) *Indicate whether credit was taken for these changes in the IPE.*
(iii) *If available, discuss the impact of these changes on the SBO CDF.*

All plant changes made as a result of the Ginna Station PSA are discussed in Section 11.1.2.

9. *This question concerns the treatment of HVAC failures, either as an initiating event or subsequent to an initiator. A description of the HVAC system is provided in Section 3.2.1.8, along with success criteria and a description of operation under normal and accident conditions. It is stated in the submittal that loss of control building ventilation will not lead to an initiator because operator inspections are performed on a regular basis. There is no discussion about other HVAC areas causing an initiating event.*

Please provide a more complete description of your investigation into the impact of loss of HVAC to the rooms containing safety-related equipment. Discuss the equipment

sensitive to temperature change, where that equipment is located, methods of assessment (e.g., calculations or tests to determine the temperatures and timing), and credits for operator actions and timing. Give this information for temporary equipment, as well. Please provide the rationale for elimination of loss of HVAC as an initiating event or as support to specific equipment. Consider, the fact that equipment may be tripped on high temperature before the damage threshold is reached.

See Sections 3.3.5 and 11.1.1.

10. *The following question concerns the treatment of flooding:*

(a) *Please discuss your consideration of drains (including back flooding to other areas and the probability of failure, i.e., due to blockage) and of doors allowing flooding of other areas. As the fire zones are used for delineation of flood zones, discuss whether all fire doors are waterproof at Ginna and whether failure of these doors to be in a closed position is accounted for in the model.*

A response will be provided by September 30, 1997 following re-analysis of the flooding PSA.

(b) *Please discuss whether inadvertent actuation of the fire suppression equipment (i.e., not just pipe failures in this system) is accounted for in the analysis and estimate its impact on the flooding scenario results if it is not.*

A response will be provided by September 30, 1997 following re-analysis of the flooding PSA.

(c) *Please discuss the operator actions needed for isolation and mitigation of the most important flood scenarios and provide the basis for flood-affected human error probabilities (HEPs) used. (It seems the same HEPs as in the internal events analysis were used for some actions, disregarding the additional stress that would be placed on the operator.) Discussion of any alarms or any other means the operators would use to detect and stop the flood.*

A response will be provided by September 30, 1997 following re-analysis of the flooding PSA.

(d) *Discuss how maintenance errors were treated in the flooding analysis. Include errors committed while in cold shutdown that were left undiagnosed until the flood event occurred while the unit was at power.*

A response will be provided by September 30, 1997 following re-analysis of

the flooding PSA.

11. *From the description of the system, "Primary Pressure Control System," it is not clear how the pressurizer power-operated relief valves (PORVS) and the block valves are modeled.*

Please provide the following information:

(a) *What fraction of time are the block valves closed?*

See Appendix C, Table C-4 (events RCMVD00515and RCMVD00516).

(b) *How are closed block valves accounted for in the model (for example, in modeling ATWS, feed and bleed, and in modeling RCS integrity after transient)? What is the estimated impact on CDF and important sequences if block valve operation is not considered?*

The potential for the block valves being closed is specifically addressed within the models for any event in which automatic or manual operation of the PORVs is required. Only for cases where manual opening of the PORVs is credited are the operators allowed to open a closed block valve (i.e., a closed block valve fails automatic actuation of the PORVs).

(c) *Discuss the operator actions required to open the block valves and the PORVs when needed.*

Operator action to open the PORVs is addressed within various operator actions described in Appendix F, Table F-4 (e.g., RCHFD01BAF, RCHFDCDDPR).

12. *The status of some of the potential plant improvements to reduce the likelihood of core damage and/or improve containment performance discussed in the submittal is not clear. Please clarify the submittal information by providing the following:*

(a) *The specific improvements that have been implemented, are being planned, or are under evaluation.*

See Section 11.1.2.

(b) *The status of each improvement, that is, whether the improvement has actually been implemented, is planned (with scheduled implementation date), or is being evaluated.*

See Section 11.1.2.

(c) *The improvements that were credited (if any) in the reported CDF.*

See Section 11.1.2.

(d) *If available, the reduction to the CDF or the conditional containment failure probability that would be realized from each plant improvement if the improvement was to be credited in the reported CDF (or containment failure probability), or the increase in the CDF (or conditional containment failure probability) if the credited improvement was to be removed from the reported CDF (or containment failure probability).*

See Section 11.1.2.

(e) *The basis for each improvement, that is, whether it addressed a vulnerability, was otherwise identified from the IPE review, was developed as part of other NRC rulemaking, such as the SBO rule, and so on.*

See Sections 11.1.2 and 11.1.3.

13. *NUREG-1335, Section 2.1.6, Part 4, requests "a thorough discussion of the evaluation of the decay heat removal function." Section 3.4.5 of the IPE, Decay Heat Removal (DHR) Evaluation, does not provide specifics and insights on vulnerabilities of DHR systems. Please discuss insights derived for DHR and its constituent systems and provide the contribution of DHR and its constituent systems (including feed and bleed) to CDF and the relative impact of loss of support systems on the frontline systems that perform that function.*

See Section 9.2.

14. *In many probabilistic risk assessments, RCP seal LOCA is a significant contributor to the CDF either as an initiating event or as a system failure consequential to another initiator. Although the submittal discusses RCP seal LOCA, please provide the following additional information:*

(a) *A discussion of the RCP seal LOCA model used. Include the probability versus leakage rate versus time data and any specific test results.*

See Section 4.2.2.3.2.

(b) *A discussion of operator actions that are proceduralized and their timing in the*

event of a loss of one or the other method (or both) of seal cooling.

See Appendix F, Table F-4 (events CCHFDCCWAB, CCHFDSTART, CVHFDPMPST, and RCHFD00RCP).

15. *NUREG-1335 requests that the following information be included for important accident sequences: "a list of major contributors to those accident sequences selected using the screening criteria. Major contributions such as those from front-line systems or functions and support states, as well as contributions from unusually poor containment performance, are important for inclusion."*

    The IPE submittal provides a table of important sequences, as well as their description. Please discuss of important contributors (e.g., "failure of operator to switch over to recirculation", or "common-cause failure of the residual heat removal (RHR) pumps") to the failure of functions in dominant sequences.

    See Sections 9.1.1 and 9.3.2.

16. *This question concerns the modeling of steam generator tube rupture (SGTR) events:*

    (a) *The SGTR initiating event frequency is somewhat smaller than expected (1.E-2 per steam generator would be expected), even though the event on the B generator was included as part of Bayesian updating. Please provide the bases for the SGTR initiating event frequency used.*

    See Sections 3.3.6 and 7.3.1.7.

    (b) *The results indicate a relatively high contribution from SGTR events, thus implying that relatively high operator failure rates were used for this event. Yet it seems that in light of the SGTR event that did occur at the plant, operator training and procedures would emphasize this kind of event. Please discuss how the HEPs for this event were derived.*

    See Appendix F, Table F-4 (events MSHFDISOLR, RCHFDCDDPR, RCHFDCDOVR, RCHFDTR2, and RCHFDCOOLD).

    (c) *If any adjustments in the initiating event frequency or post-initiator modeling are necessary in order to reflect the "as-built, as-operated plant,' please provide an estimate of the impact on the CDF and the dominant sequences.*

    Section 4.2.2.3.3, SGTR discusses the success criteria for a SGTR, Section 5.7 describes the SGTR event tree, and Section 7.3.1.7 discusses the SGTR

frequency determination. No changes to these sections are required.

17. *The pressurizer safety valves might be challenged when certain transient initiators (LOOP, loss of instrument air) occur because the PORVs depend on instrument air.*

    (a) *Please provide the conditional probabilities of PORV challenges for various classes of transients, particularly the ones leading to a loss of instrument air. Please provide the bases for the numbers used.*

    See Section 4.2.2.3.1 for a description of the events which challenge the PORVs and Section 6.15 for a description of the PORVs as used in the models.

    (b) *Please provide the conditional probability used for the safety valves sticking open once challenged in scenarios under (a). above, along with the bases for the numbers used.*

    See Table 7-1, events RY Q, RY T, RZ Q, and RZ T.

ENCLOSURE 2

HUMAN RELIABILITY ANALYSIS QUESTIONS
PRE-INITIATOR HUMAN ERRORS

1. *The submittal is not completely clear on the organizations that participated in the human reliability analysis (HRA) portion of the analysis. Please clarify the extent to which the HRA was performed by the licensee's staff versus contractors and which contractors were involved. Also, please describe any independent peer review performed for the HRA and indicate the extent to which HRA experts were involved in the review.*

   See Sections 11.3 and 11.4.

2. *The submittal does not clearly discuss the process that was used to identify and select pre-initiator human failure events (HFEs) involving miscalibration of instrumentation. The process used to identify and select these types of human events may include the review of procedures, and discussions with appropriate plant personnel on interpretation and implementation of the plant's calibration procedures. Please describe the process used to identify human events involving miscalibration of instrumentation. Please provide examples illustrating this process.*

   See Section 7.4.1 and Appendix F, Table F-3. It should be noted that miscalibration issues are typically addressed within the failure data for transmitters, indicators, etc (see "fails low," "fails high," and "fails to respond" in Table 7-1).

3. *The submittal does not clearly discuss the process used to identify and select pre-initiator HFEs involving the failure to properly restore to service after test or maintenance. This process used to identify and select these types of human events may include the review of maintenance and test procedures, and discussions with appropriate plant personnel on the interpretation and implementation of the plant's test and maintenance procedures. Please describe the process that was used to identify human events involving failure to restore to service after test or maintenance, and examples illustrating this process.*

   See Section 7.4.1 and Appendix F, Table F-3.

4. *The submittal is unclear on details of the quantitative screening approach used for HFEs involving restoration of equipment and instrument miscalibration. In Section 3.3.3, on page 3.3.3-1, the submittal notes that all HFEs were initially quantified with screening values. A review of Table 3.3.3-4 indicates that all pre-initiators had a human error probability (HEP) of 0.003. However, a discussion of the basis for this value is not provided. Please provide the rationale for the choice of the screening value*

and discuss whether any additional analyses of pre-initiators were conducted. In addition, provide the rationale for how the selected screening value did not eliminate (or truncate) important human events. Finally, if Table 3.3.3-4 does not present all the pre-initiators modeled, please provide a list of events that were screened.

See Section 7.4.1 and Appendix F, Table F-3.

5. *If Table 3.3.3-4 presents all the pre-initiator events modeled, it is not clear why are there no events representing miscalibration of level transmitters. Such events are usually modeled in probabilistic risk assessments of nuclear power plants and in some cases are found to be important. Please provide a discussion of why these events were not modeled in the R.E. Ginna Individual Plant Examination (IPE).*

   See Section 7.4.1 and Appendix F, Table F-3. As discussed earlier, miscalibration issues are typically addressed within the failure data for transmitters, indicators, etc (see "fails low," "fails high," and "fails to respond" in Table 7-1).

6. *The submittal is unclear on how dependencies associated with pre-initiator human errors (restoration faults and instrument miscalibrations) were addressed and treated. There are several ways dependencies can be treated. In the first example, the probability of the subsequent human events is influenced by the probability of the first event. For example, in the restoration of several valves, a bolt must be "tightened." It is judged that if the operator fails to "tighten" the bolt on the first valve, he will subsequently fail on the remaining valves. In this example, subsequent HEPS in the model (i.e., representing the second valve) will be adjusted to reflect this dependence. In the second example, poor lighting can result in increasing the likelihood of unrelated human events; that is, the poor lighting condition can affect the abilities of different operators' to properly calibrate or to properly restore a component to service, although these events are governed by different procedures and performed by different personnel. This type of dependency is typically incorporated in the HRA model by "grouping" the components so that they fail simultaneously. In the third example, pressure sensors x and y may be calibrated using different procedures. However, if the procedures are poorly written such that miscalibration is likely on both sensors x and y, then each individual HEP in the model representing calibration of the pressure sensors can be adjusted individually to reflect the quality of the procedures. Please provide the following information concerning the treatment of pre-initiator dependencies:*

   (a) A concise discussion of how dependencies (and human action common cause factors, where appropriate) were addressed and treated in the pre-initiator HRA.

   Dependencies were not specifically addressed within pre-initiator human errors. As described in Appendix F, Table F-3, the available indication to

operators of the status of instrumentation and major equipment eliminates this concern to a large degree. In addition, the common cause failures included within the model (see Table 7-3) specifically address the potential consequences of any common failure mechanisms.

(b) *Specific examples illustrating how dependencies were considered for pre-initiator events modeled in the IPE.*

See response to (a) above.

(c) *If dependencies and human action common-cause issues were not addressed for both miscalibrations and restoration events, please justify.*

See response to (a) above.

POST-INITIATOR HUMAN ERRORS

1. *The submittal distinguishes human failure post-initiator events from recovery actions but does not clearly describe the method used to identify and select post-initiator human failure events for analysis (only a reference to the "HRA Task Procedure" is provided). The method utilized should confirm that the plant emergency procedures, design, operations, and maintenance and surveillance procedures were examined and understood to identify potential severe accident sequences. Please describe the process that was used for identifying and selecting the post-initiator human failure events included in the event and fault tree models.*

   Section 4 addresses the success criteria used in the PSA with respect to four core protection functions and the specific need for operator action for each of these functions (see Table 4-10). Section 6 describes all human actions related to each system. Finally, Table 7-15 and Appendix F, Table F-4 describe the post-initiator human errors in detail.

2. *The submittal is unclear on the basis for the quantitative screening approach that was used for post-initiator human failure events. Table 3.1.1-12 (or Table 3.1.2-12) indicates that screening values of 0.1, 1.0, 0.21, and 1E-4 were used. Please provide the following:*

   (a) *The basis for the screening value(s) used and the rationale that led to assigning a given human action a particular screening value. Please use several examples (at least two examples for each of the four screening values used) to illustrate how it was determined that a particular action would be assigned one of the four values.*

   See Section 7.4.2. In the revised PSA, the only screening value used was 0.1.

   (b) *In addition to the examples used above, please provide the rationale for the screening values assigned to the following events: (1) AFHFDSAFWX - operators fail to start standby auxiliary feedwater (SAFW) Pump IC and ID; (2) AFHFDO4297 - operators fail to close air-operated valve 4297 to isolate steam generator (S/G) A, and (3) RCHFDOLBAF - operators fail to initiate feed and bleed.*

   Events AFHFDSAFWX, MSHFDISOLR (versus AFHFD04297) and RCHFD01BAF (versus RCHFDOLBAF) all used a screening value of 0.1. Their final values are discussed in Appendix F, Table F-4.

   (c) *Provide a rationale for how the selected screening value(s) ensured that*

important post-initiator human events were not eliminated and/or important sequences truncated.

See Section 7.4.2.

3. *Please provide a detailed discussion of the basis for assuming that an action was time independent. Apparently, an action with more than an hour available was assumed to be time independent. For these actions, what did "time available" refer to? Were the time required to perform the action and the temporal occurrence of cues relevant to a correct diagnosis considered?*

   This assumption is no longer used (see Section 7.4.2). Appendix F, Table F-4 provides a description of the times used for human actions.

4. *The values from Techniques for Human Error Rate Prediction (THERP) that were used to quantify the time-independent events appear to ignore potential diagnosis errors and the associated performance shaping factors (PSFS) that might influence such diagnoses. In addition, Chapter 15 of the THERP methodology discusses the table from which the time-independent HEPS were apparently taken and notes that the values may not be appropriate when symptom-based procedures are used. Please discuss in detail why it was unnecessary to consider potential diagnosis errors and the associated PSFs that might influence such diagnoses in quantifying time-independent events. Also, provide the basis for the use of values from Table 15-3 of the THERP when symptom-based procedures are being used.*

   The THERP approach is no longer used; instead the ASEP method per NUREG/CR-4772 was used (see Section 7.4.2).

5. *In discussing the time-independent quantification technique in Section 3.3.3.1.6 on page 3.3.3-3 of the submittal, it is stated in the third paragraph that "typically, the basic values given above were reduced by a factor of three in order to account for dependencies between events." Please explain what is meant by this statement and illustrate how the reduction is used to account for dependencies. Please provide several examples that illustrate the process.*

   This approach is no longer used for dependicies (see Section 7.4.2.). Instead, all cutsets with multiple post-initiator operator actions were identified and further evaluated to determine if the human failure probabilities should be adjusted (see Tables 7-14 and 7-15).

6. *The submittal is unclear on how the "time-dependent" quantification technique was applied to those post-initiator human events surviving initial sequence quantification.*

The submittal presents two "time-dependent" quantification tables (Tables 3.3.3-1 and 3.3.3-2), which were used to generate HEPS for human events depending on whether a given action could be considered rule-based with hesitation or rule based without hesitation. Please describe the meaning of the parameters listed at the top of these tables (m= 2 min., EF_r, EF_u) and the digits 0 through 9 at the top of the columns of the tables. So, provide the following:

(a) *Using three or more examples, please illustrate how the various parameters of the tables were considered in determining HEPS. Please provide examples that illustrate whether the values in the column headings were relevant and discuss how it was determined whether or not a particular action was "with or without hesitancy."*

This approach is no longer used (see Section 7.4.2).

(b) *On page 3.3.3-7, it is noted that for the operator action to cool down to residual heat removal (RHR) after safety injection fails, "explicit guidance on procedure transitions is not provided" and that successful performance would require a "circumvention." Yet, the HEP of 1.8E-3 would seem to be optimistic for an action without clear procedural guidance. Please provide a detailed description of the derivation of this HEP and a justification for what appears to be an optimistic HEP.*

This approach is no longer used (see Section 7.4.2). Also, the above statement is incorrect in that there is procedural guidance for this action (see Appendix F, Table F-4, events RCHFDCDOSS and RCHFDCDTR2).

7. *The submittal is unclear on what plant-specific PSFs were considered in determining HEPS for time-independent and time-dependent huma n actions. This plant-specific information could include the size of the crew, the availability of procedures, and the training, stress, and human factors aspects of the control room, and so on. On the basis of the discussion in Section 3.3.3.2, it would appear that many of these types of plant specific PSFs were not explicitly considered. If any of these types of factors were considered, please provide a list and show (by example) how their influence was factored into determining the HEPS for the various events. (Include examples of both time-independent and time-dependent events.) If none of these factors were considered, pleas provide a concise discussion of the rationale and justification for not considering such plant-specific information during the quantification of post-initiator human actions.*

This type of information was incorporated using the ASEP methodology (see Section 7.4.2).

8. *HRA methods, in general, attempt to consider both the diagnosis portion or phase of post-initiator operator actions and the execution demands of the action. Please discuss how these two different aspects of human failure events were considered in determining post-initiator human failure probabilities with the time-dependent technique. If the response execution phase of the action and the associated PSFs are not explicitly considered please provide a justification for how the values obtained with the time-dependent technique accurately reflect human failure probability.*

   Both the diagnosis and execution r equirements of each human action were considered (see Section 7.4.2).

9. *On page 33.3-2, the submittal states that "estimates for the required timing of operator actions were determined with the assistance of the Accident Sequence Analysis Task Leader." Please provide a detailed discussion of the process for determining the time required to complete operator actions and indicate how it was ensured that the resulting estimates were not overly optimistic.*

   The timings used as discussed in huma n actions are described in Appendix F, Table F-4.

10. *It is not clear from the submittal how dependencies were addressed and treated in the post-initiator HRA. The performance of the operator is both dependent on the accident under progression and the past performance of the operator during the accident of concern. Improper treatment of these dependencies can result in the elimination of potentially dominant accident sequences and, therefore, the identification of significant events. Please provide a concise discussion and examples illustrating how dependencies were addressed and treated in the post-initiator HRA for all types of actions to ensure that important accident sequences were not eliminated. The discussion should address the two following points:*

    Human events are modeled in the fault trees as basic events such as failure to manually actuate. The probability of the operator is performing this function is dependent on the accident in progression--what symptoms are occurring, what other activities are being performed (successfully and unsuccessfully), and so on. When the sequences are quantified, this basic event can appear not only in different sequences but in different combinations with different systems failures. In addition, the basic event can potentially be multiplied by other human events when the sequences that should be evaluated for dependent effects are quantified.

    Human events are modeled in the event trees as top events. The probability of the operator's peforming this function is still dependent on the accident progression. The quantification of the human events needs to consider the different sequences

and the other human events.

See Section 7.4.2.

11. *The submittal states that the specific HFEs that required detailed analysis were all associated with in-control room actions. Yet, at least one such action (CTHFDISOLA-operator action to isolate ruptured SG EMSO1A) apparently also required actions to be performed outside the control room. In the discussion of this event on page 3.3.3-5, it seems that the potential failure of actions outside the control room were ignored. Please discuss how ex-control room actions were quantified for this event or provide a justification for why it was unnecessary to do so. Provide the same discussion for any other HFEs that included ex-control room actions.*

Both in-control room and ex-control room actions were modeled. In addition, all activities required by operators in either scenario were considered in generating the final human error failure probability (see Section 7.4.2 and Appendix F, Table F-4).

12. *The submittal is unclear on how recovery actions were quantified. Although the discussion on page 3.3.7-12 regarding the determination of the indices for performance influencing factors is clear, the basis for the"multi-factored approach" is not provided. That is, how was it ensured that the summation of the indices and the insertion of the overall index into the formula on page 3.3.7.12 produced valid estimats of human failure probability? There is no evidence provided that the method has been used outside of the Ginna IPE and no indication that it has been peer reviewed or "benchmarked" in any way. Please provide a discussion addressing the validity of the quantification approach. In addition, please provide the following:*

    (a) *At least four examples that illustrate all aspects of the application of the quantification technique corresponding to the recovery events modeled in the IPE. In particular, illustrate how HEPS are derived using the formula on page 3.3.7.12. For two of the examples, address events (1) NRHSOALTCD - failure to cool down after steam generator tube rupture using steam dump or ruptured S/G and (2) NRHLETDOWN - failure to locally isolate letdown valve AOV-371 using 204A.*

    This approach is no longer being used. As described in Section 7.4.2, recovery events were treated the same as post-initiator human errors. In most cases, recovery events were added to the fault tree models in order to allow the events to be added only to the correct scenarios. With respect to the two events above, these are described in Appendix F, Table F-4, events RCHFDCDTR2 and CVHFD00371, respectively.

    (b) *A brief description of each of the recovery events modeled (apparently seven of*

them) and the HEPS assigned to these human actions.

As described above, recovery actions were treated the same as post-initiator human errors. See Appendix F, Table F-4.

13. *Guidance from NUREG-1335 requests the identification of core damage sequences that drop below the core damage frequency (CDF) screening criteria because the frequency was reduced by more than an order of magnitude by taking credit for operator actions. In addition, information was also requested on the timing and complexity of the associated human actions. Please identify the relevant sequences and provide a discussion of the related operator actions.*

    See Sections 9.1.2, 9.3.1.1, and 9.3.2.2.

14. *On page 3.3.8-17, the submittal notes that the same techniques used to recover internally initiating sequences were used to recover flood-related sequences and that previously refined HFE probabilities were incorporated. Was it necessary to modify any of the existing HFEs to reflect flooding conditions? Were any HFEs added to address potential human isolation of flood sources? If the answer to eitherquestion is yes, using examples please address how the operator actions were quantified. If the existing HFE HEPS were not modified to reflect flooding conditions or flood-specific human actions were not included, please discuss why it was unnecessary to do so.*

    A response will be provided by September 30, 1997 following re-a nalysis of the flooding PSA.

15. *The submittal is unclear on what human reliability analysis was performed during the Level 2 analysis. Please provide the following regarding the HRA for the Level 2-analysis:*

    (a) *On page 4-11 of the submittal (last paragraph), it is implied that the recovery measures considered in the Level 1 analysis are generally applied in the Level 2 analysis. Please discuss how this was done and provide a list of the relevant recovery actions and their associated HEPS. If the HEPS differed from those used in the Level 1 analysis, please describe how the HEPS were calculated.*

    A response will be provided by May 1, 1997 following the new analysis of the Level 2 PSA.

    (b) *Please list any additional operator/recovery actions considered in the Level 2 analysis (e.g., "certain containment isolation recoveries") and describe the technique used to quantify the event(s) through examples.*

A response will be provided by May 1, 1997 following the new analysis of the Level 2 PSA.

ENCLOSURE 3

LEVEL 2 QUESTIONS

1. *Containment Wall Liner Isolation. According to the Ginna Independent Plant Examination (IPE) submittal, there is a 1.25-inch thick liner insulation on the sidewalls to a point 15 feet above the spring line. The liner insulation is a closed-cell polyvinyl chloride foam insulation with low conductivity, low water absorption, and high strength and is covered with metal sheeting. Please discuss how this insulation is modeled in the modular accident analysis program (MAAP) code model and the effect this insulation has on the ability of the containment structure to absorb heat.*

   The insulation was modelled in the MAAP code by developing a containment liner/wall "gap resistance" that offered equivalent resistance to heat flow. The nominal value for the thickness and thermal conductivity of the insulation were obtained from the UFSAR, and the gap resistance was calculated to be their ratio. For conservatism, the thermal properties of the insulation were assumed not to degrade under severe accident conditions. Since MAAP 3.0B only allows one outer wall to exist in each containment control volume, the uninsulated portion of the outer wall of the containment was represented in the MAAP "upper compartment" and the insulated portion of the outer wall was represented in the MAAP "annular compartment". Since the upper and annular compartments experience very similar thermal-hydraulic conditions, this slight simplification does not impact the results.

   The presence of the insulation largely decouples the insulated parts of the outer wall from the containment atmosphere over the time frames of interest in severe accident calculations. This is partly responsible for the relatively short containment failure times calculated for sequences with quenched core debris and no containment heat removal.

2. *Reactor Cavity and Depth of Debris in the Sump for a Flooded Cavity.*

   (a) *It is stated in the Ginna IPE submittal that "the concrete thickness in the cavity away from the sump is 2.0 feet above the imbedded liner and 2.0 feet below the liner. Below the cavity sump the total thickness of the basemat concrete is 1.5 feet." However, Figure 4.1-1 indicates that the total basemat thickness in the cavity is 2 feet. Please clarify this apparent discrepancy. If the total thickness is 2 feet instead of 4 feet, please discuss how this would affect your analysis of containment basemat melt-through resulting from core-concrete interaction.*

   The thickness of the concrete basemat is as follows:

   a. In the reactor cavity sump - 1.5 feet

b. In the reactor cavity (away from the sump) - 2 feet
c. The containment floor outside of the cavity - 2 feet above the liner and 2 feet below the liner.

Since the basemat was poured directly onto bedrock, basemat melt-through is a somewhat arbitrary definition of containment failure, and the rate at which fission products would be lost after the debris penetrates the basemat is not clear.

The basemat thickness affects the analysis only in the definition of the timing of late containment failures due to basemat melt-through. Such sequences are represented by source term category (STC) case 12, for which containment failure was assumed to occur at the time when 1.5 feet of concrete attack had occurred, as would be appropriate for cases in which debris enters the sump. In cases where debris does not enter the sump; the timing of containment failure would thus be slightly underpredicted in case STC 12 (i.e., is conservative); otherwise, the error in the submittal does not affect the results.

(b) The Ginna cavity consists of a cylindrical portion with an attached rectangular volume, which contains the sump. A model developed by F. Moody was used in the Ginna IPE to estimate debris spreading and the probability of the depth of the debris in the sump for a flooded cavity with no steam explosion. Table 4.5-7 of the IPE shows that if 50 percent of the total core debris is released upon vessel failure, the debris spread radius is 4.1meters (m). Since the distance from the centerline of the reactor vessel to the closest edge of the cavity sump is 7.8m, it is stated in the IPE that "spreading to cover the entire cavity for high debris masses (40-60%) may be possible, but not likely. Hence, a probability of 0.1 is assigned to the SUMP FULL branch and a probability of 0.2 is assigned to the PART FULL branch." However, the 4.1-m debris spread radius is significantly greater than both the radius of the cylindrical portion and the half width of the rectangular portions of the cavity region, which is about 2m. The total area covered by a circle with a 4.1-m radius is about 53 m^2, which is also significantly greater than the total cavity area of 29 m^2. Please discuss how the effect of the restricted spread area in the reactor cavity is considered in the assignment of the probability values for debris depth in the sump and what the effect of higher probability values for debris in the sump would be on containment failure probabilities.

After investigating the assumptions used in the development of the Moody model, RG&E agrees with the inference by the reviewer that it would have been more appropriate in an irregular geometry such as the Ginna Station reactor cavity to reformulate the model to calculate the spread area rather than spread radius. If this is done, the model calculates that debris should spread to fill an area nearly twice that represented by the cavity floor. This would imply that debris could be

expected to enter the cavity sump.

However, it should also be noted that the heat flux between the debris and the water is calculated in the Moody model by assuming that enhanced film boiling occurs at the surface of the debris. The parameters used in the PSA analysis were drawn from Moody's paper and result in an assumed heat flux of about 800 kW/m^2. Small scale experiments performed by Henry et al. to address the Mark I liner melt-through issue measured heat fluxes during the period of simulated debris discharge that were an order of magnitude larger than this value [1]. If such an augmentation in heat flux also occurred in the reactor case over the period of debris discharge, this would overwhelm the problem noted by the reviewer and prevent the debris from flowing to the sump.

Theofanous et al. [2] has criticized the experiments of Henry et al., since no analysis of debris flow regime was conducted. Given the pressurized (150 psi) discharge of simulant core debris in the experiments, Theofanous et al. noted that intense mixing created by melt splashing off the floor and running up the side walls could explain the high measured heat fluxes. However, for the Mark I transient (in which the water level is only ~0.9 m deep, much less than the level in the Ginna Station cavity if the RWST is discharged) Theofanous et al. agreed that rapid fragmentation and quenching of the core debris would be expected as the debris traverses the water pool, and that a local pileup of debris immediately under the RPV failure location could occur.

The probability of the cavity sump being either partially or completely full of debris was estimated in the PSA for three separate ranges of debris mass discharged at vessel failure:

Fraction of debris discharged at vessel failure	Probability of partly full sump	Probability of full sump
< 20 percent	0.01	Impossible
> 20 and < 40 percent	0.1	0.05
> 40 and < 60 percent	0.2	0.1

Considering both the irregular cavity geometry and the likelihood of debris jet break-up and larger debris/water heat fluxes during the debris ejection phase, these probability values are still considered reasonable.

Recognizing that this was an uncertain issue, a sensitivity case was run as part of the PSA in which it was assumed that the probability of having a full sump was unity (except for the < 20 percent discharged debris case in which the sump

cannot fill and the probability of a partly full sump was instead considered unity). As discussed in Section 4.8.1.4 and Table 4.8-4 of the March 15, 1994 submittal, the effect of this change was to increase the probability of source term category (STC) 12 (which represents late containment failure due to basemat melt-through) from 13 percent to 30 percent of the total CDF; the no containment failure STC was reduced by the same amount to about 16 percent of CDF. Note that even if heat transfer was assessed using the film boiling assumption, the probability of a full cavity in the medium debris mass cases would be less than unity, and this sensitivity calculation therefore somewhat overstates the magnitude of the effect noted by the reviewer. It is also worth noting that late basemat melt-through sequences such as STC 12 have low source terms and that the fission product releases in the Ginna Station PSA were dominated by containment bypass sequences which are unaffected by the spreading issue (see Question 9). Although the calculations for debris spread in the March 15, 1994 submittal were determined to be non-conservative, consideration of these other factors leads RG&E to believe that our results are acceptable.

3. *Cut set (CSET) Structure and Power Recovery. According to the IPE submittal, the CSET tree structure of the branch for recovery of ac power prior to vessel failure (PRV) is identical to that shown in the tree structure leading to endpoints 1-46 in Figure 4.3-1. On the other hand, the structure of the branch for the recovery of power prior to containment failure (PRC) is assumed to be identical to that leading to endpoints 1-8. Figure 4.3-1 shows that all core injection and recirculation systems (i.e., low-pressure injection and recirculation and high-pressure injection and recirculation) are available for endpoints 1-8; while one or more of these systems are not available for endpoints 9 to 36: Please discuss why all these systems are assumed to be available for the cases of power recovery prior to containment failure but some of the systems may not be available for the cases of power recovery prior to vessel failure.*

In Figure 4.3-1 of the March 15, 1994 submittal, branch 49 shows a blackout sequence where power is not recovered at all, branch 48 shows a blackout sequence where power is recovered prior to containment failure and branch 47 shows a blackout sequence where power is recovered prior to vessel failure. Since branches 1-46 are non-blackout sequences, power recovery is not an issue in any of these branches and the questions of what injection systems are available are instead addressed.

4. *External Cooling of the Reactor Pressure Vessel. It is stated in the submittal (page 4-4) that "whenever the contents of the RWST [refueling water storage tank] are injected into the containment the cavity will be completely filled and will remain filled." Based on this statement, it seems possible that the ex-vessel water may provide sufficient cooling to the core debris inside the vessel sothat vessel failure could be avoided or significantly delayed. As a result, fission product production and release paths could be affected (e.g., in-vessel*

release from a dry debris bed versus ex-vessel release from a debris bed covered by water). The release of fission products to the environment may actually increase if the containment fails and external cooling was accounted for in the source term calculation. Please discuss the potential of ex-vessel cooling for Ginna and its effect on source term definition. Because external cooling may maintain the reactor coolant system (RCS) at high temperature for a longer time, please also discuss the effect of external vessel cooling on the probability of creep rupture of the RCS boundaries and the steam generator tubes and, consequently, the effect on containment performance and source terms for Ginna.

When the PSA was initially performed, the subject of ex-vessel cooling of core debris in the lower head was being actively studied. Another area of active research at that time was the expected mode of vessel failure, primarily through the TMI Vessel Inspection Program (VIP). The PSA analysts judged it likely that vessel failure would occur at a penetration soon after core relocation, which was the assumed mode of vessel failure in the IDCOR program. This made it unlikely that the debris could be retained in the lower head, and for this reason, no credit was taken for debris retention in the RPV. This assumption was considered a potential conservatism of the PSA.

More recently, a consensus appears to be developing that retention of debris in the lower head is fairly likely. For sequences with injection of the RWST and containment heat removal available, this would serve to prevent vessel failure and thus the potential for containment failure.

RG&E is unaware of any calculations of fission product release from debris being cooled in the lower head of the reactor vessel. However, such releases would be expected to be very small in the long term, since debris temperatures would be relatively constant, the surface area would be small, and most importantly, there would be no sparging of the debris by gasses as is the case for the in-core and ex-vessel fission product release mechanisms. Thus, cooling of the debris in the lower head is not expected to represent a new safety issue from the standpoint of fission product release from debris.

With respect to inducing the failure of RCS components, induced rupture of the hot legs is expected to occur long before debris relocation to the lower head in high pressure accident sequences. Hot leg rupture would depressurize the RCS and remove any threat to the steam generator tubes. Recent calculations performed by both NRC contractors as well as the industry indicate that hot leg or surge line rupture is also likely in medium pressure sequences. For sequences involving RCS pressures low enough not to result in an induced rupture of the hot leg or surge line prior to core relocation, subsequent thermally-induced rupture of the steam generator tubes is only credible and of concern if:

a. The sequence is not a bypass event and the steam generators are initially intact (i.e. sequence is not initiated by a steam generator tube rupture);

b. One or both steam generators are depressurized to pressures less than the RCS pressure (otherwise the steam generator tubes would be in compression rather than tension);

c. The RCS pressure is significantly greater than the secondary side pressure so as to threaten the tubes but sufficiently small as not to have already caused an induced rupture of the hot leg or surge line;

d. The affected steam generator(s) secondary side is dry so that the tubes are not cooled (otherwise thermally-induced rupture cannot occur); and

e. There is no supply of water to the debris (a continuous supply of water, i.e. from a RHR pump, would cover the debris and cool the RCS).

Past experience indicates that the most likely scenarios fitting these criteria are total loss of feedwater sequences involving LOCA(s) (e.g. large RCP seal LOCAs or open pressurizer safety or relief valves) which also result in one or more depressurized steam generators. Given the presence of LOCA(s), one would not expect RCS pressure to be maintained significantly above the steam generator pressure after the water initially in the lower head or that supplied by the accumulators is boiled away and the RCS depressurizes. For this reason, it is believed that induced rupture of the steam generator tubes caused by debris retention in the lower head is very unlikely.

Therefore, we conclude that failing to credit debris retention in the lower head of the RPV represents a conservatism of the PSA rather than a concern.

5. *The Availability of Containment Fan Coolers and Containment Spr ays. In the IPE model, if the containment fan coolers are available, containment sprays in both the injection and recirculation modes are modeled as failed (page 4-13). Although both systems have the same effect on containment heat removal, their effect on source term definition may be different. Sprays are usually credited with being more efficient than fan coolers in removing fission products from the containment atmosphere. Please state whether containment sprays are included in the MAAP model for source term calculation (for cases in which containment heat removal is available). If they are, please justify the use of containment sprays for source term calculations in all sequences with ontainment heat removal. If they are not, please estimate the potential source term mitigation effect achieved by the operation of containment sprays.*

None of the MAAP calculations of representative source terms (MAAP calculations STC 1-STC 20) credited the sprays, consistent with the assumptions made in the sequence quantification. For sequences in which the debris is water-covered in the containment and fan coolers were credited, the only substantial benefit from assuming that the sprays also

operate would be expected in non-containment bypass sequences involving early containment failure (especially those with large containment breech areas). The high rate of removal of aerosols afforded by the sprays could be expected to reduce the source terms in these cases since the fission product release is determined by the relative magnitude of the rates of aerosol leakage and deposition. Bypass sequences are unaffected by containment sprays (which usually do not operate anyway), and fan coolers would have sufficient time to be effective in sequences with late containment failure.

The source term categories involving early containment failure with containment heat removal are STCs 2, 3, 5, 13, and 14. Of these, STC 2 and 3 involve small containment leak areas and only STC 13 has a conditional probability greater than 0.1 percent. STC 13 represents 3 percent of the CDF and is caused by a failure of containment isolation. The calculated source terms for this sequence, shown in Table 4.7-2 of the March 15, 1994 submittal, are moderate, but are much less than those resulting from containment bypass scenarios (STC 16, 18, and 20) which are also much more likely (over 40 percent of CDF). Thus, while neglect of the sprays represents a conservatism, primarily in that STC 13 may have too large a source term, inclusion of the sprays would have a negligible effect on the results as a whole. If these results change significantly due to the re-quantification of the Level 1 sequences, operation of the sprays may need to be evaluated in the Level 2 update.

6. *Steam Generator Tube Rupture (SGTR) Releases. The IPE results show that the steam generator power-operated relief valve (PORV) cycles during and aftercore damage in about half of the SGTR events, and there is a stuck-open PORV in the remaining half of the SGTR events. Please discuss how the probability of steam generator (SG) valve failure is determined in the analysis and how the effect of the harsh conditions (e.g., the flow of extremely high temperature gases with entrained debris) on the operation of the SG valves is considered in the analysis.*

The steam generator PORV or atmospheric relief valve (ARV) failure probability is determined in the Level 1 data analysis portion of the PSA. No adjustment to the ARV failure probability was made based on harsh conditions; however, this will be considered in the update to the Level 2 PSA.

7. *The Probability of Power Recovery. According to Table 4.3-2 of the IPE submittal, the probability of power recovery prior to vessel failure is 0.622, and the probability of power recovery after vessel failure but prior to containment failure is 0.127. However, the plant damage state (PDS) results show that among all station blackout sequences (included in PDS 2 through 8), 54 percent have power recovery after vessel failure but prior to containment failure (PDS 7). The recovery probability of 0.54 is much greater than the value obtained in the power recovery analysis (0.127). Please explain this apparent discrepancy.*

The probability of power recovery for the Level 2 analysis was based on the Level 1 power recovery curve. This has since been re-performed (see Appendix C). This curve will be used when the Level 2 portion of the PSA is re-quantified.

8. *Capacity of the Containment Vessel. Section 4.4 of the IPE submittal discusses the evaluation of containment ultimate strength. According to the IPE submittal, the ultimate strength and failure modes of the Ginna containment were determined why a finite element analysis performed by Ebasco Services. However, the criteria used to determine the ultimate failure pressure are not discussed in the submittal. In the IPE, the failure pressures obtained were assigned an uncertainty of 5 percent for containment failure evaluation. The 5 percent uncertainty used in the Ginna IPE seems to be less than that used in other IPEs. The difference between the 5th percentile failure pressure and the median failure pressure for Ginna is much less than that found in other IPEs or in the NUREG-1150 analyses. Please discuss the criteria used to determine the containment failure pressures and the basis for the 5 pecent uncertainty associated with these pressures.*

The Ebasco study separately considered gross failure of the containment and local failures caused by liner tears at containment penetrations. The study used the ABAQUS and HEATING6 computer codes, and was based on minimum steel and concrete properties as defined in the UFSAR and relevant material specifications. Gross containment failure was associated with a rapid increase in plastic strain at 155 psia; specifically, this failure pressure corresponds to the achievement of 3 percent strain in the circumferential reinforcement at mid-containment height.

Liner tearing was evaluated by estimating the failure strain near penetrations. Such locations are empirically associated with a three-dimensional state of stress, for which the value of the "triaxiality factor" can be bounded by 5. The triaxiality factor allows the effective failure strain to be related to the known uniaxial elongation failure strain measured for the liner steel. In this case, the uniaxial failure strain of 9.8 percent results in an effective failure strain of 0.6 percent. Such a strain was observed in the finite element results at a pressure of 145 psia; the calculated strain increases rapidly above this pressure, so any errors in the effective failure strain will result in only a small change in failure pressure associated with liner tears.

Ebasco judged that the uncertainty in the results of the finite element analysis itself was 5 percent. As discussed in Section 4.4.3 of the March 15, 1994 submittal, this value was taken to be the standard deviation of a normal distribution centered around the calculated failure pressure which was taken to be the median. As stated above, the Ebasco analysis was based on minimum material properties, and no uncertainty in material properties was considered in the analysis. For this reason, the fragility curve developed in the PSA should be regarded as a representation of the uncertainty in the minimum failure pressure that is consistent with the various materials specifications.

Qualitatively, consideration of material property uncertainties would shift the fragility curve toward higher pressures and would broaden the distribution; the expected magnitude of the broadening is not known. The Ginna Station composite fragility curve, Figure 4.4-3, spans a pressure difference of about 12 psid in going from the fifth percentile to the median. Comparison of the Ginna Station results to those shown for Zion in Figure 1-1 of Reference [3] indicates that this measure of uncertainty in the Ginna Station curve is actually larger than that estimated by one analyst (expert C), comparable to that estimated by a second (expert A), and much less than that estimated by a third (expert B).

Based on this comparison, it is believed that the Ginna Station fragility curve is conservative, but not to a substantial degree. As noted in Section 4.8.1.3, sensitivity calculations indicate that the PSA results are insensitive to reasonable changes in the fragility curve.

9. *Containment Isolation Failure. According to the IPE submittal, loss of containment isolation sequences represents 3.0 percent of the total core damage frequency (CDF), or 5.2 percent of the frequency for noncontainment-bypass CDF. Section 3.2.1.3 provides a description of the containment isolation system and the operating experience of the containment isolation system at the Ginna plant. Section 4.3.1.2.4 mentions that a fault tree was used for containment isolation quantification. However, details of the analysis and the results are not provided in the submittal. With respect to the analysis of containment isolation failure probability, NUREG-1335 (Section 2.2.2.5, page 2-11) states that "the analyses should address the five areas identified in the Generic Letter, i.e., (1) the pathways that could significantly contribute to containment isolation failure, (2) the signals required to automatically isolate the penetrations, (3) the potential for generating the signals for all initiating events, (4) the examination of the testing and maintenance procedures, and (5) the quantification of each containment isolation failure mode (including common-mode failure)." The 5 percent probability of containment isolation failure is significantly greater than that of most IPES. Please discuss the significant containment isolation modes (e.g., the penetrations that fail to isolate and the causes for isolation failure) obtained from the IPE analysis and for these major containment isolation modes discuss how the five areas listed above were addressed.*

The five areas required to be evaluated by NUREG-1335 were, in fact, considered in the Level 2 PSA. These areas were evaluated in detail in the Containment Isolation System Work Package and summarized in Section 3.2.1.3 of the March 15, 1994 submittal. The 5 percent probability of containment isolation failure was due almost entirely to ISLOCA sequences. These sequences have since been re-evaluated (see Section 8.2 of the new submittal) with significantly different results. These new results will be included in the forthcoming Level 2 re-analysis. Additional details of the containment isolation analysis will also be provided at that time.

10. *Penetration Seal Failure. Failure of containment penetrations is dismissed in the Ginna IPE as a potential containment failure mode because the analysis in NUREG-1150 indicated that this failure mode was significantly less important than the overpressure failure of the containment cylinder wall (page 4-28 of the submittal). Please provide a description of the seal materials used for the penetrations in Ginna, their properties, and the potential harsh containment conditions to which they could be exposed. On the basis of this plant-specific information, please explain how you concluded that the findings in the NUREG-1150 analysis cited in the submittal were applicable to the Ginna plant.*

Systematic Evaluation Program (SEP) Topic VIII-4, Electrical Penetrations of Reactor Containment provides detailed information about the penetration seal materials and their properties. Reference [4] provides the NRC's safety evaluation of this topic. UFSAR Section 3.11 and the MAAP runs made for the Level 2 analysis detail the containment conditions to which the containment penetrations are expected to be exposed. The containment penetrations will be evaluated using this information and the information in NUREG-1037, "*Containment Performance Working Group Report*". The results of this evaluation will be included in the forthcoming Level 2 re-analysis.

11. *Containment Performance Improvement and Hydrogen Issues. The generic letter containment performance improvement recom mendation for pressurized-water reactor dry containments is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures).*

Please discuss whether plant walkdown inspections have been performed to determine the probable locations of hydrogen releases into the containment. Including the use of walkdown inspections, discuss the process used to assure that (1) local deflagrations would not translate to detonations given an unfavorable nearby geometry and (2) the containment boundary, including penetrations, would not be challenged by hydrogen burns.

Please identity potential reactor hydrogen release points and vent paths. Estimates of compartment free volumes and vent path flow areas should also be provided. Specifically address how this information is used in your assessment of hydrogen pocketing and detonation. Your discussion (including important assumptions) shou ld cover the likelihood of local detonation and the po tential for missile generation as a result of local detonation.

A containment walkdown was performed on May 4, 1992 for the purpose of assessing containment features important for understanding the response of the Ginna Station containment during severe accidents. One focus of the walkdown was to establish the most important inter-compartment gas flow paths. The results of the walkdown indicate that:

a. Substantial global natural convection currents would be expected in a severe

accident, driven primarily by heat loads from the RCS and where applicable, from debris in the containment, and by heat removal by the walls and active safeguards systems. Gas flow is expected to occur upward in the lower compartment region surrounding the RCS components and through grating over the reactor coolant pumps and the annular regions around the steam generators and pressurizer. The gas will flow into the upper dome where it will cool by contact with uninsulated heat sinks (and spray droplets where available), and flow down the annulus outside the shield wall. From the outer containment annulus, the gas will flow back to the lower compartment region through two stairwells, grating, and open areas around HVAC ducting located at the 253' elevation.

b. The limiting flow area on this circulation loop is quite large, about 450 square feet [5]. Driven solely by the nominal heat losses from the RCS, the overall containment mixing time is calculated to be only ~6 minutes [5]. In actuality, this calculation underpredicts containment mixing since the effect of containment heat removal systems has been neglected; sequences without containment heat removal operational will result in too high a steam concentration to result in deflagrations, much less deflagration to detonation transition (DDT).

c. The principal hydrogen release points to the containment prior to RPV failure are RCS breaks in the regions surrounding the coolant loops or the pressurizer relief tank rupture disk for sequences with no cycling or open pressurizer safety or relief valves. All of these locations are well-ventilated, have substantial flow areas available directly overhead, and thus do not pose a threat of localized hydrogen build-up. Hydrogen released to these locations would be expected to be entrained into the overall containment circulation and mixed with the other gasses as they flow into the upper dome region.

d. Diffusion flames occurring at any of these hydrogen release points inside the shield wall would not endanger containment penetrations. Global hydrogen burns would be of short duration and would not be expected to threaten the penetrations.

If one assumes that hydrogen released to containment is well-mixed, the potential for DDT can be qualitatively assessed by calculating the maximum average hydrogen concentration that can be achieved. Even if all the zirconium in the core is oxidized (a more realistic upper bound is about 75 percent), the hydrogen concentration in the containment would be only 11 percent in the complete absence of steam. In actuality, some steam would always be in containment during a severe accident. Also, it is very unlikely that such a large amount of hydrogen could be released to containment so quickly as to avoid deflagrations at lower hydrogen concentrations. All of these considerations would limit the peak achievable hydrogen concentration. In any event, 11 percent is too small to support DDT, even for geometries that promote flame acceleration.

Additional hydrogen and carbon monoxide could be released during core-concrete interactions (CCI) as steel in the core debris or basemat reinforcement is oxidized. However, the release of combustible gas from steel oxidation in CCI would occur relatively slowly, would be accompanied by diluents such as carbon dioxide and steam, and would promote mixing in the containment. Thus CCI would also be very unlikely to result in combustible gas concentrations sufficient to cause DDT.

High combustible gas concentrations could potentially be created if containment heat removal was restored after a long period of CCI. In the PSA, STC 9 represents a sequence in which a late deflagration is initiated that causes containment failure. In fact, experiments suggest that burns will be initiated as soon as sufficient steam is condensed that the containment gas is no longer inerted [6]. These burns are observed to be so incomplete that no significant pressurization of the containment results (except when near-stoichiometric mixtures are created). Again, detonations are unlikely.

Based on these considerations, the threat of hydrogen detonations is judged to be very small in the Ginna Station containment. This is consistent with past studies on other PWR large, dry containments [7].

12. *Equipment Survivability. The availability of containment fan coolers and containment sprays is considered in the plant damage state (PDS) definition of the IPE. The effect of harsh environmental conditions on the operation of this equipment are not discussed in the containment event tree (CET) quantification of the submittal . Please provide a description of how the survivability of this equipment under severe accident conditions was evaluated. Please include in the discussion the environmental conditions (e.g., temperature, pressure, radiation, and debris) derived and used in the evaluation.*

The Level 2 PSA did not consider the effect of environmental conditions on the operability of these systems. To our knowledge, previous analyses have not concluded that containment spray operation would be threatened by containment conditions, except perhaps those resulting in catastrophic containment failure. For example, in the NUREG 1150 study on Zion [8], only a 10 percent likelihood was assigned to failure of containment sprays following catastrophic containment rupture; sprays would be relatively ineffective in reducing source term after gross containment failure in any event. The Zion NUREG 1150 study did not explicitly consider the deleterious effect of containment conditions on the fan coolers at all (see APET questions 24, 43, 45, 46, 53, 65, and 66) [8].

To assess whether the Ginna Station fan coolers would continue to operate in a severe accident, the MAAP calculations used to characterize the various source-term categories were reviewed. This set of calculations is useful for representing the spectrum of accident calculations that could be encountered in a severe accident. As noted previously,

containment sprays were not credited in these calculations; for this reason, gas temperatures would be somewhat higher than expected in sequences where the sprays were actually available.

STCs 2, 3, 4, and 12 involve operable fan coolers prior to vessel failure. The MAAP-calculated conditions in these sequences were compared to the design basis envelope for the fan coolers described in Chapter 6.1.2 of the UFSAR. The pressure and radiation conditions specified for the DBA are generally more severe than the severe accident conditions that would be encountered for all sequences not involving a late recovery of containment heat removal. STC 9 involves a late recovery of fan coolers which gives rise to a late failure of containment from a hydrogen burn; failure of the fan coolers to operate in this sequence would not therefore lead to higher risk, as calculated by the PSA.

Since the fan coolers are located in the annular region outside the primary component shield walls and take suction there, the MAAP-calculated gas temperatures in this region were used to assess the threat from high temperatures. For all but STC 9, gas temperatures are less than the DBA envelope (286°F for 2 hours) except for brief spikes observed during hydrogen burns. Since the spikes are of short duration, it is judged that the fan coolers would survive these events. In this regard, it should be noted that the Ginna Station fan cooler motors are actively cooled by service water and also that the TMI fan coolers continued to operate after the global hydrogen burn that occurred during the accident [9].

It is noted that a possibility exists that the fission product filters could become clogged due to high aerosol loadings during a severe accident. This should not threaten the effectiveness of the fan coolers since only 2 of the units have filters and since the filters can be manually bypassed.

Therefore, we conclude that operation of the fan coolers and sprays is unlikely to be threatened by the environmental conditions in the containment and no changes are required.

13. *EVNTRE Events. In the Ginna IPE, a small CET is developed for accident progression analysis. The top events of the CET are determined in the IPE by the use of decomposition event trees (DETs). The event progression analysis code EVNTRE, which was developed and used in the NUREG-1150 analyses, is used for event tree quantification in the IPE. To use the code, the top events developed in the Ginna event trees are numbered and incorporated into the EVNTRE model. The highest number for the EVNTRE events that can be identified from the trees in the Ginna submittal is 47. This seems to indicate that there are 47 events (or questions) in the EVNTRE model. However, examination of the event trees in the Ginna IPE shows that the total number of events in the trees is less than 47. As a result, some EVNTRE questions (e.g., 14, 17, 18, etc..) cannot be identified from*

the Ginna event trees presented in the submittal. Please provide a list of all EVNTRE questions used in the Ginna analysis and discuss the questions that are in the EVNTRE model but not in the Ginna event trees.

A complete list of the EVNTRE events in the CET is provided in Table 1 at the end of this section. The first 12 events in the model correspond to the top events shown in Fig 4.3-2 of the March 15, 1994 submittal. The remaining events are either described in the text of the submittal or are "dummy" events included for the convenience of the analyst that do not affect the results. The dummy events are numbered 14, 17, 18, 27, 29, 31, 35, 37, 38, 40, 41, and 45. All non-dummy events were described in the submittal.

14. *Modular Accident Analysis Program (MAAP) Calculation Results. A number of MAAP calculations were performed in the Ginna IPE to provide data for the accident progression analyses. A brief description of each of these accident progression cases is given in Table 4.6-1 of the submittal. MAAP calculations were also performed to derive release fractions for the various source term categories. Brief descriptions of these source term cases are also provided in Table 4.6-1. It can be seen that the conditions of Case MLOCA03 (page 4-150) and Source Term Case STC12 (page 4-161) are similar. For both cases, the auxiliary feedwater (AFW) is available and the safety injection system is unavailable. However, the core uncovery time and the vessel failure time shown in Table 4.6-1 are significantly different for these two cases (they are 4.27 and 5.96 hours, respectively, for Case MLOCA03, and 0.7 and 1.55 hours, respectively, for Case STC12). Please discuss the reasons for the time difference between these two cases, where and how each case was used in the IPE analysis, and the impact of the data from each case on IPE quantification.*

As used in Table 4.6-1, "No SI" refers to the unavailability of high pressure injection. Cases with no safety injection at all (i.e., high and low) are labelled "No injection" (see for example case SLOCA00). The apparent discrepancy between the two cases cited is caused by the availability of low pressure injection in MLOCA03 which delays core uncovering and vessel failure relative to case STC 12 which involves a complete failure of all active injection systems.

The results of MAAP calculations were used in the Ginna Station PSA to assess sequence timing (see, for example, the discussion in Section 4.3.1.1.2), to obtain insights useful for the quantification of the CET (e.g., events such as induced rupture of the RCS, direct containment heating, deinerting the containment atmosphere, etc.), and to assign explicit source terms to source term categories. This is discussed in Sections 4.5, 4.6, and 4.7 of the March 15, 1994 submittal.

The results from MAAP calculation MLOCA03 were not used explicitly in the PSA and were performed simply to provide the analysts a qualitative understanding of the effect of break size and safety injection system availability on sequence timing. The results of

calculation STC 12 were directly used to assign source terms to the associated source term category.

15. *Sequence Selection for Source Term Determination. It is stated in the IPE submittal (Section 4.7.3) that "specific accident progression sequences were chosen to best approximate the representative source tem results for each relevant Source Term Category (STC) end state. Based on consideration of the dom inant sequence for each end state and based on other factors which influence t@ source term results, representative sequence descriptions were developed to perform MAAP calculations to quantify the source terms." However, in the submittal, the PDSs that contribute to the STCs are discussed only for a few STCS, and for some of these cases the sequences selected for MAAP calculations are not the dominant sequences in the PDSS. For example, according to the submittal; PDS 15 and PDS 17 represent the majority of STC 2. The representative sequence chosen to represent this STC is a medium-break loss-of-coolant accident (LOCA) sequence. However, according to Table 4.3-5 of the submittal, the sequences that contribute to these PDSs are small LOCA or reactor coolant pump (RCP) seal LOCA sequences, and medium LOCA sequences are not involved in either of these two PDSS. There is no discussion in the submittal why a medium LOCA instead of a small LOCA sequence was chosen to represent this STC. Please provide a complete list of the contributing PDSs for all the STCs and provide the rationale for the selection of each of the sequences chosen to represent the STCS.*

Contributing PDSs are not discussed for each of the STCs because of the many STCs that a PDS might contribute to and because of the low probabilities associated with a number of the STCs. For example, PDS 2 comprises only 0.35% of the PDSs per Table 4.3-4. Almost 69% of PDS 2 contributes to STC 15, but it is still not a significant contribution since the PDS is of such a low frequency (about 23% of PDS 2 contributes to STC 1 and the remaining 8% contributes to 6 other STCs). The following is a listing of the top 11 PDSs per Table 4.3-4 and the STCs they primarily contribute to in approximate percentages:

- PDS 12 contributes to STC 1 (72%) and STC 12 (28%)
- PDS 22 contributes to STC 18 (100%)
- PDS 24 contributes to STC 20 (100%)
- PDS 20 contributes to STC 16 (100%)
- PDS 15 contributes to STC 1 (59%) and STC 12 (41%)
- PDS 17 contributes to STC 1 (97%)
- PDS 11 contributes to STC 15 (95%)
- PDS 1 contributes to STC 13 (100%)
- PDS 9 contributes to STC 15 (65%), STC 1 (21%) and STC 12 (13%)
- PDS 7 contributes to STC 9 (48%) and STC 14 (48%)
- PDS 14 contributes to STC 12 (100%)

The source term characterization was considered less important than the characterization of the core damage sequence and the achievement of an overall understanding of containment performance. Selection of sequences to represent STCs was primarily guided by a desire to:

a. Represent the key phenomenological effects that were used to characterize the STCs. These effects, e.g. the occurrence of CCI, availability of water on the debris, availability of containment heat removal, size and timing of containment failure, etc. are shown in Figure 4.7-1 of the March 15, 1994 submittal. Break size was not considered sufficiently important to be used in the classification of source term categories, although it does affect the degree to which fission products are retained in the RCS.

b. Minimize analytical effort for sequences which do not contribute significantly to the risk

As described in Section 4.7.3.1, STC 2 represents sequences with vessel failure, a leak-type early containment failure, containment heat removal, and no sustained CCI. As such, the source term is dominated by the release of fission products from the vessel in the period just leading up to and following vessel failure. A medium LOCA sequence was chosen to reduce in-vessel retention of fission products so as to provide a conservative, yet reasonable assessment for the various PDSs comprising this release category.

Note also in Table 4.7-1 that STC 2 represents less than 0.1 percent of the CDF. By contrast, STCs 16, 18, and 20 represent nearly half the CDF and all involve bypass of the containment. All three of the latter sequences can a priori be expected to have the largest source terms, at least as modeled by MAAP. These source terms are probably over-predicted by the MAAP calculation since the code neglects turbulent deposition in piping (STC 16) and on the secondary sides of the steam generators (STCs 18 and 20) and since retention of fission products in the auxiliary building was not modeled (STC 16). In any event, given the MAAP-calculated source terms, negligible errors are incurred in the results as a whole by failing to more precisely model sequence-specific aspects of the much lower-probability end states such as STC 2 that have relatively small fission product releases.

16. *Induced SGTR -- The likelihood of induced hot leg or steam generator tube ruptures in high pressure scenarios is analyzed in the Ginna IPE using MAAP analysis data and an empirical formula, developed by Larson and Miller, relating the expected time to rupture with temperature and stress. Since the conclusions from this analysis are consistent with the results from the NUREG-1150 data for Surry, the probability values used in the NUREG-1150 Surry analysis are used in the Ginna IPE. It should be noted that in some IPES the probability of induced SGTR due to forced circulation caused by the restart of the*

RCPs is addressed because the insufficient core cooling (ICC) guidelines call for the RCPs to be restarted. Please discuss whether there are procedures at Ginna that call for the restart of the RCPs and, if there are, please discuss theirffect on the probability of induced SGTR.

Ginna Station Procedure FR-C.1, Response to Inadequate Core Cooling, potentially calls for restart of the RCPs after the core is uncovered. During the time frame when the PSA was being performed, concern was expressed that this might cause an increase in heat transfer from the core to the steam generator tubes, potentially causing their failure. This concern would only be relevant in the relatively small number of sequences in which the secondary side of the steam generators was dry, RCPs were available, and RCS pressure was elevated compared to the steam generator pressure. In any event, this issue was not addressed in the Ginna Station PSA.

The current version of the procedure, revision 10 dated 11/29/95, calls for an RCP to be restarted in a given loop only if the narrow range level exceeds 5 percent in the associated steam generator (see step 23). This ensures that the affected tubes will be cooled and eliminates concerns for thermally-induced tube rupture.

17. *Typographical Errors*

(a) *AC power recovery considers only the recovery of off-site power. The recovery of DGs is not credited. It seems that there is a typographical error in Table 4.3-2. The probability of power non-recovery at 21 hours should be 0.00566 instead of 0.0566 shown in the table.*

The typographical error will be corrected during the new Level 2 analysis.

(b) *The title of Reference 4.9-19 of the submittal is given as "Evaluation of Severe Accident Risks: Ginna Unit 1 ". It should be "Surry Unit 1" instead of "Ginna Unit 1".*

The typographical error will be corrected during the new Level 2 analysis.

(c) *Event 9 of the CET top event "Ex-Vessel CCI" is described in the submittal (p4-74) as CAV_WAT_F. However, it is described as L_RWST in the CCI decomposition event tree (Figure 4.5-6, Paragraph 4-196).*

The typographical error will be corrected during the new Level 2 analysis.

References:

1. E. Fuller, ed., Containment Performance and Fission Product Release Determination for Individual Plant Examinations (IPES), Proceedings of a Workshop in Chicago, IL, July 1989, Electric Power Research Institute. These results are also reported in R. E. Henry et al., "Experiments Relating to Drywell Shell-Core Debris Interactions", Proc. 16th Water Reactor Safety Information Meeting, Gaithersburg, October 24-27, 1988, NUREG/CR-0097.

2. T. G. Theofanous and H. Yan, "The Probability of Liner Failure in a Mark-I Containment, Part II: Melt Release and Spreading Phenomena", Nucl. Tech., 101, March 1993.

3. R. J. Breeding et. al, Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, NUREG/CR-4551, Vol. 2, Rev. 1, Part 3.

4. Letter from D.M. Crutchfield, NRC to J.E. Maier, RG&E, Subject: :SEP Topic VIII-4, Electrical Penetrations of Reactor Containment Safety Evaluation Report for R.E. Ginna Nuclear Power Plant", dated October 8, 1981.

5. Memorandum from M. Kenton (D&M) to Daphne Mays (RG&E) dated July 16, 1996.

6. T. Blanchat and D. Stamps, "Deliberate Ignition of Hydrogen-Air-Steam Mixtures Under Conditions of Rapidly Condensing Steam", SAND 94-3101C, January 1995.

7. J.W. Yang, Z. Musicki, and S. Nimnual, Hydrogen Combustion, Control, and Value-Impact Analysis for PWR Dry Containments, NUREG/CR-5662, BNL-NUREG-52271, June 1991.

8. C. Park et. al, Evaluation of Severe Accident Risks, Zion, Unit 1: Appendix A., NUREG/CR-4551 Vol. 7 Rev 1 Part 2A, March 1993.

9. J. O. Henrie and A. K. Postma, Analysis of The Three Mile Island (TMI-2) Hydrogen Burn, RHO-RE-SA-8 P, Rockwell International, October 1982.

Table 1

1	Containment Bypass Sequence
2	Containment Isolation Status
3	Transient or LOCA Type
4	Reactor Shutdown
5	Station Blackout
6	Power Recovery
7	RCS Pressure at Core Damage
8	Status of In-Vessel Injection
9	Containment Fan Coolers
10	Containment Spray Status
11	Steam Generator Isolated
12	Steam Generator Break Covered
13	Mode of Induced RCS Failure
14	Dummy
15	RCS Pressure at Time of RPV Failure
16	Debris Cooled In-Vessel
17	Dummy
18	Dummy
19	In-Vessel Steam Explosion Fails Containment
20	RWST Injected Early
21	Containment Atmosphere Inert
22	Mass Debris Expelled Early
23	Fraction Debris Involved in DCH
24	Fraction Debris Dispersed Outside Lower Compartment
25	Fraction Debris Metal Reacted
26	Hydrogen Burn
27	Dummy
28	Early Containment Failure
29	Dummy
30	RWST Injected Late
31	Dummy
32	Debris Depth Against Liner
33	Containment Liner Meltthrough
34	Steam Explosion Disperses Debris
35	Dummy
36	Depth of Debris in Sump
37	Dummy
38	Dummy
39	Type of Ex-Vessel CCI
40	Dummy

Table 1 Continued

41	Dummy
42	Power Available Prior to RV Failure
43	Power Recovery Late
44	Containment Heat Removal
45	Dummy
46	Late H2 Burn Fails Containment
47	Mode of Late Containment Failure