Dr. Robert C. Mecredy
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89 East Avenue
Rochester, NY 14649

February 23, 1996

SUBJECT: GINNA PROBABILISTIC RISK ASSESSMENT PROJECT REPORT OF MARCH 15, 1994, TO THE NRC IN RESPONSE GENERIC LETTER NO. 88-20; REQUEST FOR ADDITIONAL INFORMATION (TAC NO. M74414)

Dear Dr. Mecredy:

On the basis of our ongoing review of the Ginna Individual Plant Examination (IPE) submittal of March 15, 1994, as supplemented by letter dated March 10, 1995, and its associated documentation, we have enclosed requests for additional information (RAIs). The RAIs are related to the internal event analysis in the IPE, including the human reliability analysis (HRA), and the containment performance improvement program. The RAIs were developed by our contractor and reviewed by an IPE Senior Review Board of the NRC staff.

We request that the licensee provide written responses to the enclosed RAIs within 60 days of receipt of this letter.

The information requested by this letter is within the scope of the overall burden estimated in Generic Letter 92-08 for the program, which was a maximum of 6 person-years per licensee response over a 3-year period. This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires July 31, 1997.

If you have any questions, please contact me at (301) 415-1497.

Sincerely,
ORIGINAL SIGNED BY:

Allen R. Johnson, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. RAI, Level 1 Questions
2. RAI, HRA Questions
3. RAI, Level 2 Questions

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Request for Additional Information (RAI) on Ginna
Individual Plant Examination Submittal

LEVEL 1 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 3.5E-3/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.

Enclosure 1

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.

(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., α<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 unmbers 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.

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.

(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.

(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 1B?). 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.

(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 1B failed, and how is this eventuality modeled? Can the 1A 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, 1B, and TSC batteries?

(e) It is stated that the TSC battery and/or the standby electric power spell out 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).

(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?

(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 "a loss of normal power," including power to all four safeguards buses, such that both EDGs were started and loaded onto the safeguards buses.

(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.

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?).

(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.

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.

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.

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.

(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.

(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.

(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.

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

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.

(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.

(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.

(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.

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.

(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).

(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).

(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.

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.

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.

(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.

(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.

(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.

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

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.

(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.

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

(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).

(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.

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.

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.

(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.

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.

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.

    (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.

    (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.

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.

    (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.

HUMAN RELIABILITY ANALYSIS QUESTIONS

PRE-INITIATOR HUMAN ERRORS

1. The submittal is not completely clear on the organizations that participated in the human reliability anaylsis (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.

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.

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.

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.

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 <u>level</u> 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).

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.

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

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

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.

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.

   (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 1C and 1D; (2) AFHFD04297 - operators fail to close air-operated valve 4297 to isolate steam generator (S/G) A, and (3) RCHFD01BAF - operators fail to initiate feed and bleed.

   (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.

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?

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.

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.

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. Also, 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."

   (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.

7. The submittal is unclear on what plant-specific PSFs were considered in determining HEPs for time-independent and time-dependent human 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, please provide a concise discussion of the rationale and justification for not considering such plant-specific information during the quantification of post-initiator human actions.

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.

9. On page 3.3.3-2, 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.

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 performing 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.

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 EMS01A) 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.

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 estimates 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.

    (b) A brief description of each of the recovery events modeled (apparently seven of them) and the HEPs assigned to these human actions.

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.

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 either question is yes, using examples please address how the operator actions were quantifed. 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.

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.

   (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.

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.

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.

   (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 4.1 meters (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.

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.

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 so that 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.

5. The Availability of Containment Fan Coolers and Containment Sprays. 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 containment heat removal. If they are not, please estimate the potential source term mitigation effect achieved by the operation of containment sprays.

6. Steam Generator Tube Rupture (SGTR) Releases. The IPE results show that the steam generator power-operated relief valve (PORV) cycles during and after core 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.

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.

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 percent uncertainty associated with these pressures.

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 non-containment-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.

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.

11. Containment Performance Improvement and Hydrogen Issues. The generic letter containment performance improvement recommendation 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) should cover the likelihood of local detonation and the potential for missile generation as a result of local detonation.

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.

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.

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.

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 term results for each relevant Source Term Category (STC) end state. Based on consideration of the dominant sequence for each end state and based on other factors which influence the 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.

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 their effect on the probability of induced SGTR.

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.

    (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".

    (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).