Mr. Roger O. Anderson, Director
Licensing and Management Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

December 21, 1995

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON THE PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2, INDIVIDUAL PLANT EXAMINATION SUBMITTAL (TAC NOS. M74454 and M74455)

Dear Mr. Anderson:

By letter dated March 1, 1994, Northern States Power Company submitted its individual plant examination (IPE) report in response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." During the review of your submittal, the NRC staff, with assistance from its contractor, Brookhaven National Laboratory (BNL), has determined the need for additional information. Enclosed is a request for additional information (RAI). The RAI is related to the internal event analysis in the IPE, including the human reliability analysis, and the containment performance improvement program.

The staff requests that you submit your responses to the enclosed RAI within 60 days to meet the staff's review schedule. If you have any questions regarding the content of the RAI, please contact me at (301) 415-1355.

This requirement affects nine or fewer respondents and, therefore, is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

Original Signed By:

Beth A. Wetzel, Project Manager
Project Directorate III-I
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-282/306

Enclosure: Request for Additional Information

cc w/encl: See next page

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

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Mr. Roger O. Anderson, Director
Northern States Power Company

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REQUEST FOR ADDITIONAL INFORMATION

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 & 2,

INDIVIDUAL PLANT EXAMINATION SUBMITTAL

1. It is not clear from the submittal how the cross-tied and shared systems are treated for the unit at power if the other unit is in cold shutdown and some of the shared (or potentially cross-tied) systems are experiencing extended downtime. How does the analysis account for the unavailability of the systems that are capable of being cross-tied or shared during the time the opposite unit is in shutdown? Please discuss how each shared/cross-tied system was treated in this regard. If this was not considered, please estimate the impact on your results.

2. No support system-on-support system matrix is given in the submittal. How did the analysis assure that no dependencies were missed? Please provide such a matrix.

3. The following question pertains to analysis of common cause failures (CCF) in the IPE:

   a) A review of the CCF data used in the IPE, and presented in Table 3.3-7 of the submittal, indicates that the list of components may not be comprehensive. Provide the basis for the omission of the following types of components from the common cause analysis:

      Circuit breakers (particularly for voltage ≥ 480V AC)
      Electrical switchgear
      Relays (ESFAS)

      Was the CCF analysis performed for these components? If so, please provide the CCF factors used. If not, please discuss the impact of these omissions on the core damage frequency (CDF) results.

   b) How was the common cause loss of AC buses or the common cause loss of DC buses as an initiating event treated?

4. The IPE does not consider the "loss of 120 VAC bus" as a potential initiating event (e.g., loss of panel 113 causes the direct loss of the chemical volume and control system component for the reactor coolant pump (RCP) cooling). Please provide the basis for omitting this initiating event.

5. The chilled water system is used for cooling the relay room which is common to both Unit 1 and 2. This would appear to be an important system which should be mentioned, the failure of which should be in the initiating event analysis. Please provide the basis for screening or not considering this initiating event. If it was not considered, please estimate the impact on the results.

ENCLOSURE

6. The IPE does not consider the initiator "excessive loss of coolant accident (LOCA)," i.e., reactor vessel rupture. Please provide a discussion of your consideration of this initiator and the basis for screening it.

7. The IPE model neglects the small-small LOCA initiating events. Please provide a discussion of your consideration of this initiator (which includes potential random failures of the RCP seals) and the basis for screening it.

8. The following information on initiators would be helpful:

   a) Please explain why consideration of the steam line and feed line break initiators is limited to locations inside the containment.

   b) Please discuss how the frequencies for loss of service water (SW) and loss of component cooling water (CCW) as initiators were calculated. Were pipe breaks considered in the analysis? If not, please provide the initiating event frequencies for single and dual loss of essential SW and single and dual loss of CCW, when passive component failures are also taken into account. Also provide the impact of these new frequencies on the results (CDF and dominant sequences). If the pipe break events in the CCW or the SW system have a potential for affecting the results of your flooding analysis, please update your flooding analysis and provide it for review (see also the question below).

9. The following questions concern the treatment of flooding in the IPE:

   a) Please describe your treatment of the spray effect resulting from the spurious actuation of the fire suppression equipment in your flood scenarios.

   b) Please elaborate on your statement that spray effect was not considered if it could affect only one train of equipment. Please discuss how the effect of the sprays was analyzed.

   c) Did you consider backflooding through the drains and drain failure (i.e., plugging) in developing your flood scenarios? If not, please estimate the impact of this omission on the results.

   d) Please discuss how maintenance errors committed while in cold shutdown, which were left undiagnosed until the postulated flood event occurred while the unit was at power, were treated in the flooding analysis.

   e) How does your consideration of pipe breaks in the CCW and SW systems (see question above) impact the results of the flooding analysis?

10. The status 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, i.e., whether the improvement has actually been implemented, is planned (with scheduled implementation date), or is under evaluation.

    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 were to be credited in the reported CDF (or containment failure probability), or the increase in the CDF or the 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, i.e., whether it addressed a vulnerability, was otherwise identified from the IPE review, was developed as part of other NRC rulemaking, such as the Station Blackout Rule, etc.

11. On page 1-9 of the Executive Summary of the submittal it is implied that station blackout (SBO) is a dominant contributor to the CDF, yet the SBO contribution to CDF is less than 10% of the total. Please explain the statement in the Executive Summary.

12. Please explain the process used to ensure that the model in the IPE reflected the "as-built as-operated" plant.

13. It is not clear from the submittal what the level of involvement of the utility was in constructing the probabilistic risk assessment (PRA) model. The level of depth of the in-house review is also not clear. Was only senior management involved in the review? Please describe the utility involvement in the PRA modeling as well as the in-house review in more detail.

14. This question concerns the failure and maintenance data used in the IPE:

    a) Spot checks of your plant-specific failure data revealed it to be generally lower or much lower than the generic data from NUREG/CR-4550 "Analysis of Core Damage Frequency from Internal Events." For example, failure of check valves to open and to close are 1 to 2 orders of magnitude below what was reported in NUREG/CR-4550. Please provide the basis for the calculation of your plant-specific data and verify that the low values are indeed appropriate.

b) Your failure data does not include the failure mode "check valve rupture." Was this failure mode considered in your ISLOCA [interfacing-systems LOCA] analysis, and if not, what is your estimate of the impact on your results?

15. The submittal states that success criteria for frontline systems are built upon Modular Accident Analysis Program (MAAP) runs given certain core damage criteria. These core damage criteria allow substantial time to elapse (30 min) while certain areas of the core are at elevated temperatures (1200°F). For shorter durations temperatures range up to 2000°F. Some of the success criteria calculated (and presumably used) significantly relax criteria used in other PRA work (e.g., NUREG-1150, "Reactor Risk Reference Document"). Examples are:

   (a) For large LOCA, one residual heat removal (RHR) pump is sufficient to prevent core damage, i.e., no accumulators need to inject. In previous analyses, the accumulator in the intact leg would need to inject.

   (b) For a medium LOCA of 5-inch break size, one RHR pump is sufficient to prevent core damage. Typically (e.g., NUREG/CR-4550 analysis for Surry) it is assumed that a high head injection (HPI) pump is needed as well.

   (c) For a medium LOCA of 12-inch break size, one safety injection (i.e., HPI) pump is sufficient. This would be a large LOCA size in the NUREG/CR-4550 analysis for Surry, and an LPI [low pressure injection] pump in conjunction with accumulator injection is needed.

      1) What is the basis for these novel success paths? Are they included in operator guidance (e.g., emergency operating procedures (EOPs)) and are the operators trained in them? If not, how were the human error probabilities (HEPs) quantified?

      2) Please estimate the impact of these novel success paths on your results (CDF and important core damage sequences).

      3) For small LOCAs, no credit seems to be given (short term) for power operated relief valve (PORV) manipulation to help with decay heat rejection. Is feed and bleed not an option used in the EOPs to deal with small LOCAs?

16. It is not clear in the submittal if plant changes due to the Station Blackout Rule were credited in the analysis. Please provide the following: (1) identify whether plant changes (e.g., procedures for load shedding, alternate AC power) made in response to the blackout rule were credited in the IPE and what are the specific plant changes that were credited; (2) if available, identify the total impact of these plant changes to the total plant CDF and to the SBO CDF (i.e., reduction in total plant CDF and SBO CDF); (3) if available, identify the impact of each individual plant change to the total plant CDF and to the SBO CDF

(i.e., reduction in total plant CDF and SBO CDF); (4) identify any other changes to the plant that have been implemented or are planned to be implemented, that are separate from those in response to the Station Blackout Rule, that reduce the SBO CDF; (5) identify whether the changes in #(4) are implemented or planned; (6) identify whether credit was taken for the changes in #(4) in the IPE; and (7) if available, identify the impact of the changes in #(4) to the SBO CDF.

17. There is no discussion in the submittal about the PORV block valve position and how it affects various scenarios (feed and bleed, anticipated transient without scram). What is the fraction of time that either or both block valves are closed? How is the possibility of the PORV block valve being closed accounted for in the model? If the block valves are not modeled, what is the effect on your results?

18. In consideration of Unresolved Safety Issue A-45, decay heat removal (DHR) evaluation, please provide the contribution of DHR and its constituent systems (e.g., auxiliary feedwater, feed and bleed) to CDF. There is a substantial discussion in the submittal on the various frontline systems performing the DHR function and the relative impact of loss of support systems on the frontline systems that perform the DHR function. There is also a system importance ranking output, but it doesn't include certain DHR functions such as feed and bleed, etc., nor a summary of your insights and any vulnerabilities found regarding this issue. Please provide this information.

19. In many PRAs, RCP seal LOCA is a significant contributor to the CDF either as an initiating event or as a system failure consequential to another initiator. While the submittal discusses RCP seal LOCA consideration, please provide the additional information requested:

    a) Please provide a discussion of the RCP seal LOCA model used. Include the probability vs. leakage rate vs. time data and any specific test results.

    b) Provide a discussion of operator actions which are proceduralized and their timing in the event of a loss of one or the other method of RCP seal cooling.

    c) Please provide an estimate of the impact of your assumptions regarding the RCP seal LOCA model on your results (CDF, significant sequences, system importance measures, and important operator actions).

Prairie Island Human Reliability Analysis (HRA) RAI

1. In Section 3.3.5 the submittal mentions that human errors such as incorrect calibration of sensors or instruments were included as explicit events in system fault trees as was failure to restore components to service after their isolation for maintenance.

a) Please provide a list of the types of pre-initiator human events in order of importance considered in the analysis.

b) Since the submittal does include pre-initiator human actions, it is important to describe the process used to identify and select the pre-initiators involving miscalibration of instrumentation and the failure to properly restore equipment to service after test or maintenance. The process used to identify and select instrumentation calibration related human action events may include the review of calibration procedures and discussions with appropriate plant personnel on interpretation and implementation of the plant's calibration procedures. For assessing the failure to restore equipment to service after test or maintenance, the process 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 provide a description of the process that was used to identify pre-initiator human actions involving miscalibration of instrumentation and failure to restore equipment to service after test or maintenance. In addition, please provide examples illustrating the processes using several relatively important pre-initiator human actions.

2. The submittal does not provide all the screening values used for pre-initiator human events or the bases for the values provided. Screening values for some of the typical operator actions are given in Table 3.3-10.

   a) Please provide all of the screening value(s) used and the basis for the value(s); i.e., provide the rationale of how the selected screening value(s) did not eliminate (or truncate) important pre-initiator human events.

   b) In addition, please provide the list of all pre-initiator human actions initially considered and all those screened.

3. The submittal does not clearly identify actual recovery factors applied in quantifying the pre-initiator human events. Factors that are used to modify the generic basic human error probabilities (BHEP) can include, for example, post-maintenance or post-calibration tests, daily written checks, independent written verification checks, administrative controls, etc. If they were used, please provide a list of pre-initiator recovery factors considered, their associated values, and provide specific examples illustrating their use. Also, if used, please provide a concise discussion of the justification and process that was used to determine the appropriateness of the recovery factors utilized.

4. It is not clear from the submittal how dependencies associated with pre-initiator human errors 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

is require to 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 different operators' abilities 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 they fail simultaneously. In the third example, pressure sensor "x" and "y" may be calibrated using different procedures. However, if the procedures are poorly written such that miscalibration is likely on both sensor "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. Section 3.3.4 of the submittal states the following human dependency related information, "the cutsets were reviewed after sequence quantification and when more than one human error appeared in the same cutset, either independence of the human actions was confirmed, or a change was made to correctly model dependence between the human errors." Please provide a concise discussion of how dependencies were addressed and treated in the pre-initiator HRA such that important accident sequences were not eliminated. If dependencies were not addressed, please justify.

5. The submittal does not clearly describe the type of human errors considered for each post-initiator human event identified. For example, a human event identified may be the failure to feed and bleed, while the types of human errors considered may involve failure to open the correct valve (error of omission), or opening an incorrect valve (error of commission). No mention of types of human errors was found in the submittal's Section 3.3.4. Please identify what types of human errors were considered for each post-initiator human event identified.

6. The submittal does not clearly describe the method used to identify and select response type actions and recovery type actions for analysis. The method utilized should confirm the plant emergency procedures, design, operations, and maintenance and surveillance procedures were examined and understood to identify potential severe accident sequences. The submittal is not clear on the identity of the response type actions and recovery type actions used (see request below). Also, the method used was not addressed. Please provide a description of the process that was used for identifying and selecting the response and recovery type actions evaluated.

7. It is not clear from the submittal what screening values were used for post-initiator human actions and the bases for the values.

a) Please provide the screening value(s) used and the basis for the value(s), i.e., provide the rationale of how the selected screening value did not eliminate (or truncate) important post-initiator human events.

b) Also please provide the list of all post-initiator human actions initially considered and all those screened.

8. In applying performance shaping factors (PSFs), the consideration of time is important. The submittal is not clear on how available time and "required" time were calculated for the various post-initiator human events. "Required" time is the time needed for the operator to detect, diagnose, and perform the necessary actions. Section 3.3.4 of the submittal and Tables 3.3-5 and 3.4-6 provide a "diagnosis time" but no available time or "required" time. For several of the important post-initiator human events examined, provide the available and "required" times estimated for the operator action and the bases (e.g., calculated from simulator exercises, estimated from walkdowns) for the time chosen. Also provide illustrations of how different times were calculated for the same task but in different sequences.

9. It is not clear from the submittal what plant-specific PSFs were used to modify the BHEP and what the bases were for reducing the HEPs through their application. The plant-specific information could include the size of crew, availability of procedures, time available and time required, etc. The process could include an examination of procedures, training, human engineering, staffing, communication, and administrative controls. No mention of plant-specific PSFs were found in the submittal. Please provide a list of the types of plant-specific PSFs considered and their values, and discuss by way of example how these PSFs were used to modify the BHEPs of important post-initiator human events.

10. The submittal is not clear whether response type actions and recovery type actions were considered. Response type actions include human actions performed in response to the first level directive of the EOPs. For example, suppose the EOP directive instructs the operator to determine reactor water level status, and another directive instructs the operator to maintain reactor water level with system X. These actions - reading instrumentation to determine level and actuating system X to maintain level - are response type actions. Recovery type actions include those performed to recover a specific failure or fault and may not be "proceduralized." For example, suppose the EOP directive instructs the operator to maintain level using system X, but the system fails to function and the operator then attempts to recover it. This action - diagnosing the failure and then deciding on a course of action to "recover" the failed system - is a recovery type action. The submittal is not clear on the identity of the response and recovery actions. Please provide separate lists of the response and recovery actions considered in the analysis. If response or recovery actions were not considered, please justify. If response and recovery actions are used, are they proceduralized? If not, please justify any credit taken for such actions.

11. 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. Section 3.3.4 of the submittal provides the following human dependency related information, "The cutsets were reviewed after sequence quantification and when more than one human error appeared in the same outset, either independence of the human actions was confirmed, or a change was made to correctly model dependence between the human errors." Please provide a concise discussion and examples illustrating how dependencies were addressed and treated in the post-initiator HRA such that important accident sequences were not eliminated. There are several ways post-initiator dependencies can be treated, namely, modeled in fault trees and event trees. If the submittal did not address dependencies in the quantification, please justify. The discussion should address the two models below:

    Human events are modeled in the fault trees as basic events such as failure to manually actuate. The probability of the operator to perform this function is dependent on the accident in progression - what symptoms are occurring, what other activities are being performed (successfully and unsuccessfully), etc. 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 are quantified which should be evaluated for dependencies.

    Human events are modeled in the event trees as top events. The probability of the operator to perform 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.

12. Please discuss the process used to assure that key HRA assumptions about operator actions, information available to operators, plant environment, etc., represent the conditions in the as-built, as-operated plant. In particular, please discuss information related to interviews with operators and plant walkdowns.

13. Please provide specific information describing the process used to assess the use of symptom-based procedures in the current plant. The information should focus specifically on justification of assumptions used in the HRA modeling.

14. As requested in NUREG-1335 "Individual Plant Examination: Submittal Guidance", please provide a listing and a discussion of any sequences that drop below the applicable core damage screening criteria because

the frequency has been reduced by more than an order of magnitude by credit taken for human recovery actions (not to exceed 50 of the most significant sequences).

15. The submittal is not clear if the need to diagnose an event (i.e., to figure out what is to be done in any given situation) was considered in the HRA analysis. The diagnosis in NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," includes the actions to "perceive, discriminate, interpret, diagnose" an event and the operators "first-level of decision making." While using symptom-based emergency operating procedures (EOPs) removes the need to identify the type of accident, such as a LOCA, their use does not remove the need for other aspects of diagnosis. Please discuss how diagnosis was considered in your analysis. If it was not considered, please justify this omission.

Prairie Island Level 2 RAI

1. According to the IPE, the containment water level will be more than 7 feet above the bottom of the reactor vessel if the RWST [refueling water storage tank] content is injected to the containment, either through core injection and subsequent condensation or through containment spray. This is several feet higher than the depth of the debris inside the vessel if all of the core material were to slump to the bottom of the vessel, assuring that any portion of the vessel wall in contact with the debris can transfer heat directly to coolant in which the vessel is submerged. In-vessel recovery by this heat removal mechanism is considered in the containment event tree (CET) as one of the top events and its effect on CET quantification is evaluated in the sensitivity studies. However, its effect on source term definition is not discussed in the IPE submittal. Since this mechanism may terminate or delay vessel penetration, fission product production and release path are consequently affected (e.g., in-vessel release from a dry debris bed versus ex-vessel release from a debris bed covered by water). Please discuss the effect of external vessel cooling on source term definition. Please also discuss whether this mechanism is included in the MAAP model used for the base case source term analysis, and if not, then please discuss how the source terms are modified by the inclusion of this mechanism.

2. The front-to-back end interfaces are provided in the IPE by the definition of 14 accident classes. These accident classes are identified by a three-character designator addressing the following parameters: the accident initiator, core melt timing, and reactor pressure at the time of core melt. The availability of containment systems (e.g., containment fan coil units and containment spray) are not explicitly included in the definition of the accident classes. Since each accident class may include many core damage sequences (or cutsets), the availability of containment systems may not be the same for all the core damage sequences in an accident class. Please provide a more detailed discussion of how the availability of containment systems is

determined and how this information is used in CET quantification for the various accident classes. Please illustrate this process with a few examples.

3. The CETs used for the Prairie Island Level 2 analysis were discussed in Section 4.5 and the results of Level 2 sequence quantification were presented in Section 4.6 of the submittal. Although the top events of the CETs, the CET end states, and the dominant sequences were discussed in relative detail in these sections, the discussions are of qualitative nature and the quantitative values used for the CET branches were not presented. Please provide the probability values allocated to each of the CET branches and discuss the basis for these values. Please include in the discussion the basis for the values used for human actions as well as how the availability and survivability of systems and components with potentially significant impact on the CET or the radionuclide release were considered.

4. 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)." Although the materials presented in the IPE submittal cover most of the above areas, some of the items in the above list are not addressed. Please discuss your findings related to all of the above five areas.

5. It is assumed in the IPE that during a high pressure vessel blowdown, a significant amount of core debris is carried out of the reactor cavity, through the instrument tunnel, to the upper compartment. Because Prairie Island uses a steel containment and the seal table is situated outside the secondary shield wall, a high pressure vessel blowdown could lead to corium coming into contact with the containment steel shell. This failure mode is discussed briefly and dismissed as a potential failure mode in the IPE. Please provide a more detailed discussion of the analytical model used to determine the flow paths and distribution of the discharged debris during a high pressure melt ejection. Please discuss the impact of the two personnel entry hatches on corium dispersal and disposition. According to the IPE, these two hatches are located on the instrument tunnel and are left slightly ajar during normal operation.

6. In most of the temperature-induced steam generator tube rupture (SGTR) failures reported in the submittal, the valves which open to relieve the steam generator pressure are assumed to reclose successfully. This limits the release to a relatively short duration puff, followed by a series of shorter puffs, and all releases are terminated upon vessel failure when the primary system depressurizes to containment pressure.

Please discuss how the probability of steam generator valve failure is determined in the analysis and whether the harsh operating conditions (e.g., the flow of extreme high temperature gases with entrained debris) is considered in the analysis.

7. Table 4.6-1 shows the frequencies of the dominant CET sequences that contribute to the CET end states (i.e., containment failure modes). This provides partial information on the conditional probabilities of the failure modes for an accident class (or plant damage state). Please provide the C-Matrix which provides a complete account of the conditional probabilities of the failure modes for all accident classes evaluated in the Level 2 analysis. Since the probability of temperature-induced SGTR is excluded in the calculation of the probability values presented in Table 4.6-1, please also provide the C-Matrix with the temperature-induced SGTR included in the evaluation.

8. The plant data that are of interest to the Level 2 analysis are provided in Section 4.1 of the IPE submittal. Although this section provides the essential data for the accident progression discussion, it lacks the detail suggested in NUREG-1335. Please provide in tabular form the data described in Table A.1 of NUREG-1335.

9. The effects of harsh environmental condition on the operation of containment sprays and containment fan cooler units are not discussed in the CET quantification of the IPE submittal. Please discuss the survivability of these components under severe accident conditions. Please include in the discussion the environmental conditions (e.g., temperature, pressure, radiation, aerosol plugging and debris effects) derived and used in the evaluation.

10. 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 walkdowns have been performed to determine the probable locations of hydrogen releases into the containment. 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 identify potential reactor hydrogen release points and vent paths. Estimates of compartment free volumes and vent path flow areas should also be provided. Please 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.

11. It is assumed in the IPE that containment spray is required to cool the debris that has been relocated out of the reactor cavity to the upper areas of the containment following a high pressure melt ejection. The results of the sensitivity studies presented in the submittal show that the probability of late containment failure increases significantly (from 21 percent to 63 percent) if the relocated debris is not coolable. A similar change in containment failure probability is expected if containment spray is not available for all accident sequences. Please discuss whether containment spray in recirculation mode is required to prevent containment failure in the cases with relocated debris, and discuss how the data for spray availability is derived in the IPE. Please discuss the effect of maintenance schedules and harsh environmental conditions on the availability and continuous operation of the containment spray.