Mr. Ross P. Barkhurst
Vice President Operations
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

January 22, 1996

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON WATERFORD 3 STEAM ELECTRIC STATION INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL (TAC NO. M74487)

Dear Mr. Barkhurst:

Based on our ongoing review of the Waterford 3 IPE submittal 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 accident sequence core damage frequency analysis, the human reliability analysis, and the containment performance analysis.

We request that you provide written responses to the RAIs within 60 days from the date of this letter in conformance with our review schedule. Although we have requested additional information, the human reliability analysis (HRA) portion of the IPE submittal, in particular, is weak. To help ensure that your responses to our RAI address our concerns, we would like to have a discussion on the telephone and determine if a visit to the Waterford 3 site is necessary to discuss the HRA RAIs. We would like to discuss this with your staff as soon as you have had the opportunity to review our RAIs.

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

Sincerely,

Original signed by:
Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure: Request For Addditional Information

cc w/encl: See next page

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Document Name: WAT74487.RAI

OFC	LA/PD4-1	PM/PD4-1
NAME	PNoonan	CPatel/vw
DATE	1/22/96	1/22/96
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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 22, 1996

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Entergy Operations, Inc.
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Waterford 3

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Waterford 3 Steam Electric Station IPE
Request for Additional Information (RAI)

Level 1 Questions

1. 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 was 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.

   (f) Please discuss the potential improvement of using the LPSI pumps for containment sprays in light of a statement made in the submittal that LPSI pumps are not used in recirculation, even though the hardware arrangement for this exists (HPSI and LPSI pumps share a common recirculation sump suction header and LPSI is connected to the shutdown cooling heat exchangers, used by the sprays in recirculation mode). According to the submittal, even if the LPSI system is used in the injection mode, the recirculation actuation system will stop these pumps upon switchover to recirculation, and the HPSI pumps must be aligned to the recirculation sump. Would not the same obstacles that prevent LPSI use for RCS recirculation preclude using the LPSI pumps in the containment spray mode (i.e., during recirculation)?

2. The value of 0.030/yr (with an error factor of 1.33) for the loss of offsite power (LOSP) initiating event frequency is at the low range of typical LOSP frequency values as, for example, industry data from NSAC-147. Furthermore, LOSP is the most dominant contributor to the Waterford 3 CDF risk, contributing 45% to the total core damage frequency from internal events. Therefore the LOSP frequency will directly influence a major portion of the results. While there may not have been losses of offsite power in the 4 year plant operating history reviewed for the initiator data base, the plant is situated in an area where severe weather occurs relatively frequently.

ENCLOSURE

(a) Please explain why the use of generic frequency for weather related losses of offsite power in your model (which accounts for about 1/3 of the LOSP frequency used) is appropriate. Also include a discussion of plant specific grid-related losses of offsite power. Include any available data on losses of offsite power in your area, at your site and on your grid.

(b) Please show a derivation of the error factor used.

(c) If an adjustment in LOSP frequency is necessary, please provide an assessment of the impact on your results, including important sequences, the total CDF and the CDF contributions from initiators and sequences.

(d) Please provide the offsite power recovery curve or data (probability of non-recovery vs. time) which was used in the model.

3. According to the submittal, small LOCA is the second most dominant initiator (after LOSP) in terms of the total internal CDF contributing about a third to the total CDF, as well as important core damage sequences. Therefore, the number used for the small LOCA initiating event frequency is important.

The small LOCA initiating event frequency seems low (4.5E-3/yr). This apparently includes what is traditionally known as small LOCA and very small LOCA. It seems that spurious RCP seal failures were not considered as a credible mechanism for having a small LOCA. An explanation is provided that the Byron Jackson design is very sturdy and that there have been no RCP seal failures at any Combustion Engineering plants. However, NUREG/CR-4550 references several events in which a spurious RCP seal failure occurred, at least one of which was in a Byron-Jackson RCP seal. This was the ANO-1 event of 1980, which had a leak rate of 400 gpm. While giving credit for improvement of the seals over the years, and including very small pipe and other component failures, NUREG/CR-4550 arrives at a generic very small break LOCA frequency of 1.3E-2/yr. In addition, the small LOCA frequency quoted in that document is on the order of 1.E-3 (some of which is inapplicable to Waterford 3 due to a lack of pressurizer PORVs). Thus, the frequency used in the submittal is about 1/3 of that recommended in NUREG/CR-4550.

Therefore, please provide details of your derivation of the small LOCA frequency, including its constituent parts: pipe breaks, component leakages, and RCP seal failures. Include sufficient detail for an understanding of the basis for your initiating event frequency number. If an adjustment of your small break LOCA frequency number is necessary, please provide an estimate on the impact on your results, including important sequences, total CDF and CDF contributors.

4. Why is the loss of a 4.2 Kv non-safety bus not considered as an initiator? In Appendix B, it is stated that 4.2 Kv systems are required during normal plant operations, thus implying that a loss of a 4.2 Kv bus would result in a reactor trip. In addition, according to the 4.2 Kv schematic in Appendix B, a loss of a 4.2 Kv non-safety bus would also fail the associated 4.2 Kv safety bus (with a recovery from the diesel

generator possible, but no recovery from the startup transformer), and would thus have a bigger impact (i.e., larger conditional core damage probability) than a loss of the 6.9 Kv bus which was considered. If this initiator should be considered, please provide an estimate on the impact on your results including the core damage frequency and important sequences.

5. In the Appendix B description of the PPCS system (pressurizer pressure control system), it is stated that this system can experience failures that result in uncontrolled increases in pressurizer pressure or level. However, no initiators originating from this system were apparently considered. Please provide a justification why such initiators were screened out, and if an adjustment is necessary please provide an estimate on the impact on your results, including core damage frequency and important sequences.

6. In the submittal, it is stated that the reactor vessel rupture initiator would have a negligible contribution and would thus be screened out. Please provide your estimate of the frequency of the RV rupture initiator. Discuss the basis for your estimate and the data used.

7. In the discussion of derivation of the initiating event frequencies, it is not clear which method was used for which initiator (other than the few specifically mentioned, e.g., reactor trip, turbine trip). For example, were plant specific fault trees with generic data used for estimating a dc bus loss frequency or was a generic frequency used for this initiator? Or, how did the expert panel arrive at the large LOCA frequency? Please provide this information for all the initiators.

8. 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:

   (a) 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;

   (b) if available, identify the total impact of these plant changes to the total plant core damage frequency and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF);

   (c) if available, identify the impact of each individual plant change to the total plant core damage frequency and to the station blackout CDF (i.e., reduction in total plant CDF and station blackout CDF);

   (d) identify any other changes to the plant that have been implemented, or which are planned to be implemented, that are separate from those in response to the station blackout rule, that reduce the station blackout CDF;

   (e) identify whether the changes in (d) are implemented or planned;

   (f) identify whether credit was taken in the IPE for the changes in (d); and

(g) if available, identify the impact of the changes in (d) to the station blackout CDF.

9. This question concerns the treatment of flooding:

(a) The submittal indicates that all but a few flood zones were eliminated from further consideration through a qualitative screening analysis. Table 3.6-1 shows some flood zones with a "high" source rating and a high source weight having a very small conditional core damage probability (< 1.E-4), implying absence of significant safety equipment in these rooms. It is stated that in the screening analysis all PRA-related equipment contained within a room is assumed failed. Please show the safety equipment contained in each such zone (e.g., those with a flood frequency greater than 1.E-3/yr).

(b) Please discuss your consideration of drains (including back flooding to other areas and probability of failure, i.e. due to blockage), and doors allowing flood propagation to other areas. As the fire zones are used for delineation of flood zones, discuss whether all fire doors are water proof in Waterford 3, and whether the failure of such doors to be in a closed position is accounted for in the model.

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

(d) Please discuss the operator actions needed for isolation and mitigation of the most important flood scenarios and provide the basis for the flood affected HEPs used. Include a discussion of any alarms or any other means the operators would use to detect and stop the flood in the 20 minutes available.

(e) The IPE assumed in development of the flood scenario frequencies that no flood would propagate for longer than 20 minutes. However, the HEP used for flood isolation in that time appears to be 0.01. That means that 1% of the floods would propagate beyond 20 minutes, presumably failing more safety equipment and having a high conditional core damage probability. Please discuss if such scenarios were considered and what was the estimated impact on the flooding scenario results.

(f) Discuss how maintenance errors were treated in the flooding analysis. Include errors committed while in cold shutdown, which are left undiagnosed until the flood event occurs while the unit is at power.

(g) There are many screened scenarios whose CDF contribution is below the 1.E-6/yr screening criterion. Please provide an estimate of the total contribution of the screened scenarios to the core damage frequency.

10. This question concerns the use of failure data:

    (a) Generic failure data were used for all components except for the diesel generators, which used a start failure rate that is about a factor of 2 lower than that used in NUREG/CR-4550. Please explain why you have plant specific data for diesels and no other equipment (implying a relatively high failure rate of the diesels), yet the plant specific start failure rate of the diesels is lower than the generic failure rate, implying a relatively low failure rate. Since ac power is the most important system in the plant, please provide a discussion as to how the plant specific failure data for the diesel generators was obtained, including plant specific data on number of failures and number of tests.

    (b) The generic data used in the IPE for turbine driven pump failure to run is about two orders of magnitude below that in NUREG/CR-4550. The turbine driven pumps are important, and are used for decay heat removal, both via the MFW system and via the EFW system. The IPE results could be significantly affected by the data used for the turbine driven pumps. Please provide the basis, the source and the derivation of such a low number. If such a justification cannot be made, please provide an estimate of the impact on the dominant sequences and the core damage frequency if the NUREG/CR-4550 number were used.

    (c) The error factors on the maintenance unavailability data are very small. This is plant specific data, and for a plant with a limited operating experience one would expect wider uncertainty considering maintenance downtime. Please discuss how the error factors were derived.

11. The description of HVAC and its modeling is unclear. Apparently there is a central chiller which supplies chilled water to individual room air handling units (although the submittal only talks about the "fans"). As an example of the lack of clarity, it is not clear if switchgear cooling in the cable vault and switchgear room is needed.

    It is also not clear what level of redundancy exists in the HVAC system (e.g. in the central chillers, or in the individual room fan units), how the system was modeled in the fault trees, which fault trees it was modeled in and what failure modes were accounted for. It is not clear why any common cause failures (in the chillers, fans, etc.) were not considered.

    Please describe the investigation performed on the impact of HVAC to the rooms containing safety related equipment. Specifically address the equipment sensitive to temperature change, where that equipment is located, methods of assessment, credits for operator actions, any indications in the control room, timing, temporary equipment, and rationale for elimination as an initiating event or as support to specific equipment. Please ensure that your discussion includes clarification of the examples cited above.

12. The following question pertains to the analysis of common cause failures:

(a) The common-cause failure data used in the plant model is listed in Table 3.3-5 of the submittal. A review of the listed components indicates that the list may not be comprehensive; e.g., the following types of components are missing:

Circuit breakers	Relays (ESFAS)
Electrical switchgear	Transmitters
Air operated valves	Switches
Check valves	HVAC Chillers
Pressurizer Safety valves	Solenoid valves
Fan cooler units	All three EFW pumps (possibly also including the AFW pump)
Ventilation fans (e.g. EDG)	
Air compressors	
Inverters	

Please consider that there could be common cause failures of all three EFW pumps (i.e., pumps only without the drivers). The fourth pump, the manually started AFW pump, might also be affected. This might be due to steam binding or other causes.

Please discuss the impact of these omissions on the CDF results and important sequences. Please include a discussion how you ensured that potential vulnerabilities were not overlooked, especially in view of your definition of vulnerabilities, which includes "common cause failures with an unusual and significant effect on the core damage frequency." There has been historical experience with common cause failures of the above listed components. Please show that you have looked at design, maintenance and operation of these components and that you do not have a potential vulnerability in these areas.

(b) Also show the CCF factor for batteries used (not shown in the submittal) and how it was derived.

(c) Please clarify if the beta factor for the safety injection pump (event MV $) in Table 3.3-1, was also used for all other pumps (e.g., EFW pumps, CCW pumps, etc.). If not, provide all the CCF factors used.

13. NUREG-1335, Section 2.1.6 part 4 requests "a thorough discussion of the evaluation of the decay heat removal function." Section 3.4.3, Decay Heat Removal Evaluation, deals with this issue. Please provide the contribution of DHR and its constituent systems to core damage frequency and the relative impact of loss of support systems on the frontline systems that perform the DHR function.

14. It is stated in the submittal that Waterford 3 batteries have a 4-hour life under SBO conditions, assuming that operators shed the nonessential loads. Without shedding those loads the estimated life is only one hour. The sensitivity analysis shows that the CDF results are very sensitive to battery life (when it is increased to 6 and 8 hours). However, the operator action to shed the loads is not modeled. The operators have little time to accomplish this action, they would be

preoccupied with EDG recovery, monitoring the TDEFW system, etc. Furthermore, SBO sequences are dominant contributors to core damage at your plant. Your justification that this will be offset by no credit given for EDG recovery appears inconsistent, as the two actions are not related. Please consider that the IPE process is supposed to search for potential vulnerabilities at your plant.

(a) In view of the above, please provide the basis for not considering this action, and the impact on important sequences and the CDF results if it is considered.

(b) Please clarify the disposition of the reviewers' comment that the TDEFW pump could operate with low quality steam or even water at its inlet. Was any credit taken for such operation, and if so, provide the basis?

15. Table 3.7-7, dominant initiators and their CDF contribution, doesn't show T9, loss of CCW. Yet this system has a relatively high initiating event frequency (4.5E-3/yr) and is used to cool the diesel generators, HPSI pumps, etc. According to Appendix B, one in 10 dead bus transfers (which must be accomplished after each initiator) will fail. As there are two 4 Kv buses, and both of them have to have a dead bus transfer failure upon trip, the contribution of CCW to the station blackout initiating event frequency is 9.E-5, i.e. it is a dominant contributor to the SBO initiating event frequency (this assumes failure of diesel generators upon loss of CCW, as implied by Appendix B). Losses of CCW leading to SBO, in conjunction with failure of the TDEFW pump to start would contribute 2.3E-6/yr to the CDF, i.e. it would be a dominant contributor, and thus should appear in Table 3.7-7.

Please show your consideration of this scenario, the basis for elimination and any impact on the dominant sequences and the results.

16. It is stated in the submittal that walkdowns of the plant were used for the flooding analysis. Please clarify if walkdowns were used for any other aspect of the analysis (e.g., common cause, environmental factors, etc.), and identify those aspects. If walkdowns were not performed, please explain how it was assured that the "as built as operated" plant was modeled.

17. Please clarify the extent and nature of utility participation in conducting the flooding analysis.

18. Please clarify if Waterford 3 has the "alternate feed and bleed" capability found at some CE plants, and if, and how, this feature was modeled.

19. Your large LOCA initiating frequency number seems low (5.E-5/yr) compared to such studies as NUREG/CR-4550, which used an order of magnitude higher frequency.

Therefore, please provide details of your derivation of the large LOCA frequency. Include sufficient detail for an understanding of the basis for your initiating event frequency number. If an adjustment of your large break LOCA frequency number is necessary, please provide an estimate on the impact on your results, including important sequences, total CDF and CDF contributors.

Human Reliability Analysis (HRA) QUESTIONS

PRE-INITIATOR HUMAN ERRORS

1. The submittal is not completely clear on the organizations that participated in the HRA portion of the analysis. Please clarify the extent to which the HRA was performed by licensee 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 is unclear on how miscalibration errors were selected. On page 3.2-1 the submittal states that operator errors were incorporated into the system fault trees where appropriate. On page 3.4-1 the submittal states that the types of human failure events (HFEs) modeled ... were identified according to standard industry classification schemes. The submittal does not clearly discuss the process that was used to identify and select pre-initiator 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 provide a description of the process that was used to identify human events involving miscalibration of instrumentation. Please provide examples illustrating this process.

3. The submittal is unclear on how failure to restore errors were selected. On page 3.2-1 the submittal states that operator errors were incorporated into the system fault trees where appropriate. On page 3.4-1 the submittal states that the types of human failure events modeled ... were identified according to standard industry classification schemes. The submittal does not clearly discuss the process used to identify and select pre-initiator human 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 provide a description of 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 3.4.3, on page 3.4-3, the submittal provides the screening value (0.003 with a 0.1 beta factor) used for pre-initiator human failure events and the stated basis for the screening value (0.003).

    (a) However, a discussion of the basis for the beta factor is not provided. Please provide the rationale for the choice of the value used for the beta factor and provide examples of how the beta factor was applied. In providing the examples, take specific HFEs and show how, where, and why failure probabilities were adjusted with the beta factor.

(b) The submittal also states that the screening value of 0.003 is intended to be a nominal estimate. In Section 3.7.5, on page 3.7-10, the submittal indicates that in performing an HRA sensitivity study, the PRA model was requantified with all operator *response* and recovery error frequencies set to 0.1. Did this requantification include pre-initiator human failure events. If not, please provide a rationale for how the selected screening value(s) ensured that important pre-initiator human events were not eliminated and/or important sequences truncated. In addition, please provide the list of the pre-initiator human failure events which were initially considered, but which were eventually screened-out.

5. The submittal is unclear on how the "time-independent" quantification technique was applied to those pre-initiator human failure events surviving initial sequence quantification. Beginning on page 3.4-4, the submittal states that for those pre-initiator human events that survived screening a "time-independent" quantification technique was used to generate updated estimates for the human events. The submittal then goes on to present the parameters included in this quantification technique. Please provide the following regarding the "time-independent" quantification technique:

(a) The basis for the parameters included in the technique and a discussion of why this set of parameters is assumed to be sufficient?

(b) The possible numerical values for parameters 1, 2, and 3 (as listed on page 3.4-4 of the submittal) and a discussion of how the numerical values would be chosen.

(c) A listing of the performance shaping factors (PSFs) considered in parameter 4.

(d) A discussion of the process whereby the performance shaping factors in parameter 4 were selected.

(e) A discussion of how the PSFs in parameter 4 would be applied in determining a human failure probability and a listing of their associated numerical values.

(f) Specific examples of the application of the technique that exercise <u>all</u> parameters in the technique as determined by events analyzed during the performance of the IPE. The examples provided should clearly illustrate the application of PSFs and also illustrate how the derived human failure probabilities reflect plant-specific characteristics. For example, the illustrations could explain how examinations of procedures, walkthrough of procedures, or interviews with plant personnel were considered in determining human failure probabilities.

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 is required 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. Please provide the following 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, please justify.

POST-INITIATOR HUMAN ERRORS

7. The submittal is unclear on the quantitative screening approach used for post-initiator human failure events. On page 2.3-5, the submittal states that screening data were used to allow the risk model to identify important human actions. Furthermore, on page 3.5-4, the submittal states that human failure events that have a significant impact on the core melt frequency of the sequence were analyzed further since they were initially set at screening values. It is not clear from the submittal what screening values were used and the basis for the values. Please provide:

(a) The screening value(s) used and the basis for the value(s); that is, 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.

(b) In addition, please provide the list of the post-initiator human failure events which were initially considered, but which were eventually screened-out.

8. The submittal is unclear on how the "time-independent" quantification technique was applied to those post-initiator human events surviving initial sequence quantification. Please see question number 5. The same information is requested here as in question number 5, but in regard to the quantification of post-initiator human events. In addition to answering items (a) thru (f) from question 5, please also provide the following:

   (g) Discuss any differences in the PSFs considered for pre- and post-initiator events when using the time-independent technique. If different PSFs were not considered, please justify how the same PSFs would be relevant to both pre- and post-initiator human failure events.

   (h) 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. In particular, discuss and illustrate with examples how the diagnosis portion of human failure events is considered in determining human failure probabilities with the time independent technique. If diagnosis and associated PSFs were not explicitly considered, please provide a justification for how the values obtained with the time- independent technique accurately reflect human failure probability.

9. The submittal is unclear on how the "time-dependent" quantification technique was applied to those post-initiator human events surviving initial sequence quantification. Beginning on page 3.4-4, the submittal states that for those post-initiator human events that survived screening, two "time-dependent" quantification techniques were used to generate updated estimates for the human events depending on whether the event was an in-control room action or an ex-control room action. The submittal then goes on to present the parameters included for these quantification techniques. Please provide for each of the time-dependent models:

   (a) The basis for the parameters included in these models and a discussion as to why the selected parameters are relevant.

   (b) Where appropriate, the possible numerical values for each of the parameters.

   (c) A discussion of how the numerical values would be chosen.

   (d) A discussion and listing of the PSFs applied in the techniques and a discussion of the process used to determine the appropriateness of applying the various performance shaping factors.

   (e) Specific examples of the application of each of the two techniques that exercise all parameters in the techniques as determined by events analyzed during the performance of the IPE. The examples provided should justify why the human failure probabilities should be reduced through the application of plant-specific performance shaping factors. This process could include examination of procedures, training, human engineering, staffing, communication, and administrative controls.

(f) For one of the examples, address event "OPER-6", which is the operator action to stop the RCPs within 30 minutes of loss of seal cooling. This event has an human failure probability of 5.2E-5, which is about a 100 times lower than values used for the same event in other IPEs. Please provide a full description of the derivation of the human failure probability for this event.

(g) For another example address operator actions ZMANTRAN and ZMANTRAN2. These appear to be similar actions which have dramatically different human failure probabilities. Please discuss these events and the differences in their derivation. These events may also be good examples for illustrating the treatment of dependencies. See question 13 below.

(h) 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 techniques. In particular, discuss and illustrate with examples how the diagnosis portion of human failure events is considered in determining human failure probabilities with the time dependent technique. If diagnosis and associated PSFs were not explicitly considered, please provide a justification for how the values obtained with the time - dependent technique accurately reflect human failure probability.

10. The submittal is unclear on how "available" time was determined. In applying performance shaping factors, the consideration of time is important. The submittal is not clear on how "available" time was calculated for the various post-initiator human events. For each of the post-initiator human events examined, provide:

(a) The available time estimated for the operator action and the bases for the time chosen.

(b) For several cases, provide examples illustrating how different times were calculated for the same task, but in different sequences.

11. The submittal is unclear on how and what "other times as human factors considerations require" (page 3.4-5) was determined. In applying performance shaping factors, the consideration of time is important. The submittal is not clear on how the time required for operators to conduct actions or "other times as human factors considerations require" were calculated for the various post-initiator human events. For

example, were times calculated from simulator exercises or from walkdowns? For each post-initiator human event examined, provide the time available for the operator to diagnose and the time needed to perform the actions and the bases for the time chosen. Also, discuss whether the arrival times for cues relevant to operator decisions were considered. Even though a particular event has occurred, the operators may not get any indication of the event for a period of time. Was this considered in determining human failure probabilities with the time dependent techniques? Please provide examples which illustrate consideration of cue arrival times in determining human failure probabilities or provide a justification for why they were not considered.

12. The submittal is unclear on how recovery actions were quantified. On page 3.5-5, the submittal states that the probabilities for failure to take or perform recovery actions were developed using SAIC's HRA techniques. Please describe these techniques and provide examples that illustrate all aspects of the technique corresponding to the recovery events modeled in the IPE. In addition, please provide the following:

    (a) List the recovery events and identify any operator recovery actions credited for which written procedures did not exist. For actions not covered by procedures, please provide a justification for the credit taken.

    (b) Please describe and discuss any cut sets in which more than one recovery action was applied.

13. The submittal is unclear on how dependencies were addressed. On page 3.4-5, the submittal indicates that interpersonal dependencies are modeled explicitly in the PRA model and that factors related to the conditions of the scenarios are accounted for in the performance shaping factors. 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 points 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 dependent effects.

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 need to consider the different sequences and the other human events.

14. The submittal is unclear on how human actions during flooding scenarios were identified. Please describe how human actions were identified as part of the flooding analysis.

15. The submittal is unclear on how in-control room actions for flooding scenarios were screened. On page 3.6-7, the submittal states that all ex-control room actions were failed and in-control room actions for flood scenarios that started in or propagated through the control room were failed. If applicable, please describe the methodology to quantify in-control room actions for flood scenarios that did not start in or propagate through the control room.

16. The submittal is unclear on the quantification of human events that survived the screening analysis. On page 3.6-11, the submittal states that detailed analysis of control room floods were done differently than non control room floods. On page 3.6-9, the submittal describes the approach used to quantify human actions for non control room floods. Please describe the following:

    (a) The basis for the general flood recovery human error rate of 1E-2 (item 4 on page 3.6-9),

    (b) The methodology used to identify and select the human actions that would not be affected by the flood (item 6 on page 3.6-9).

    (c) The quantification technique(s) used to reevaluate the selected human actions.

    (d) The methodology used to identify and the techniques used to quantify the additional flood specific recovery actions (item 7 on page 3.6-9). Provide examples illustrating the application of the methodologies and techniques described in b) and c) above.

    (e) On pages 3.6-11 and 3.6-12, the submittal describes the approach used to quantify human actions for control room floods. Please describe this approach by way of examples from the flooding analysis that exercise all aspects of the methodology. Please provide the basis for all information used in the examples.

17. The submittal is unclear on what human reliability analysis was performed during the Level 2 analysis. Indications that operator actions were considered include:

    On page 4.5-3, the submittal states that "Operator recovery actions ... are addressed by this event."

On page 4.6-4, the submittal states that "... identifies the possibility for successful operator action to recover an alternate injection system ...."

Indications that operator actions were not credited include:

On page 4.5-4, the submittal states that "No operator action to depressurize the RCS are credited in this study."

On page 4.6-4, the submittal states that "... no credit is given for this type of recovery in this analysis."

And finally, on page 4.8-7, the submittal states that "Operator recoveries ... have not been credited in the Level 2 analysis."

Please provide the following:

(a) Describe and list which human actions were considered in the Level 2 analysis.

(b) Discuss whether the actions were quantified with a value other than 1.0.

(c) For any actions with a value other than 1.0, please describe the technique used to quantify the event(s) by way of examples.

Level 2 Questions

1. Reactor Cavity Configuration -- A key feature of the Waterford reactor vessel is that all core instrumentation is routed from the top of the vessel. Since there are no bottom head instruments at Waterford, there is no instrument tunnel that provides access from the reactor cavity to the upper containment volumes. The reactor cavity is open to the upper containment through the very small annulus between the vessel and cavity wall and to the steam generator compartments through the RCS pipe penetrations in the cavity wall. According to the IPE submittal, most of the core debris that is ejected through this path is expected to strike the missile shield above the reactor vessel and not be dispersed into the upper containment.

   Besides the above-mentioned small areas that communicate with the containment volumes, a relatively small tunnel connects to ductwork that provides reactor cooling. The ductwork rises up a chimney that allows personnel access to the containment sump. A door in the duct allows personnel access and water from the containment sump into the cavity. According to the IPE, "This door would likely be blown out during a high pressure vessel failure, allowing core material to be ejected into the containment atmosphere. However, the amount of material would be relatively small because of the small area, the 90° turns, and interference from the ductwork that would likely be torn off the wall and block the flow up the chimney." The IPE also states that "when the containment sump fills (800 gallons), water overflows into the reactor cavity. .... This key design feature of the Waterford 3 containment ensures that the cavity will in almost all cases be filled with water prior to vessel failure."

   According to the above discussion, the detailed cavity configuration and the paths that connect the cavity region and the containment volumes are important in accident progression. However, detailed figures of the reactor cavity region are not provided in the IPE submittal. Please provide simplified schematic drawings showing the important features of the cavity region that are involved in the above discussion. Please discuss, using some simple, but adequate, quantitative values (instead of general qualitative discussion), the following questions related to the cavity configuration:

   (a) Please discuss the pressure in the reactor cavity and the uplift force on the reactor vessel after high pressure vessel failure and whether they will cause any structural damage or lift the vessel from its original location. Although structural failure of the reactor cavity wall due to HPME is considered in the IPE, it is not clear from the discussion presented in the IPE submittal (p4.6-10) whether a plant-specific analysis was performed for the Waterford 3 cavity.

   (b) Please discuss the uplift force on the missile shield above the reactor vessel after high pressure vessel failure, and whether this force is sufficiently large to move the missile shield from its original location and cause structural damage.

(c) Please provide a more detailed discussion on how the core debris is forced through the cavity tunnel and the potential of the core debris to come in contact with the containment shell structure and cause structural thermal failure. Please include in the discussion the path of core debris dispersion and the pressure that drives the dispersion.

(d) Please discuss the status (open or closed) of the door (in the duct) during normal operation and the effect of this door on water flow to the cavity (Although a basic event is presented in Table 4.6-2, Event PRCAVDROPN, it is not included in the CET logic trees and is not discussed in the submittal.). Please include in the discussion the possibility of the door closing due to water flow.

2. External Vessel Cooling -- According to the IPE the reactor cavity will be filled with water prior to vessel failure in almost all cases. It is stated in the IPE submittal that "Only about 56,000 gallons of water are necessary to fill the cavity to the point that ex-vessel cooling can be effective. The amount of water can be provided solely from the RCS during boil off, such as during a station blackout, and accounting for steam in the containment, a water film on containment surfaces, and water left in the reactor vessel lower plenum." However, the external cooling model is not mentioned in Section 4.2 when the MAAP model for Waterford 3 is discussed. Since external vessel cooling may delay, if not terminate, vessel penetration, fission product production and release paths are 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. Additionally, external cooling may maintain the RCS at high temperature for a longer time.

(a) Please discuss the effect of external vessel cooling on source term definition and on the probability of creep rupture of RCS boundaries and steam generator tubes, and consequently, the effect of external vessel cooling on containment performance and source terms for Waterford 3.

(b) The probability of successful ex-vessel cooling (such that vessel breach is avoided) is assigned values of 0.75 to 0.90 in the IPE. Please discuss the basis for their derivation (e.g., the available analyses and test results and their roles in the determination of the values used in the IPE).

3. MAAP Analysis and Sequence Selection -- In NUREG-1335, it is suggested that one or more sequences would be selected to represent each PDS bin. These representative sequences can be used to quantify the containment event trees (Step 2 of Appendix A). In the Waterford 3 IPE, the MAAP code was used to develop information to assign basic event and containment failure probabilities. The sequences that are calculated by MAAP include (1) large break LOCA, (2) small break LOCA, (3) total loss of feed water, and (4) containment bypass. However, the relationship between sequence selection and the results of the Level 1 accident analysis is not discussed in the submittal. Please discuss the criteria

used to select representative sequences for the individual PDS bins provided in Table 4.3-4 of the submittal and the use of the MAAP results for the quantification of accident progression for these PDS bins.

4. Sensitivity Studies of Containment Phenomena -- Recognizing the uncertainty in various severe accident phenomena and how the accident progression can be affected, Waterford 3 performed some sensitivity analyses with the MAAP code to ensure that a broad spectrum of possible outcomes were covered (p4.2-3). The issues that were investigated by MAAP analyses include (1) in-vessel hydrogen production, (2) direct containment heating, (3) debris bed coolability, and (4) vessel failure penetration radius. General results of these sensitivity analyses are discussed in Section 4.2.3 of the IPE submittal. Results from the sensitivity cases are presented in the submittal to show the uncertainty of individual issues on some containment parameters (e.g., the uncertainty of DCH on containment pressure load). However, their effects on containment release profiles are not discussed. A sensitivity study in the latter sense is found in Section 4.9 of the submittal, but it does not include the above issues. Since the concern about the uncertainties of these issues is their effect on containment failure and fission product release, please discuss the effect that the uncertainty of these issues has on containment failure probabilities.

5. ISLOCA -- The PDSs that have a frequency greater than 1E-7 are presented in Table 4.3-4. According to this table, the containment system conditions for ISLOCA are classified as B, which means that containment sprays are available in both injection and recirculation modes. Since in an ISLOCA the coolant is most likely lost through the break area to outside the containment, the availability of containment sprays in recirculation mode is questionable. Please clarify this question.

6. Induced SGTR -- Temperature induced SGTR is represented in the IPE by a basic event PRSGOK. However, this basic event appears only in the logic tree determining RCS depressurization. It is not clear from the IPE submittal how temperature induced SGTR is treated in the IPE as a fission product release mode. Please discuss how temperature induced SGTR is treated in the IPE. Also, the probability of induced SGTR due to forced circulation caused by the restart of the RCPs is not addressed. This mechanism is considered in some IPEs, since the Inadequate Core Cooling guidelines call for the RCPs to be restarted. Continuous operation of the RCPs would cause the high temperature gases to be transported to the SG and a higher probability of induced SGTR. Please discuss this issue.

7. Pressure Capacity of the Containment Vessel -- Section 4.4.2 of the IPE submittal discusses the containment ultimate strength evaluation. It is noted that the distribution provided in figure 4.4-1 for failure probability versus pressure (i.e., the fragility curve) is almost linear from 40 psig (approximately the design pressure) to 135 psig (the mean failure pressure). The use of this distribution contributes to the relatively high containment failure probability for early containment pressure loads predicted in the IPE. For example, according to Table 4.6-3 of the submittal, the containment failure probability is 0.286 for a containment pressure load of 89 psia. It seems that the failure probability would be much smaller if a distribution similar to that

developed in the NUREG-1150 PRAs, which shows a steeper drop from the mean value to the design value, was used. Please discuss the applicability of the information obtained in the NUREG-1150 studies to Waterford 3, and, if applicable, the effect of the different distribution on containment failure profiles.

8. CET and Logic Trees -- The following requested information pertains to the containment event and logic trees:

   (a) The CETs presented in Figure 4.6-1 to 4.6-4 and some of the logic trees presented in Figure 4.6-5 are illegible. Please provide copies of better quality.

   (b) The probability values for the logic tree basic events are listed in Table 4.6-2. However, the following basic events which are presented in this table are not found in the logic trees presented in Figure 4.6-5: PRALPHAH, PRALPHAL, PRCAVDROPN, PRHEATUPLO, PRIGNITES, and PRVRYLATE. Please discuss the reasons for their omission in the logic trees presented in the submittal.

9. Recovery In SBO Sequences -- Results of Waterford 3 CET analysis show that over 50% of SBO sequences do not result in containment failure. Please discuss the important factors that contribute to this result. If recovery of electric power is the primary cause, then, please discuss the derivation of the values used for electric power recovery (i.e., basic events PRAC<CFLEM, PRAC<CFLLT, PRAC<VBEM, and PRAC<VBLT).

10. Isolation Failure -- Containment isolation failure is evaluated in the CET under top event CFE (Containment fails early) and is addressed in the associated logic trees. It is stated in the submittal that "The probability is determined by solving a separate fault tree for failure to isolate these penetrations." (p4.6-7). However, the fault trees for isolation failure are not provided and details are not discussed in the IPE submittal.

    With respect to the analysis of containment isolation failure probability, NUREG-1335 (Section 2.2.2.5, p2-11) 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)." Please discuss your findings related to all of the above five areas.

11. Ex-Vessel Debris Coolability -- The following requested information pertains to ex-vessel debris coolability:

    (a) According to the IPE submittal, "Three non-coolable debris bed events are used to distinguish when HPME occurs (PRCDB-HP), ex-vessel steam explosion occurs (PRCDB-LPSE), or no dispersal and fragmentation occurs (PRCDB-LPNS)." (p4.6-12). The logic trees that involve PRCDB-HP and PRCDB-LPNS are presented in Page 24 of Figure 4.6-5. However, the logic tree involving PRCDB-LPSE is missing. Please discuss whether this is because it is not included in the CET model or simply missing from the submittal. Please discuss the reason for omission from the CET model if the former is the case or provide the missing information if the latter is the case.

(b) The probability values assigned to the three basic events for "a coolable debris bed does not form" are 0.4, 0.2, and 0.5 for PRCDB-HP, PRCDB-LPNS, and PRCDB-LPSE, respectively. Core concrete interaction (CCI) occurs if ex-vessel debris is not coolable. The containment failure mechanisms considered in the IPE for CCI include those associated with non-condensible gas generation and basemat melt-through. A containment failure probability of 0.005 is assigned to both of these mechanisms (i.e., basic events PRNCG-FAIL and PRMT1). Please discuss how these values were arrived at in the IPE. Please also discuss the sensitivity of containment failure (or containment release category, CRC) results for Waterford 3 to these parameters.

(c) Also, please compare and discuss, with respect to the assumed coolable depth of 25 cm in Generic Letter 88-20, the depth of the core debris in the cavity (including the cavity sump) and the effect of non-uniform spread of debris on debris coolability if all debris is retained in the reactor cavity.

12. Harsh Environmental Condition -- The effects of harsh environmental conditions on the operation of containment fan coolers are addressed in the IPE by the basic events in the CET logic trees (i.e., Basic Events PRH2BCFC3, PRHPME_CFC, PRHARSHCFC, and PRHARSHCS). However, no discussion is provided in the submittal on how the probability values of these basic events are derived. Please discuss the basis and the environmental conditions used in the IPE for their derivation. Please also discuss why the effect of environmental conditions on containment spray (i.e., basic event PRHARSHCS) is considered only for fission product removal (FPR) but not for late containment failure (CFL).

13. CPI and Hydrogen Combustion Issue --The Generic Letter CPI recommendation for PWR dry containments is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures).

Containment failure due to hydrogen combustion is considered in the IPE analysis for both early and late time frames. Containment loading due to hydrogen combustion is obtained primarily from MAAP calculations. However, the CPI issue is not addressed specifically in the IPE submittal.

Please discuss whether plant walk downs have been performed to determine the probable locations of hydrogen releases into the containment. Including the use of walk downs, 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. 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.

14. Sensitivity to Hot Leg Creep Rupture -- Sensitivity to hot leg creep rupture is investigated in the IPE only for sequences with medium RCS pressure. The values of basic events PRHLSLOK1 (for small break LOCA) and PRHLSLOK2 (for sequences where RCS pressure drops below the SRV set point) are varied in the IPE for the sensitivity study. However, the sensitivity to hot leg creep rupture at high RCS pressure is not investigated. The basic event that address this issue is PRHLSLOK (hot leg and surge line remain intact for high pressure). Although there is a higher probability of hot leg creep rupture for high RCS pressure, uncertainty exists nonetheless. Additionally, the high RCS pressure used in the Waterford IPE for PDS definition is "greater than 1400 psia" which is less than that used in NUREG-1150 or other IPEs (e.g., greater than 2300 psia). The uncertainty on hot leg creep rupture thus increases. Please discuss the basis for the value of PRHLSLOK used in the IPE and the sensitivity of the containment response analysis to PRHLSLOK.

15. High Temperature Failure of Elastomer Penetration Seals -- High temperature failure of elastomer penetration seals is considered in the IPE as a late failure mechanism. This failure mechanism is briefly discussed in the IPE submittal. Containment failure probability due to this mechanism is calculated in the IPE by assigning probability values to basic events defined for this mechanism (PRHARSHCI3 and PRHARSHCI6). However, detailed discussion is not provided in the submittal on their derivations. Please discuss in more detail the sealing materials used for the Waterford 3 penetrations, their thermal properties, and the expected harsh environmental conditions, and outline the derivation of the values for PRHARSHCI3 and PRHARSHCI6.

16. Source Terms for SGTR -- SGTR sequences are grouped to Containment Release Category (CRC) BP-E5A in the IPE (Table 4.8-2). According to the IPE submittal, the release for CRC BP-E5A is scrubbed by the water in the steam generator (p4.8-3). The release fractions predicted in the IPE for the SGTR sequences are therefore much less than those for some early failure sequences and ISLOCA sequences (Table 4.8-3). The availability of water scrubbing for the SGTR sequences used in the IPE for source term definition does not seem to be consistent with the discussion provided in the submittal for radionuclide release characterization, in which (p4.7-5) it is stated that "No credit is taken for the availability of feedwater which provides substantial scrubbing."

    Please discuss how water is available in the steam generator for fission product scrubbing. Please discuss the conditions (e.g., pressure, temperature, water level, etc.) in the RCS and the steam generators during accident progression for SGTR sequences to show that water is not boiled off but still available in the steam generator for fission product scrubbing during fission product release phases. If water scrubbing is available in some, but not all SGTR sequences, then, please discuss the probability of SGTR sequences for which water scrubbing is not available.

PD IV-1 DOCUMENT COVER PAGE

DOCUMENT NAME: WAT74487.RAI

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON WATERFORD 3 STEAM ELECTRIC STATION INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL (TAC NO. M74487)

ORIGINATOR: Chandu Patel

SECRETARY: Shirley Poonai

DATE: January 22, 1996

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