February 4, 1994

Docket No. 50-213

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Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

SUBJECT: HADDAM NECK PLANT - INDIVIDUAL PLANT EXAMINATION, REQUEST FOR ADDITIONAL INFORMATION (TAC NO. M74417)

By letter dated June 29, 1993, Connecticut Yankee Atomic Power Company submitted the Haddam Neck Plant Individual Plant Examination (IPE) for staff review. Based on our review of your submittal, the staff has determined that additional information is necessary to complete our review. Enclosed are additional questions regarding the internal events analysis in the IPE and the containment performance improvement program. Please respond within 60 days of receipt of this letter.

This requirement affects one respondent and, therefore, is not subject to Office of Management and Budget review under P.L. 96-511.

Sincerely,

Original signed by:

Alan B. Wang, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

OFFICE	LA:PDI-4	PM:PDI-4	D:PDI-4		
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DATE	2/3/94	2/3/94	2/4/94	/ /	/ /

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

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Enclosure:
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cc w/enclosure:
See next page

cc:

Gerald Garfield, Esquire
Day, Berry and Howard
Counselors at Law
City Place
Hartford, Connecticut 06103-3499

J. M. Solymossy, Director
Nuclear Quality and Assessment Services
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Kevin T. A. McCarthy, Director
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, Connecticut 06106-5127

Allan Johanson, Assistant Director
Office of Policy and Management
Policy Development and Planning Division
80 Washington Street
Hartford, Connecticut 06106

J. P. Stetz, Vice President
Haddam Neck Plant
Connecticut Yankee Atomic Power Company
362 Injun Hollow Road
East Hampton, Connecticut 06424-3099

J. J. LaPlatney
Haddam Neck Unit Director
Connecticut Yankee Atomic Power Company
362 Injun Hollow Road
East Hampton, Connecticut 06424-3099

Donald B. Miller, Jr.
Senior Vice President
Millstone Station
Northeast Nuclear Energy Company
Post Office Box 128
Waterford, Connecticut 06385

R. M. Kacich, Director
Nuclear Planning, Licensing & Budgeting
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

S. E. Scace, Vice President
Nuclear Operations Services
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Board of Selectmen
Town Office Building
Haddam, Connecticut 06438

Resident Inspector
Haddam Neck Plant
c/o U.S. Nuclear Regulatory Commission
361 Injun Hollow Road
East Hampton, Connecticut 06424-3099

Nicholas S. Reynolds
Winston & Strawn
1400 L Street, NW
Washington, DC 20005-3502

Enclosure

REQUEST FOR ADDITIONAL INFORMATION REGARDING HADDAM NECK INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL

1. The submittal notes that the calculations indicate that the Haddam Neck Plant (HNP) cavity wall can withstand the greatest pressure load (calculated by the MAAP code) with acceptable margin. (page 4-32) What is the maximum pressure that the HNP cavity wall can withstand given the stated temperatures?

2. Discuss the conditional probability of vessel breach at the HNP, given core damage.

3. Discuss which radionuclide retention structures were credited in the HNP IPE.

4. Top Events 9 and 10 (X2 and S2) of the HNP containment event tree (CET) are system-related. According to Table 4.6-1 of the submittal, for most of the plant damage states (PDSs) the split fraction is 0.9999 for both X2 and S2. In addition, many of the split fractions in Table 4.6-1 are listed as 0.9999 and 0.0001, which were used to represent the Values 1 and 0, respectively. (Pg. 4-33) CET nodes H3 and C3 are concerned with containment vulnerability due to hydrogen combustion and were specified as 0.0001 each, resulting in very low probability of containment failure. Describe the process used to derive the split fractions. What was the rationale for using the extreme values 0.9999 or 0.0001.

5. Please provide a copy of Reference 6 to this report, which the IPE team used to obtain a containment isolation failure probability for the HNP.

6. Table 3.1.5.1 of the submittal gives the time of core damage and the reactor coolant system (RCS) pressure at the core damage. What codes/analyses were used to obtain these two parameters?

7. Section 4.1.2, page 4-2 of the submittal, notes that the cavity and instrument tunnel access the lower containment via a cylindrical, vertical shaft, 3 feet in diameter. This access shaft is surrounded by a 5 feet high curb that prevents containment sump water from spilling over into the reactor cavity.

   According to Figure 4.1.2-3 on page 4-44 of the submittal, this vertical shaft appears to be 3 feet in radius. Also, the 5 feet high curb is not shown in the figure. Please clarify and illustrate.

8. Section 5.1, pages 5-1 and 5-2 of the submittal, does not list PLG, Incorporated, as part of the back-end analysis team. However, Section 5.2, page 5-2, notes that PLG provided technical support and served as the primary independent reviewer of the back-end analysis.

   Please describe the role that PLG played on the back-end analysis team and on the back-end review team.

9. Section 4.2.2, page 4-9 of the submittal, notes that in the MAAP sensitivity study "TTRX = 1,800 seconds" was used as the input value to find the effect of molten core mass at the time of vessel breach, instead of using the default value of "TTRX = 60 seconds." This sensitivity study appears to have been ineffective because of the insignificant increase of the containment pressure. How did you select parameters to vary in sensitivity analyses? Describe how the MAAP sensitivity analysis outlined in Table 4.2.2-1 conforms with the EPRI-MAAP sensitivity analysis guidance referenced in Ref. 4-4.

10. Generic Letter 88-20 states that the following should be reported:

> any functional sequence that has a core damage frequency greater than $1x10^{-6}$ per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to BWR-3 or PWR-4 release categories of WASH-1400.

It is not clear whether the HNP IPE submittal meets this reporting requirement because only iodine releases were reported. Please indicate how the HNP IPE meets this provision, i.e., how does the iodine release correlate with the other nuclides?

11. Discuss the leakage characteristics of personnel and equipment hatch inflatable seals (if any) and silicon gaskets given their likely condition at the onset of a severe accident (Figure 4.4.7-2). Has a program been initiated to control the equipment hatch preload, as suggested in the IPE, to prevent premature flange separation?

12. (Pg. 4-10) In discussing MAAP parameter FCHF, it is indicated that when CARFANS are not operational, the containment will be moisture (steam) inerted; therefore, no hydrogen burn will take place. Discuss the steam inerting limits (in relation to hydrogen levels) used to justify this assumption.

13. (a) (Pg. 4-10) In discussing MAAP parameter DXHIG, it is indicated that this value was changed from 0.0 to 1.0, presumably to insure global burns at hydrogen levels above the global limit. However, the impact of these global burns is characterized as negligible. Discuss the pressures and temperatures resulting from global burns that form the basis for this conclusion.

(b) (Page 4-28) Containment Event Tree top event C1, "No Containment Failure Prior to Vessel Breach," is structured to include early hydrogen combustion and detonation effects prior to vessel breach. However, in Table 4.6-1, "A Summary of Split Fractions Used for PDSs", the split fractions include contributions from failure to isolate containment and α mode failure but apparently none from hydrogen effe[illegible] Discuss the possibility of a stratified containment atmosphere producing local hydrogen detonation and its potential imp[illegible]ntainment. Does the reliance on no detrimental h[illegible] effects account for the fact that the conditional [illegible]e probability for the HNP is less than one-fifth that [illegible] (NUREG-1150) even though Surry has a higher ultimate strength? Discuss what structural characteristics of the HNP containment cause the lower containment ultimate strength in relation to Surry.

(c) (Pg. 4-33) The IPE indicates that the HNP is not vulnerable to hydrogen-related failures. Have plant walkdowns been performed to determine the probable locations of hydrogen released into the containment? Including the use of walkdowns, discuss the process used to assure that: (1) local deflagrations would not translate to detonations given a propitious nearby geometry, (2) equipment considered necessary in mitigating severe accidents will not be impaired by hydrogen burns, and (3) the containment boundary, including penetrations, would not be challenged by hydrogen burns.

(d) (Pg. 4-33) In scenarios involving ex-vessel generation of hydrogen via core-concrete interaction, have sensitivity calculations been run to see how much margin exists between the best estimate and the upper bound for a global hydrogen burn? Has a calculation been performed to estimate the maximum amount of hydrogen combustion, in terms of equivalent core zirconium oxidation, the containment can withstand? Please discuss this aspect of the analysis.

(e) Provide a brief summary of the analysis (including important assumptions and conclusions) mentioned in Reference 4-24 which was used to conclude that hydrogen detonation is unlikely at the HNP.

14. (Pg. 4-21) The second phase of the containment review concluded that restraint of containment by attached structures would likely inhibit containment failure below grade. Please expand upon this discussion. For example, does it mean that above grade failure is more likely due to the tearing or "punching" effect of rigid attached structures?

15. (Pg. 4-32) Given certain containment configurations high pressure melt ejection (HPME) and associated molten debris could impact the integrity of the containment wall. Discuss the possibility of this happening at the HNP via inducing an instrument tunnel/seal table failure.

16. (Pg. 4-34) Seven of the PDSs were quantified using results from the other 13 PDSs. Describe how you used "appropriate results" from PDSs with similar containment responses to quantify the 7 PDSs.

17. (Pg. 7-3) Back-end insights gained from the IPE indicate that HPME, direct containment heating (DCH), and the lowering of containment steam content via containment spray operation (thus increasing hydrogen combustion likelihood) all have minor or insignificant effects on containment failure. Have sensitivity calculations been run which consider the possibility of higher resulting containment pressure from hydrogen combustion and/or lower containment failure capabilities? If so, discuss these insights.

18. (Pg. 7-3) One back-end insight gained regarding containment bypass indicated that a utility-initiated project had begun to install a pressure interlock to prevent bypass. What is the status of that effort? When will it be completed?

19. Table 3.1.5-1, "Core Damage States from the Front-End Analysis", lists as "Late" the RCS pressure at Time of Core Melt. Please clarify.

20. Table 3.2.1-1, "Summary of Major System Changes", indicates that the B train of DC power now has a "battery-eliminating battery charger". Please clarify.

21. Per NUREG-1335 reporting guidelines, submittals should contain a concise discussion of the criteria used to define "vulnerability,". However the criteria provided in Section 3.4.2 in the submittal use the word "significant," which is not defined. If explicit criteria had not been developed, discuss the process used to evaluate the need for plant improvements during and upon completion of the IPE, and the level of significance at which plant improvements were implemented.

    (a) For instance, the highest hardware-related importance measure given in Table 3.4.1-4, "Module/Component Fussell-Vesely Importance Measure", relates to failures of the "B" train of the emergency diesel generator. Discuss your decision process in disposition of this importance measure.

    (b) (Pg. 7-3) Have you evaluated the need for earlier manual actuation of containment sprays to cool ex-vessel core debris? If you have, please discuss.

22. (a) The submittal indicated that the methodology used for the analysis was changed from support state to linked fault tree. It is not clear from the submittal to what degree reductions, in contributions from sequences, or in core damage frequency (CDF) are due to changes in plant or methodology. Please discuss the impact of the methodology change.

    (b) Please identify the date for which the model represents the plant.

23. The loss of systems or components such as, closed component cooling water, heating ventilation and air conditioning (HVAC), and AC buses (4160v) have been analyzed as initiating events in some probablistic risk assessments (PRAs) and found to be significant contributors to CDF. Please describe your investigation into the loss of these items including loss of HVAC to electrical equipment rooms, and the control room at the HNP and their possible contribution to the estimated CDF or the rationale for the decision not to include them as initiating events. Please identify the frequency if the magnitude of the frequency was part of the reason for their elimination.

24. Interfacing systems loss of colant accident (LOCA) is identified as a type considered in the analysis. However, no frequency or description of how the frequency of this event is assessed is provided. Provide a description of the assessment of each system considered for this type LOCA, also provide the rationale for any interfacing systems not included for consideration. Include in the description your consideration of maintenance, test, and human error for values in the interface between systems.

25. The IPE indicates that flood scenarios which contribute < 1E-6 to CDF were screened out. Depending on the magnitude of the sequences and the number of sequences leading to core damage the contribution from these screened scenarios may be somewhat important. Is the contribution to CDF from all of the screened-out flood sequences greater or less than those flood sequences which were reported?

26. Section 3.3.8 of the submittal indicates that there is no modeling for a reactor trip caused by a circulating water pump seal or expansion joint rupture that is isolated in time to prevent damage to the service water (SW) pump motors. It is also indicated that the frequency of the expansion joint rupture (1E-6; 1 failure in about 114 years) was reduced from what was used in the original study, because HNP has operated for 25 years without a flood caused by an expansion joint rupture. However, the current value was never identified. Please identify the current value used for frequency of expansion joint rupture and the proportion of the expansion joint ruptures that have not been modeled because they were isolated before the SW pump motors were damaged. If the proportion is high, please identify the basis used for eliminating those not modeled.

27. (a) The need to borate for overcooling events is not addressed in the event trees. Please discuss your consideration of the need for borating under these conditions and the possible contribution to CDF.

    (b) The containment air recirculation fans (CARFs) and charging system (CS) are shown as top events in the large LOCA tree. Does failure of both of these functions cause core damage as indicated in the tree? Does failure of either or both of these functions cause failure of the pumps used for recirculation due to loss of net positive suction head (NPSH) or high temperature thus leading to core damage?

28. The UFSAR implies that a value of 1300 psia vessel pressure is the point at which no flow can be injected from the high pressure safety injection (HPSI). The shutoff head of the HPSI is identified in the submittal as 1500 psig. Will flow to the vessel be available if the vessel pressure is above 1300 psia, and how does this affect your analysis?

29. (a) For PWRs, consideration of RCP seal failure is important, because of its impact on the determination of dominant sequences (i.e., transients vs. LOCAs). It is not clear from the submittal and the changes made in the interim since the original PRA if your treatment of RCP seal LOCAs has changed. Please describe your treatment of RCP seal degradation and failure during loss of key support systems which could lead to loss of seal cooling. Include in your discussion, the timing of RCP seal degradation, the probability of failure and seal flow rates, and any recovery actions or improvements that would enhance mitigation of RCP seal cooling accidents.

    (b) What impact does containment isolation (CI) have on the ability to cool the RCP seals, i.e., does CI isolate both charging flow and thermal barrier cooling for the RCP seals and what impact does it have on the contribution to CDF from seal LOCA?

30. The submittal indicates that the cnarging pumps are manually loaded onto the diesels after a loss of offsite power. It is not clear how this is addressed in the model. Please discuss the credit taken for this action for LOCAs and transients.

31. Makeup to the refueling water storage tank (RWST) is listed as one of the success criteria for steam generator tube rupture (SGTR), but the information in the submittal regarding this function is not clear. Please identify the systems and the required supports necessary to provide this function and describe how they are addressed in the model.

32. (a) The submittal did not contain the notes for the dependency table as was provided in the PRA, and the current table does not contain the indications as to the type of dependency that exists for the systems. Since there have been a number of changes to the plant and some of the dependencies may have changed since the PRA was issued, please provide a copy of the notes for the table.

    (b) The dependency matrix identifies a charging pump dependence on DC power. However the event tree for total loss of DC power questions the operation of the charging pumps during this event. How is this dependency taken into account in this event?

(c) The dependency table indicates a dependence of the auxiliary feedwater (AFW) on DC and instrument air (IA). However, due to changes since the 1986 PRA, it is not clear from the submittal how the impact of these dependencies and the actions to operate the AFW are accounted for in the analysis. Please provide a discussion regarding this aspect of the analysis.

33. (a) It is not clear from the submittal if the air used to cool rooms in the building is outside air or cooled air. If outside air, to what degree has loss of the HVAC systems been taken into account for the impact on room temperature during the summer months when the outside ambient air temperature is high?

34. (b) The basis for the elimination of HVAC systems as supports for frontline and support systems (including control room, diesel generators, AC, MCC5 and VAC, as the dependency matrix shows a dependency on HVAC for some of these) is not clear from the submittal. Please identify the systems for which HVAC was included in the systems model (fault tree) or the basis on which it was eliminated from analysis. In addition, it is indicated that the charging pump oil coolers are capable of being cooled by a fan, indicating that they use room air for the cooling. If cooling is lost to the room for the charging pumps, and the temperature of the room rises, what is the impact on the capability of the fan to cool the oil coolers?

35. (a) Please identify the components for which plant-specific data was used, as requested by NUREG-1335, and discuss how the new data was used to update the previous component failure rates.

(b) Please identify if any plant-specific data was used for common cause component failures and/or the source of the generic data used.

36. The IPE submittal presents the contribution to CDF from the initiating events, but does not present other methods of identifying the contributors which may provide insights or identify concerns in your search for vulnerabilities. Please discuss any <u>insights gained</u> and provide the <u>contribution to CDF</u> identified for the following:

(a) Loss of offsite power, station blackout, RCP seal LOCA, loss of high pressure injection

(b) Systems (top 10) contained in the cutsets for the sequences leading to core damage (including support systems)

(c) Common cause failure

(d) Maintenance

37. Section 3.4.3 (Decay Heat Removal (DHR) Evaluation) discusses the DHR system. However, it does not provide a thorough discussion of the evaluation of the DHR as indicated in NUREG-1335. It does not address its contribution to CDF as an entity, nor does it provide insights into the relative contribution to CDF of its separate constituent systems. In addition, Generic Letter 88-20, Appendix 5 indicates that support systems are important to the DHR function and suggests that they be considered in the search for DHR-related vulnerabilities. Therefore, provide a discussion of insights derived and provide the contribution of DHR and its constituent systems, including bleed and feed, to CDF and the relative impact of loss of support systems on the frontline systems that perform the DHR function.

38. The submittal provides limited information on the human reliability analysis (HRA) methodology and references Section 5 of the HNP PSS for discussion. Section 5 provides a fairly comprehensive summary of the HRA as it was originally performed. However, the submittal notes significant changes in the basic HRA structure (SHARP1 vs. SHARP), use of different techniques and data sources (EPRI-ORE vs. HCR, OAT, etc.), and different operator actions identified (some added, some deleted). Additional information in the recently transmitted Calculation Record (C2-517-1044-RE, dated 1/25/93) provides additional detail on the modified HRA performed for the IPE. Please explain the process used in the IPE to identify important human actions for quantification. Include in your discussion:

    (a) What was done to provide reasonable assurance that all important human interactions were included?

    (b) What process/criteria were used to reduce the initial list of "candidates" to select the relatively few that were quantified?

    (c) Discuss the process and rationale used to add and delete human interactions from the original PRA.

39. The submittal does not discuss pre-initiator human actions, such as calibration errors, failure to properly restore equipment after maintenance, test or calibration, etc., which may have an impact on availability of equipment needed to cope with an abnormal event. The Appendix Calculation File shows a Millstone 3 screening model that apparently was used to quantify some pre-initiators. The review of important modules/components identifies some pre-initiator actions, but the human error probabilities (HEPs) do not match the screening values in this appendix.

    (a) Identify which pre-initiator actions were identified as important and provide examples of their quantification process.

    (b) Explain how dependencies associated with pre-initiator human errors were addressed and treated in the IPE to assure that important accident sequences were not eliminated. These dependencies could, for example, affect all the human events simultaneously, or could only affect certain human events such that only a series of human events are determined to fail simultaneously (e.g, complete dependence may be assumed for miscalibration of all reactor water level sensors). Please explain.

40. Regarding post-initiator events, we note that the original HRA includes a concise description of the performance shaping factors (PSFs) selected as important for "cognitive" (OAs) and "manipulative" (HIs). Given the changes in operator actions and HRA methodologies and lack of equivalent discussion in the current HRA:

    (a) Identify PSFs that were identified as important, how they were evaluated, and provide examples to show how they were used to modify generic HEPs.

    (b) Explain how the plant-specific simulator data reported in the Calculation File were used in the quantification of human error.

    (c) Explain why the "old" HCR correlations based on "skill-, rule-, or knowledge-based" behavior are retained for some estimates.

    (d) When "new" HCR/ORE correlations are selected, what is the basis for using "generic" ORE curves to estimate the parameters as opposed to plant-specific simulator data or operator interviews that are recommended as preferred options by ORE.

    (e) Provide the data source for the basic human error probabilities (BHEPs) for the HI component of post-initiators.

    (f) Please explain by way of example how "available" time vs "required" time was calculated for the various post-initiator human events.

41. Explain how dependencies associated with post-initiator human errors were addressed. These dependencies could, for example, affect all the human events simultaneously. On the other hand, dependencies could affect a certain set of human events such that only a specific series of human events are determined to fail simultaneously (e.g, complete dependence may be assumed for manual actuation of all injection systems). The discussion should particularly address the two points below:

    (a) Post-initiator human events can be modeled in the fault trees as basic events such as failure to manually actuate a system. The probability that the operator performs this function is dependent on the accident in progression (e.g., what symptoms are occurring, what other activities were previously successfully and unsuccessfully performed). When this basic event (i.e., failure to manually actuate the system) is modeled in the fault tree and 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, resulting in artificially low calculated human error contributions if dependencies are not taken into account.

(b) Post-initiator human events can also be modeled in the event trees as top events. The probability that the operator performs this function can still be dependent on the accident progression. The quantification of the human events needs to consider the performance shaping factors (PSFs) associated with each different sequence and the dependencies between other human events.

42. Please discuss how recovery actions were identified and evaluated. In particular:

    (a) Discuss the evaluation recovery actions that are not directed by procedures.

    (b) Discuss how recovery actions outside of the control room were evaluated, in particular, plant walkdowns performed to verify timing estimates and evaluate accessibility, environmental factors, etc.

    (c) Provide a listing, as requested in NUREG-1335 (Para. 2.1.6.6), of those sequences which, except for low human error probabilities in recovery actions would have been above the core damage frequency screening criterion.

43. Identify any human actions taken credit for in the back-end analysis, any sensitivity performed and any insights.

44. The Flooding Analysis (December 20, 1988) identifies a number of credited operator actions and provides quantitative estimates of HEPs. Discuss any differences in the treatment of these actions vs other operator actions.

45. Our front-end review has identified the actions: a) use of SW if component cooling water (CCW) is lost, b) borate if the reactor is cooled below hot zero power, and c) cross-tying of electrical power trains, as potentially important. How were these actions addressed in the HRA?