50-271

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

October 8, 1998

EA 97-387

Mr. Don K. Davis
President & Chief Executive Officer
Yankee Atomic Electric Company
580 Main Street
Bolton, MA 01740-1398

SUBJECT: YANKEE ATOMIC ELECTRIC COMPANY RESPONSE TO U. S. NUCLEAR REGULATORY COMMISSION (NRC) DEMAND FOR INFORMATION (OI Report No. 1-95-050)

Dear Mr. Davis:

This letter is in response to your letter of March 11, 1998, which replied to an NRC Demand for Information (DFI) issued on December 19, 1997, to the Yankee Atomic Electric Company (YAEC) and Duke Engineering and Services, Inc. (DE&S). Concurrently, the NRC staff issued a related letter to Maine Yankee Atomic Power Company (MYAPCo) identifying apparent violations of NRC requirements by MYAPCo. MYAPCo responded to the apparent violations in writing on April 6, 1998, and in a predecisional enforcement conference on April 23, 1998. DE&S responded to the DFI by letter dated February 27, 1998. Although the DFI did not require a response from the two individuals identified therein as the Loss-of-Coolant Accident (LOCA) Group Manager and the Lead Engineer, they responded by letter dated March 12, 1998.

The DFI articulated NRC concerns regarding actions of the LOCA Group at YAEC that may have caused: (1) the use of unacceptable evaluation models by MYAPCo to calculate emergency core cooling system (ECCS) performance for the Maine Yankee Atomic Power Station (MYAPS) because the evaluation models were not capable of calculating ECCS performance over the entire spectrum of postulated break sizes, (2) the maintenance and submission by MYAPCo of information that was not complete and accurate in all material respects, (3) the use of an unacceptable evaluation model to calculate ECCS performance at MYAPS because of the incorrect application of the Alb-Chambre correlation, and (4) the use of an unacceptable best estimate evaluation model to calculate ECCS performance at MYAPS in an analysis of reduced steam generator pressure. The DFI required YAEC and DE&S to explain why they should be permitted to perform LOCA analyses or any safety-related analyses to meet NRC requirements and why the NRC should not consider the unacceptable analyses to be the result of willfulness on the part of YAEC and/or DE&S personnel. The DFI was also adressed to DE&S because shortly before issuance of the DFI, DE&S had acquired the YAEC Nuclear services Division (NSD), which includes the LOCA Group.

The NRC staff has completed its review of the responses of YAEC, DE&S, and the two individuals. The YAEC response to the DFI stated that the LOCA Group was part of a DE&S acquisition of certain YAEC assets, that insofar as YAEC is aware, the statements of fact contained in the DE&S response are accurate, and that YAEC adopts the conclusions of the

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DE&S response. The DE&S response discussed the circumstances surrounding the violations, and addressed the actions taken by YAEC and by DE&S to prevent recurrence of the events that gave rise to the DFI. In light of the DE&S acquisition of the YAEC NSD and the YAEC LOCA Group, and the DE&S response to the DFI, the staff's concerns regarding the provision of LOCA analyses and other safety-related analyses by YAEC to NRC power reactor licensees have been addressed by the response of DE&S to the DFI. The staff concludes that the actions taken by the YAEC LOCA Group caused MYAPCo to be in violation of Commission requirements in a number of areas, but that these actions did not result from willfulness on the part of DE&S and/or YAEC personnel. The violations are cited in a Notice of Violation issued concurrently on this day to MYAPCo, a copy of which is enclosed for your information. The staff has determined that it shall take no further enforcement action against YAEC or DE&S with regard to the actions of the LOCA Group at concern in the DFI. Copies of the staff's letters to DE&S and to the two individuals, issued concurrently on this day, are enclosed for your information.

In accordance with 10 C.F.R. § 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and any response (although none is required) will be placed in the NRC Public Document Room (PDR).

Sincerely,

Samuel J. Collins

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enc: Maine Yankee Atomic Power Company
Duke Engineering & Services, Inc.
Mark R. Robeck, Baker & Botts

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Samuel J. Collins, Director
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Duke Engineering & Services, Inc.
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Samuel J. Collins, Director
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

October 8, 1998

EA Nos. 96-299; 96-320; 96-397; 97-034; 97-147 (ISA)
EA Nos. 96-397; 97-375; 97-559 (Investigations)

Mr. Michael J. Meisner, President
Maine Yankee Atomic Power Company
329 Bath Road
Brunswick, Maine 04011

SUBJECT: NOTICE OF VIOLATION
(NRC Inspection Report Nos. 50-309/96-09;96-10;96-11;96-16;97-01)
NOTICE OF VIOLATION
(NRC Office of Investigations Report Nos. 1-95-050, 1-96-025 & 1-96-043)

Dear Mr. Meisner:

This refers to the results of several NRC inspections conducted between July 15, 1996, and March 15, 1997, and three investigations of Maine Yankee Atomic Power Company (Maine Yankee) conducted by the NRC's Office of Investigations (OI) between December 1995 and October 1997. The inspections included an Independent Safety Assessment (ISA), as well as several inspections conducted by resident and Region I based inspectors to follow-up on the ISA findings. The purpose of the ISA was to determine whether Maine Yankee was in conformity with its design and licensing bases; to assess operational safety performance; and to evaluate Maine Yankee's self-assessment and corrective action processes. All of the related inspection reports were sent to you previously. The investigations concerned (1) the adequacy of Maine Yankee's small break loss-of-coolant accident (SBLOCA) emergency core cooling system (ECCS) analyses, (2) the submittal to the NRC of inaccurate information pertaining to the capacity of the facility's atmospheric steam dump valve, and (3) the failure to perform station test procedures as required by facility technical specifications. The synopses of the referenced reports were previously sent to you.

With respect to the ISA and related inspections, the NRC has determined that numerous violations of NRC requirements occurred. The majority of the violations were discussed at a predecisional enforcement conference at the Maine Yankee media center in Wiscasset, Maine on March 11, 1997. While the conference was held to discuss the violations, their causes and your corrective actions, the conference focused on the broader programmatic deficiencies underlying the violations and which contributed to the performance problems at Maine Yankee. The information you presented at the conference was considered in reaching our enforcement decision. Additional violations identified subsequent to the March 11, 1997, conference (Reference: Inspection Report No. 97-01) are also included in this enforcement action, although they were not discussed during the conference. Mr. G. Leitch, formerly of your staff, informed Mr. J. Yerokun of my staff on April 2, 1997, that Maine Yankee agreed that another enforcement conference was not needed to discuss these additional violations.

With respect to the OI investigations referenced above, the NRC transmitted to you on December 19, 1997, a letter describing 13 apparent violations identified as a result of the investigations, to which you responded in writing on April 6, 1998. A closed, transcribed, predecisional enforcement conference was held on April 23, 1998, to discuss the issues associated with the NRC investigations. Based on the results of the investigations, the review of your April 6, 1998, response, and the information you presented at the conference, the NRC has determined that additional violations of NRC requirements occurred.

ISA ISSUES

The specific violations pertaining to the ISA follow-up inspections are described in a Notice of Violation (Enclosure 1, hereinafter referred to as Notice 1). A number of the violations adversely impacted the operability of safety related equipment. These violations are generally related to four broad categories, namely, the failure to: (1) adequately test equipment; (2) environmentally qualify equipment; (3) perform adequate safety reviews; and (4) either identify deficiencies, or take appropriate corrective actions in a timely manner to address known deficiencies, including design related issues. Some of the violations led to safety equipment being inoperable or degraded for extended periods contrary to technical specifications. The Notice also contains several violations of lesser significance which pertain to inadequate procedures or the failure to properly implement procedures.

The violations related to inadequate testing (Section I of Notice 1) involve failures to adhere to Technical Specifications (TS), which were failures to assure that various safety related instrument channels, logic actuation circuits, and safety related pump discharge check valves functioned as required. For example, Maine Yankee's testing process failed to detect a cut wire in the safety injection actuation circuit for a high pressure safety injection (HPSI) pump which would have prevented that pump from automatically starting, as required, during an accident. This condition had apparently existed since 1991, but was not detected until 1996 after prompting by the ISA. These violations are significant because testing requirements ensure the implementation of a "defense in depth" barrier for detecting inoperable safety related equipment and ensuring proper operation within expected tolerances.

The violations related to inadequate environmental qualification (Section II of Notice 1) involve: (1) 30 instruments which were either not qualified or could not be qualified for submergence during containment flooding following a loss of coolant accident (LOCA); and (2) the component cooling water pumps which were not qualified for a harsh environment in the turbine building. The failure to environmentally qualify the instruments in containment had potentially significant safety consequences. Operators rely on these instruments to monitor safety parameters such as steam generator water level during post-accident conditions. Failure of the instruments could hamper operator actions to mitigate the accident. For example, all four narrow range channels and one wide range channel for level indication for each of the three steam generators could have been unavailable due to submergence when the containment was flooded post-LOCA. These violations are also significant because the submergence issue was identified previously, yet was not effectively corrected. Seven components that were below the submergence level in the containment were identified in your Environmental Qualification (EQ) submittal dated October 31, 1980, and the NRC safety evaluation report (SER) dated June 1, 1981. However, corrective actions were not taken for several of these components in that they were still below the submergence level in 1996 and not environmentally qualified for such submergence.

The violations in Section III of Notice 1 pertain to your failure to perform adequate design basis safety review activities. Specifically, you failed to determine that a change to emergency power supplies for safeguards equipment, which provided for cross connecting of redundant 125Vdc vital buses, constituted an unreviewed safety question and required Commission review and approval prior to implementation. Section III also contains multiple examples of changes to the facility as described in the FSAR without performing safety evaluations required by 10 CFR 50.59, as well as a violation of 10 CFR 50.71(e) for failure to update the facility FSAR to reflect 27 changes to the facility implemented between 1980 and 1996. These violations are significant because they are indicative of Maine Yankee's failure to maintain strict control of the design basis of the facility.

The violations in Section IV of Notice 1 involve conditions adverse to quality that were either not identified or for which corrective actions were not taken in a timely manner commensurate with the safety significance of the condition. Most notably, although testing identified that one train of control room ventilation could not maintain a positive pressure in the control room, the condition was not corrected due to inadequate evaluation of the test results. Also, even though a design deficiency that could have rendered the containment spray building ventilation system inoperable was identified in 1991, the degraded condition was allowed to exist for 5 years due to failure to recognize the significance of the deficiency and weaknesses in Maine Yankee's corrective action programs. These violations are significant both because of their programmatic nature and the fact that Maine Yankee's inaction resulted in safety-related equipment being degraded or inoperable for extended periods.

In your letter dated February 28, 1997, in a response to NRC Inspection Report No. 50-309/96-16, and at the March 11, 1997, predecisional enforcement conference, you admitted all the violations that were the subject of that conference.

Each Section, I through IV, of Notice 1 constitutes a separate Severity Level III problem due to the safety significance and significant regulatory concern involved in each of the four broad categories of violations. Section V of the Notice includes multiple Severity Level IV violations pertaining to procedure or procedural implementation deficiencies.

<u>ECCS ANALYSIS ISSUES</u>

Based on the information developed during the OI investigations and provided in your written response and at the April 1998 conference, 6 violations associated with your small break-loss-of-coolant (SBLOCA) analyses (RELAP5YA)[1] are cited in Sections I and II of the second Notice of Violation (Enclosure 2, hereinafter referred to as Notice 2).

The two most significant violations involve your use of unacceptable evaluation models (EM) to determine emergency core cooling system (ECCS) performance for Cycle 14 and 15 operations, contrary to 10 CFR 50.46(a). The NRC interprets 10 CFR 50.46(a)(1)(I) to require that in order to be acceptable, EMs must be capable of analyzing the entire spectrum of break sizes that may result in a loss of coolant accident. Maine Yankee's EMs for operating Cycles 14 and 15 were inadequate because the Large-Break Loss-of-Coolant Accident (LBLOCA) and Small-Break Loss-of-Coolant Accident (SBLOCA) EMs were not, singly or combined, capable of analyzing or reliably analyzing the entire break spectrum, specifically, the region between $0.35ft^2$ and at least $0.6ft^2$.

Maine Yankee relied on engineering judgement to conclude that the ECCS analyses had identified and bounded the most severe postulated loss-of-coolant accidents. This judgment was not well-founded. Maine Yankee's LBLOCA EM had been run down to 0.6 ft^2, only after Cycle 14 had ended and after issuance of the January 1996 Order,[2] and was never demonstrated, by comparison to applicable experimental data, to reliably calculate ECCS performance in the small-break region. In addition, the technical report of Maine Yankee's contractor, acknowledges that, although the SBLOCA code, RELAP5YA,[3] was authorized to analyze break sizes up to 0.7 ft^2, it could only run the RELAP5YA EM up to 0.35 ft^2. Nonetheless, Maine Yankee reasoned that despite termination of the RELAP5YA EM at 0.35 ft^2, the limiting small break had been identified at 0.15 ft^2. Maine Yankee based this conclusion on a continual decrease in peak fuel cladding temperature after the 0.15 ft^2 break size, despite the fact that the previous SBLOCA EM, used for approximately 20 years in licensing basis SBLOCA analyses, had calculated the limiting SBLOCA break size at about 0.5 ft^2. In addition, increasing instability and oscillations occurred in the RELAP5YA SBLOCA EM as it approached $0.35ft^2$, where the model terminated following safety injection tank actuation. Therefore, it was unreasonable to assume that in analyzing the small-break spectrum up to only 0.35 ft^2, the most severe postulated SBLOCA, or limiting break, had been identified. Finally, Duke Engineering & Services (DE&S),[4] after reviewing development of the RELAP5YA SBLOCA EM by Yankee Atomic Electric Company (YAEC), acknowledged that it was unable to draw a definitive conclusion

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[1]RELAP5YA was the NRC-approved code for performing SBLOCA analyses at Maine Yankee

[2]On January 3, 1996 the NRC issued a Confirmatory Order Suspending Authority for and Limiting Power Operation and Containment pressure (Effective Immediately).

[3] RELAP5YA was the NRC-approved code for performing SBLOCA analyses at Maine Yankee

[4] DE&S purchased that portion of Yankee Atomic Electric Company (YAEC) that actually performed the SBLOCA analysis for Maine Yankee. DE&S and YAEC were served with a Demand for Information concurrent with the transmittal of the OI investigation results to Maine Yankee on December 19, 1997.

regarding the RELAP5YA peak cladding temperatures (PCTs) for the unanalyzed portion of the Maine Yankee SBLOCA spectrum. DE&S also acknowledged at the predecisional enforcement conference, that during the initial application of a new ECCS code, it is standard practice to perform an initial run from the top to the bottom of the break spectrum at regular intervals, rather than to perform a truncated run as was done in this case.

The NRC considers these violations, involving the failure to demonstrate by calculations using acceptable EMs the cooling performance of ECCS over a full spectrum of postulated LOCA break sizes, to be very significant. In fact, when this issue was first identified, the NRC issued an Order on January 3, 1996 modifying the facility operating license to derate the plant to the original licensed thermal power limit to regain the necessary assurance that ECCS performance was acceptable for continued operation. It was only after subsequent substantial additional review, that Maine Yankee demonstrated that there was no actual safety consequence of the failure to analyze the entire SBLOCA spectrum because the LBLOCA accident analyses contained the limiting condition, and therefore determined the facility operating limits.

The significance of these violations stems from the fact that for Cycle 14 operations, Maine Yankee operated the facility without having demonstrated that its ECCS systems were capable of mitigating the most severe postulated loss-of-coolant accident. While evidence indicates that individuals at Maine Yankee and Yankee Atomic Electric Company (YAEC) believed in good faith that they had identified the most limiting break size in the small-break spectrum, it was inappropriate to rely on unfounded engineering judgement to reach this conclusion. This judgment was not sufficient to demonstrate compliance with the 50.46 requirement that the ECCS cooling performance must be demonstrated by calculations over the entire break spectrum using acceptable EMs, especially when instabilities and oscillation of the peak cladding temperature and other parameters resulted in termination of the RELAP5YA SBLOCA computer run, and since the previous SBLOCA analyses of record identified the most limiting break in the region of the small-break spectrum that the new RELAP5YA SBLOCA EM could not calculate.

For these reasons, the two violations in Section I of Notice 2 have collectively been classified as a Severity Level II problem. Severity Level II violations or problems are defined in NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," (Enforcement Policy) as being of very significant regulatory concern.

It should be noted that the issues which constituted the Severity Level II problem cited in Section I of Notice 2 were considered by the Office of Investigations (OI) as willful acts on the part of Maine Yankee and YAEC. The staff, however, after thorough review of all the evidence concluded these violations were the result of poor judgment being exercised in both performing and reviewing analyses rather than on willful acts on the part of Maine Yankee personnel. As previously discussed, the RELAP5YA evaluation model was authorized to analyze break sizes from 0 to $0.7ft^2$, but the model was only able to calculate results up to $0.35ft^2$ which YAEC and Maine Yankee concluded covered the most limiting break size. The licensee used engineering judgment without sufficient basis for concluding that the most limiting break size had been identified. While the licensee's judgment was seriously flawed, and its actions constituted violations of 10 CFR 50.46, the NRC accepts Maine Yankee's explanations that it believed in good faith that it had sufficient justification to conclude that the limiting break for the SBLOCA region had been properly identified. Furthermore, it is clear from technical report, YAEC-1868, "Maine Yankee Small Break LOCA Analysis," which was incorporated by reference into the Core Performance Analysis Report, that Maine Yankee did not try to conceal the fact that it was unable to analyze the entire small-break spectrum. For these reasons, the staff concludes that the violations were not the result of willfulness on the part of Maine Yankee. As to YAEC, a Demand for Information was issued on December 19, 1997, requesting that YAEC and its successor, Duke Engineering & Services (DE&S), address why its deficient actions associated with the SBLOCA analysis should not be considered the result of willfulness, either deliberateness or careless disregard. The NRC has reviewed the responses of YAEC, DE&S and several individuals and is addressing the results of that review in separate correspondence issued concurrently with this action.

SANCTIONS

The violations described in both of the enclosed Notices of Violation appear to relate to the same fundamental underlying concerns with Maine Yankee's conduct of licensed activities. Many of these violations and underlying causes were longstanding and appeared to be caused by ineffective engineering analyses, review and processes which led to inadequate design and configuration control; a corrective action program which was fragmented; a quality assurance function which was not effective at both an individual and organizational level; and ineffective oversight as well as inadequate knowledge of vendor activities. The NRC's assessments, along with your own assessment as described at the March 1997 conference, found that Maine Yankee was a facility in which pressure to be a low-cost performer led to practices which overrelied on judgment, discouraged problem reporting, and accepted low standards of performance, as well as informality rather than rigorous adherence to program and procedural requirements. Lastly, Maine Yankee had become insular, failing to keep up with industry practice and failing to communicate adequately with the NRC.

The Commission considered a substantial civil penalty for the broad programmatic deficiencies described herein, and because Maine Yankee is still performing regulated activities important to safety. However, a civil penalty is not being proposed given the specifics of this case. Among the issues considered were: (1) Maine Yankee essentially replaced the entire management infrastructure since the time these problems occurred, and the new management has been effective in safely managing shutdown and decommissioning operations; (2) the fact that the Maine Yankee facility has been shutdown since December 5, 1996, was permanently retired on August 6, 1997, and the violations at issue here are not reflective of Maine Yankee's post shutdown and decommissioning performance; and, (3) unlike Haddam Neck in which a substantial civil penalty was imposed after declaring permanent retirement of the facility, Maine Yankee is not in the business of operating other nuclear power facilities. Accordingly, the NRC considers that civil penalties are not necessary in this case to provide the emphasis for a high standard of compliance in the future.

DISPOSITION OF REMAINING ECCS ANALYSIS ISSUES

Violation II.A of Notice 2, involving the failure to use the SBLOCA code specified in technical specifications to determine core operating limits and the subsequent provision of inaccurate information to the NRC in the Core Operating Limits Report (COLR), was characterized as the result of apparent careless disregard in our letter of December 19, 1997. After extensive staff review of the evidence, your April 6, 1998 response to our letter of December 19, 1997, and your presentation at the predecisional enforcement conference on April 23, 1998, the NRC has determined that the violation did not result from careless disregard. It is clear that Maine Yankee did not use the SBLOCA code specified by its Technical Specifications to determine Core Operating Limits (limits) for Cycle 13, and subsequently submitted inaccurate information to the NRC in its COLR that the limits had been developed by using the codes specified by the Technical Specifications. The NRC, however, accepts Maine Yankee's explanation that it believed in good faith that the LBLOCA was the most limiting accident scenario and that it had used the LBLOCA analysis only to determine Core Operating Limits. Furthermore, it is clear that Maine Yankee did not try to conceal the fact that it had not used the SBLOCA code specified in the Technical Specifications because the Core Performance Analysis Report, which was submitted to the NRC, clearly revealed that Maine Yankee had used the CE SBLOCA[5] code, rather than the RELAP5YA SBLOCA code that was specified in the Technical Specifications at the time. The NRC, therefore, does not conclude that there was willfulness, and absent willfulness, the NRC categorizes the Cycle 13 violation at Severity Level IV.

The NRC concludes that three additional violations associated with Maine Yankee's SBLOCA analyses occurred and have been classified at Severity Level IV. These violations involve: (1) & (2) use of unjustified ECCS penetration factors and cross flow resistance factors in the SBLOCA EM for cycle 14 and 15 operations; and (3) the use of an unacceptable ECCS EM for the 1993 analysis of a decrease in steam generator pressure. The unjustified ECCS penetration factors and cross flow resistance factors resulted from a calculational error in the application of the Alb-Chambre correlation. YAEC exercised unfounded judgment in selecting the penetration and resistance factors. Therefore, the resultant EM was unacceptable. The EM used to evaluate the effect of reduced steam generator pressure was a best estimate model, not an Appendix K model as approved by the staff. Furthermore, the methods used to determine fission product decay heat and two-phase discharge flow were different from those required by 10 CFR Part 50, Appendix K. For these reasons, the EM used for the reduced steam generator pressure analysis was unacceptable. These violations are cited in Sections II.B - II.D of Notice 2.

The remaining three apparent violations associated with ECCS analyses, described in our December 19, 1997, letter, will not be cited. The NRC concluded that the use of the CE SBLOCA code in determining core operating limits for Cycle 12 operations and the statement to the NRC in the Core Operating Limits Report that the analyses specified by the Technical Specifications were used, did not constitute a violation of NRC requirements

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[5]The CE SBLOCA code had been the approved method for demonstrating compliance with 10 CFR 50.46 at Maine Yankee prior to staff approval of the RELAP5YA code.

because the Technical Specification amendment requiring the use of the RELAP5YA SBLOCA code became effective beginning with Cycle 13 operations. The apparent violations associated with the failure to develop and maintain a complete and accurate Core Performance Analysis Report (CPAR) for Cycles 14 and 15 were not sustained because, upon further review, the NRC concluded that only an abstract of a document referenced by the CPAR was misleading and the document, when read in its entirety, clearly demonstrates there was no intent to mislead.

ATMOSPHERIC STEAM DUMP VALVE

At the enforcement conference, Maine Yankee acknowledged it was responsible for the acts of its employees. However, Maine Yankee contended that the submission of known inaccurate information by a working level employee did not constitute a willful violation on the part of Maine Yankee. Nonetheless, the NRC contends the 1986 submission of materially inaccurate information relative to the capacity of the facility's atmospheric steam dump valve was willful. However, the NRC has decided, given the circumstances of this case, including the age of the violation, to exercise discretion pursuant to section VII.B.6 of the Enforcement Policy and not cite the violation described in our letter of December 19, 1997.

SAFETY SYSTEM LOGIC TESTING

Based on the findings of the OI investigation and the information provided in your response and at the April 1998 conference, one violation is being cited associated with safety system logic testing as set forth in Section III of Notice 2 . Two engineers violated station test procedures (a Technical Specification violation) and caused a violation of 10 CFR 50.9, "Completeness and accuracy of information." Work orders specified that specific contacts be verified as open with a volt-ohm meter. The field engineers performing the tests, however, obtained a quantifiable electrical resistance value when they used the volt-ohm meter, indicating a problem. Because of a resistor in the circuit, it was not possible to verify an open contact with the volt-ohm meter. The engineers visually verified the open contacts without first stopping the test and following the process required by the Technical Specifications for implementing a minor technical change (MTC) to the procedure. The engineers then signed the work order as completed according to test procedures. OI concluded that the violations were deliberate. The staff concludes, however, based on all the evidence, that the engineers believed in good faith that they were not required to implement a MTC. The engineers had executed several other MTCs as they encountered other difficulties with the same work order. Also, the two engineers had the authority to approve the MTCs themselves and the engineers did not believe that an MTC was required in this particular instance. Thus, it does not appear that the engineers attempted to circumvent the MTC process. Therefore, the staff concludes that it is not unreasonable for the engineers to have believed that they had the authority to document the execution of the steps in the manner they did and consequently, the act was not a deliberate or willful violation of station procedures. Absent willfulness, this violation is categorized at Severity Level IV.

You are required to respond to this letter and should follow the instructions prescribed in the enclosed Notices when preparing your response. The NRC will consider your response, in part to determine whether further enforcement is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be placed in the NRC Public Document Room.

Sincerely,

Hubert J. Miller
Regional Administrator

Docket No. 50-309
License No. DPR-36

Enclosures:
(1) Notice of Violation (Notice 1) (EA Nos. 96-299, 96-320, 97-034, 97-147)
(2) Notice of Violation (Notice 2)(EA 96-397, 97-375, 97-559)

cc w/encl:
R. Fraser, Director - Engineering
J. M. Block, Attorney at Law
P. L. Anderson, Project Manager (Yankee Atomic Electric Company)
L. Diehl, Manager of Public and Governmental Affairs
T. Dignan, Attorney (Ropes and Gray)
G. Zinke, Director, Regulatory Affairs
W. Odell, Director, Operations
M. Ferri, Director, Decommissioning
M. Lynch, Esquire, MYAPC
P. Dostie, State Nuclear Safety Inspector
P. Brann, Assistant Attorney General
U. Vanags, State Nuclear Safety Advisor
C. Brinkman, Combustion Engineering, Inc.
W. D. Meinert, Nuclear Engineer
First Selectmen of Wiscasset
M. Kilkelly, State Senator, Chair - Community Advisory Panel
Maine State Planning Officer - Nuclear Safety Advisor
State of Maine, SLO Designee
State Planning Officer - Executive Department
Friends of the Coast

NOTICE OF VIOLATION
(NOTICE 1)

Maine Yankee Atomic Power Company	Docket No.	50-309
Maine Yankee Atomic Power Station	License No.	DPR-36
	EA Nos.	96-299; 96-320; 97-034; 97-147

During NRC inspections conducted between July 15, 1996 and August 26, 1996, and between December 8, 1996 and March 15, 1997, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

I. VIOLATIONS RELATED TO INADEQUATE TESTING

A. Technical Specification (TS) 3.9.B, "Engineered Safeguards Features Actuation System," Table 3.9-2 No. 1, "Safety Injection," requires, in part, a minimum of 3 operable channels for both high containment pressure and low pressurizer pressure per safety injection actuation system (SIAS) subsystem to be operable whenever automatic initiation of Engineered Safeguards Feature (ESF) systems is required to be operable. TS 3.6.C requires, in part, two operable and redundant emergency core cooling system (ECCS) trains including one in each high pressure safety injection (HPSI) pump subsystem, an ESF system, to be operable whenever the reactor is in a power operation condition.

Contrary to the above, during periods of power operation from December 1991 until August 17, 1996, there were no operable channels of high containment pressure or low pressurizer pressure in the 'A' subsystem of the SIAS. Specifically, the 'A' HPSI pump would not have automatically started in response to a SIAS signal (high containment pressure or low pressurizer pressure) due to a missing wire in the HPSI pump circuit. (01013)

B. TS 4.0, "Surveillance Requirements," requires that each surveillance requirement in Section 4 be performed within the specified surveillance interval.

1. TS 4.1, "Instrumentation and Control," requires, in part, that testing of engineered safeguards system logic channels be performed as specified in Table 4.1-2. TS Table 4.1.2, requires, in part, that Channel 3, SIAS actuation relays; Channel 10, refueling water tank level recirculation actuation signal (RAS) initiation; Channel 20, feedwater trip system; and Channel 21, emergency feedwater (EFW) initiation, be tested at least once every 18 months.

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Contrary to the above, prior to August 18, 1996, surveillance tests required by TS 4.1, Table 4.1-2, were not performed at least once every 18 months. Specifically:

a. Channel 3 - HPSI pump start signals for SIAS and undervoltage (UV) conditions were not tested independently; and the dual function swing pump (P-61S) was not tested as a low pressure safety injection (LPSI) and containment spray pump for UV and SIAS actuation;

b. Channel 10 - Manual initiation of RAS was not tested; and the automatic trip of swing pump (P-61S), when used as a LPSI pump, was not tested;

c. Channel 20 - The SIAS permissive was not adequately tested in that the main feedwater pump, condensate pump, and heater drain pump trip systems were not tested with a SIAS coincident with a steam generator low pressure signal; and

d. Channel 21 - Emergency feed water pump circuit breaker closure was not tested. (01023)

2. TS 4.5, "Emergency Power System Periodic Testing," A.2, "Diesel Generators," requires, in part, testing of the diesel generators (DGs) during each refueling interval that demonstrates their readiness to start automatically and restore power to vital equipment on loss of all normal a-c station service power supplies.

Contrary to the above, during each refueling interval prior to August 18, 1996, tests required by TS 4.5.A.2 were not being performed in that emergency bus loading and load shedding, necessary to demonstrate the DGs readiness to start automatically and restore power to vital equipment on loss of all normal a-c station service power supplies, was not adequately tested. Specifically, for the following vital equipment:

a. Service water (SW) pumps P-29B and P-29C were not verified to remain operating on the bus if they were the only available pumps in the train.

b. Primary component cooling (PCC) pump P-9B was not tested as the preferred pump.

c. Secondary component cooling (SCC) pump P-10B was not tested as the preferred pump. (01033)

3. TS 4.6, "Periodic Testing," D.1.a, "Feedwater Trip System, Main Feedwater Pumps," requires that each main feedwater pump,

condensate pump, and heater drain pump trip system shall be tested during each refueling interval by tripping the actuation circuitry with a safety injection signal coincident with a steam generator low pressure signal.

Contrary to the above, during each refueling interval prior to August 18, 1996, the testing required by TS 4.6.D.1.a was not performed to verify tripping of each main feedwater pump, condensate pump and heater drain pump circuit breaker with a safety injection signal coincident with a steam generator low pressure signal. (01043)

C. TS 4.7.A, "Inservice Inspection and Testing of Safety Class Components," requires, in part, the establishment of an "Inservice Inspection Program" that meets the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Inservice Testing of Pumps and Valves," for safety class 3 pressure retaining components.

10 CFR 50.55a(f), "Inservice testing requirements," requires, in part, that safety related valves must meet the requirements applicable to components which are classified as ASME Code Class 3 set forth in section XI of the ASME Boiler and Pressure Vessel Code.

ASME Code, Section XI, IWV-3520, "Check Valve Tests," requires that valves normally open during plant operation whose function is to prevent reversed flow, shall be tested in a manner that proves that the disk travels to the seat promptly on cessation or reversal of flow.

Contrary to the above, as of August 18, 1996, inservice testing for 15 safety class 3 pressure retaining check valves that were located at the discharge of safety related pumps did not meet the requirements of the ASME Code, Section XI. This inservice testing failed to demonstrate that the standby pump's discharge check valves, which are normally open during operation and whose function is to prevent reversed flow, would properly close on the cessation or reversal of flow which would be necessary to prevent short-cycling of the operating pump. Specifically, the following safety class 3 valves were not adequately tested:

1. Charging/HPSI pump discharge check valves CH-10, 19 and 26;
2. EFW pump discharge check valves EFW-15, and 314;
3. LPSI pump discharge check valves LPSI-50 and 51;
4. PCC pump discharge check valves PCC-6 and 13;
5. SCC pump discharge check valves SCC-7 and 14; and

6. SW pump discharge check valves SW-1,4,7 and 10. (01053)

These violations in Section I represent a Severity Level III problem (Supplement I).

II. VIOLATIONS RELATED TO ENVIRONMENTAL QUALIFICATION

10 CFR 50.49(d) requires, in part, that the licensee shall include in a qualification file the environmental conditions, including temperature, humidity, and submergence, at the location where electrical equipment important to safety covered by 10 CFR 50.49 must perform.

10 CFR 50.49(j) requires that a record of the environmental qualification must be maintained in an auditable form to permit verification that each item of electric equipment important to safety is qualified for its application and meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function.

10 CFR 50.49(f) requires each item of electric equipment important to safety to be environmentally qualified by (1) testing of identical or similar equipment under identical or similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable, (2) experience with identical or similar equipment under similar conditions with a supporting analysis, or (3) analysis in combination with partial type-test data that supports the analytical assumptions and conclusions.

10 CFR 50.49(b) defines electric equipment important to safety within the scope of 10 CFR 50.49 as safety-related electric equipment, non-safety-related electric equipment whose failure under postulated accident conditions could prevent safety related equipment from accomplishing the functions identified in 10 CFR 50.49(b)(1), and certain post-accident monitoring equipment.

10 CFR 50.49(e) specifies the conditions and other location dependent considerations that the electric equipment qualification program must be based upon. These conditions and considerations include, in part, temperature and pressure, humidity, and submergence, as applicable, during and after the most severe accident environment for which electrical equipment important to safety must remain functional.

A. Contrary to the above, as of August 2, 1996, the qualification files for 30 items of electric equipment important to safety inside the reactor containment did not permit verification that the items were qualified for their applications and met their specified performance requirements when subjected to submergence, a condition predicted to be present when they must perform their safety functions after a loss of coolant accident (LOCA). The qualification files for these 30 items of electric equipment did not include the correct submergence level at the location where they must meet their specified performance requirements. Specifically, safety-related valve limit switches and associated pigtails, Rosemount transmitters and associated electrical connectors, and certain Rockbestos cables were not qualified for post-LOCA submergence in that there were no documents in Maine Yankee's environmental qualification (EQ) file to demonstrate qualification of the items by testing or a combination of testing, experience, or partial type-test data with analysis. (02013)

B. Contrary to the above, as of March 11, 1997, the qualification files for two PCC pump motors and two SCC pump motors, safety related components, did not permit verification that they were environmentally qualified to remain functional during and following a high energy line break (HELB) in the turbine building, which is the most severe design basis event at their location during or after which they must remain functional. Specifically, there were no documents in the Maine Yankee EQ file to demonstrate that the PCC and SCC pump motors were qualified for high temperature and high humidity resulting from a HELB. (02023)

These violations in Section II represent a Severity Level III problem (Supplement I).

III. VIOLATIONS RELATED TO INADEQUATE SAFETY REVIEW

A. 10 CFR 50.59, "Changes, tests and experiments," permits the licensee, in part, to make changes in the facility and procedures as described in the safety analysis report without prior Commission approval provided the change does not involve an unreviewed safety question (USQ). A proposed change shall be deemed to involve a USQ, in part, if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created. The licensee shall maintain records of changes in the facility and these records must include a written safety evaluation which provides the bases for the determination that the change does not involve an USQ.

1. Contrary to the above, in May 1992, Maine Yankee made a change to procedures as described in the FSAR that involved an USQ without prior Commission approval due to an inadequate safety evaluation. Specifically, Maine Yankee established procedure 1-22-2, "AC and DC Vital Bus Operation," which allowed cross connecting redundant 125 Vdc vital buses for up to 72 hours during plant operation. This was a change from FSAR Appendix A, Criterion 39, "Emergency Power for ESFs," which provides, in part, that the alternate power systems be provided and designed with adequate independence and redundancy to permit the functioning required of the ESFs and, as a minimum, that the onsite power system shall independently provide required capacity assuming a single failure. With the 125 Vdc buses cross connected, all 125 Vdc power to the ESFs could have been lost due to a single failure. This created the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report and represents an USQ. As of August 30, 1996, the safety evaluation performed for this procedure change was inadequate in that it failed to identify this USQ. (03013)

2. Contrary to the above, Maine Yankee made the following changes to the facility as described in the FSAR without performing a written safety evaluation for these changes to provide the basis for the determination that the changes did not involve a USQ, each of which constitutes an individual violation:

   a. In January 1996, Maine Yankee restricted the maximum SW operating temperatures to 70.2 °F for component cooling water (CCW) heat exchangers E-4B and E-5A, and 78.5 °F for CCW heat exchangers E-4A and E-5B to support design basis post-LOCA condition heat removal capability. This was a change from FSAR Section 9.4.1 which assumed SW inlet temperatures of 80 °F for E-4B and E-5A, and 90 °F for E-4A and E-5B. As of August 30, 1996, no safety evaluation had been performed for the change in SW operating temperatures. (03023)

   b. On February 21, 1997, Maine Yankee changed the layout within the protected area by installing and filling a 1000 gallon propane tank contrary to FSAR, Section 1.3, "Plant Description Summary." This addition had the potential to damage the circulating water (CW) pumphouse if it exploded, and could negatively affect both trains of the SW system since the SW pumps are located in the CW pumphouse. As of March 5, 1997, no safety evaluation had been performed for the propane tank. (03033)

   c. On March 11, 1997, a drain hose was temporarily installed on a spent fuel pool pump suction pipe which was contrary to the configuration of the spent fuel pool cooling system as shown in plant drawings and the FSAR, Section 9.8, "Fuel Pool Cooling System." As of March 15, 1997, no safety evaluation had been performed for this change in the configuration of the spent fuel pool cooling system. (03043)

   d. As of August 30, 1996, no safety evaluation had been performed for approximately 89 equipment and procedure changes that were made to equipment and procedures described in the FSAR. These changes were identified by Maine Yankee as a result of an initiative to upgrade the FSAR and are listed in the "Final Safety Analysis Report (Revision 13) Maine Yankee FSAR Update (MFU) Status Report." (03053)

B. 10 CFR 50.71(e) requires the licensee to update the FSAR to assure that the information included in the FSAR contains the latest material developed. Updates must be filed annually or 6 months after each refueling outage. The updates must reflect all changes made in the facility or procedures as described in the FSAR up to a maximum of 6 months prior to the date of filing.

Contrary to the above, as of August 1996, the FSAR was not updated to reflect 27 changes made to the facility as a result of Engineering Design Change Requests and Plant Design Change Requests that were implemented between 1980 and August 1996. These changes were identified by Maine Yankee as a result of an initiative to upgrade the FSAR and are listed in the "Final Safety Analysis Report (Revision 13) Maine Yankee FSAR Update (MFU) Status Report." (03063)

These violations in Section III represent a Severity Level III problem (Supplement I).

IV. <u>VIOLATIONS ASSOCIATED WITH INADEQUATE CORRECTIVE ACTIONS</u>

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

A. Contrary to the above, from October 31, 1995, until August 16, 1996, the inability of train 'A' of the control (CR) breathing air system to maintain a positive pressure in the control room during accident conditions was not corrected. Specifically, during testing of the 'A' train of the CR breathing air system on October 31, 1995, in accordance with Surveillance Procedure 3.17.5, pressure in the CR was slightly negative. These test results indicated that the 'A' train of control room ventilation system (CRVS) was not operable, a significant condition adverse to quality. Maine Yankee did not take measures to assure that the cause of this condition was determined and did not take corrective actions to preclude repetition. No action was taken to restore operability of the 'A' train of CRVS prior to making the reactor critical on January 11, 1996 contrary to Technical Specifications.(04013)

B. Contrary to the above, as of August 3, 1996, a significant condition adverse to quality identified in 1991 had not been corrected. Specifically, a loss of non safety-related instrument air could cause the air operated dampers (VP-A-56 and VP-A-57) in the containment spray building (CSB) fans' ducts to fail shut, rendering the fans (FN 44A and 44B) incapable of performing their safety function of providing ventilation to the low pressure safety injection (LPSI) and containment spray pumps and heat exchangers area (i.e., by removing more than 10,000 cfm of air as specified in the Maine Yankee FSAR, Section 9.13.2.3) in the CSB. Without adequate ventilation, the LPSI and containment spray pump motors could fail due to overheating. This potential to lose CSB safety-related fans was identified during a ventilation system review by engineering in 1991 and was not corrected until August 3, 1996. (04023)

C. Contrary to the above, between 1994 and 1996, actions to determine the cause and preclude repetition of icing and clogging of the CSB heating, ventilating, and air conditioning (HVAC) unit, HV-7, a significant condition adverse to quality, were inadequate. Specifically, the clogging occurred at least three times during that period, and even though corrective actions were taken, they were not effective in precluding repetition of the adverse condition. The clogging of the HVAC unit caused the CSB ventilation system (a support system for the LPSI and containment spray systems) to be inoperable, thereby potentially rendering both trains of LPSI and containment spray systems inoperable. (04033)

D. Contrary to the above, as of August 30, 1996, actions to determine the cause and preclude repetition of Auxiliary Feedwater (AFW) control system failures, a significant condition adverse to quality, were inadequate. Specifically, repetitive problems between 1992 and 1996 resulted in degraded reliability for the AFW pump to respond to a start/run demand. Even though corrective actions were taken, they did not preclude repetition of the control system problems. (04043)

E. Contrary to the above, as of April 1996, a design deficiency, which was a condition adverse to quality, involving the plant being outside of its design basis for a turbine hall flood, had not been promptly corrected. Specifically, during the Service Water System Operational Performance Inspection in 1994, Maine Yankee identified that the plant was outside of the design basis for a turbine hall flood in that during a design basis flood in the turbine building, safety-related equipment in the control room, the DG room, and the turbine building would be rendered inoperable. (04053)

F. Contrary to the above, from December 20, 1996 until February 21, 1997, Maine Yankee did not promptly establish compensatory corrective actions regarding an identified condition adverse to quality that would challenge the operability of the SW system. Specifically, in a ventilation system assessment report, dated December 20, 1996, Maine Yankee identified that a loss of ventilation in the circulating water pumphouse during periods of extreme cold temperatures, could create potentially freezing conditions for SW system components. Frozen water in stagnant lines could restrict flow to the SW pump bearings and gland cooling or create the potential for a line break. Compensatory actions to prevent freezing in the circulating water pump house were not taken until February 21, 1997. (04063)

These violations in Section IV represent a Severity Level III problem (Supplement I).

V. SEVERITY LEVEL IV VIOLATIONS

TS (TS) 5.8.2.a requires, in part, that written procedures, as recommended in Appendix A of Regulatory Guide 1.33, (Rev. 2), February 1978, shall be established and implemented.

A. Regulatory Guide 1.33, Appendix A, section 1, "Administrative Procedures," states, in part, that the maintenance of minimum shift complement; log entries; and authorities and responsibilities for safe operation and shutdown should be covered by written procedures.

1. Contrary to the above, as of August 30, 1996, Maine Yankee had not established procedural requirements, such that, in the event of a fire coincident with a medical emergency, the minimum control room staffing required by TS Section 5.2.2/Table 5.2-1, would be satisfied. Specifically, only two Senior Reactor Operators (SROs) were required to be on duty. As a result, there would be no SRO in the control room, as required, if the two SROs on duty had to respond to a fire and a medical emergency concurrently. (05014)

This is a Severity Level IV violation (Supplement 1).

2. Maine Yankee administrative procedure No. 1-200-10, "Conduct of Operations", section 4.13, "Operability Assessment," specifies that if there is not a reasonable expectation that the equipment is operable, then the equipment shall be declared inoperable. Section 4.13 also specifies that an operability determination must assess the ability of the equipment to perform its intended safety action in the accident environment it would be subjected to when it would be called upon to do so and that tests or partial tests should be used for completing operability assessments.

Contrary to the above, on August 17, 1996, administrative procedure No. 1-200-10 was not implemented in that the Operations Manager issued a memorandum that stated that TS testing discrepancies did not render the HPSI and containment spray swing pumps inoperable. This was contrary to the requirements of procedure 1-200-10 in that without performance of the testing that verifies that the pumps would perform their intended safety action when called upon, there was no reasonable assurance that the pumps were operable. (06014)

This is a Severity Level IV violation (Supplement 1).

B. Regulatory Guide 1.33, Appendix A, section 9, "Procedures for Performing Maintenance," states, in part, that maintenance that can affect the performance of safety-related equipment should be performed in accordance with written procedures or documented instructions; that preventive maintenance schedules should be developed to specify inspection or replacement of parts that have a specific lifetime; and that general procedures for the control of maintenance should include the method for obtaining permission and clearance for work.

1. Maine Yankee maintenance procedure 5-9-3, "Maintenance of Emergency and Auxiliary Feedwater Pumps," Rev. 4, section 6.3.11

specifies the inspection of parts to determine if they are suitable for reuse. Maintenance procedure 5-9-3, section 6.3.12 and preventive maintenance (PM) card, M-18-3X-J, "P-25A Emergency Feedwater (EFW) Pump and Motor," specify performance of a liquid penetrant or magnetic particle examination of the cast iron diffuser assembly.

Contrary to the above, during the 1995 overhaul of the EFW pump P-25A, maintenance procedure 5-9-3 and PM card M-18-3X-J were not implemented in that no liquid penetrant or magnetic particle examinations were performed prior to reuse of the cast iron diffuser assembly. (07014)

This is a Severity Level IV violation (Supplement 1).

2. Maine Yankee maintenance procedure 0-16-3, "Work Order Process," Rev. 10, Attachment A, section I.A specifies that work performed on safety class equipment must be performed in accordance with procedures that provide specific information for the intended actions.

Contrary to the above, as of August 7, 1996 Maine Yankee failed to establish procedures that provided specific instructions to reinstall fastener lock wire as intended and, as a result, lock wire was not reinstalled after maintenance was performed on the following safety class equipment: Reactor coolant system loop No. 3 stop valve's motor operated valve actuator mounting fasteners and In-core instrumentation seal housings F-11, V-11, N-17, D-11, and T-16. (08014)

This is a Severity Level IV violation (Supplement 1).

3. Maintenance procedure 0-16-3, sections 6.5 and 6.6 specify that, if necessary, equipment shall be tagged out prior to commencing work and that maintenance governed by this procedure shall not commence until the Work Order has received all required reviews and approvals. Work Order No. 94-02278-01 for replacement of a pipe support specified that a white tagging order was required for SW pump P-29C to be out of service.

Contrary to the above, on August 13, 1996, procedure 0-16-3 was not implemented in that maintenance personnel removed a seismically qualified pipe support on a seal water line for SW pump P-29C without a white tagging order being issued to tag the pump out of service. Removal of the existing pipe support caused the pump to be inoperable and; therefore, out of service. (09014)

This is a Severity Level IV violation (Supplement 1).

Pursuant to the provisions of 10 CFR 2.201, Maine Yankee Atomic Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region , and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at King of Prussia, Pennsylvania
this 8th Day of October 1998

NOTICE OF VIOLATION
(NOTICE 2)

Maine Yankee Atomic Power Company
Maine Yankee Atomic Power Station

Docket 50-309
License No. DRP-36
EA 96-397; 97-374; 97-559

Based on investigations by the NRC Office of Investigations (OI), conducted between December 1995 and October 1997, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions, NUREG-1600, the violations are listed below:

I. PRINCIPAL PROBLEM RELATED TO INADEQUATE SMALL-BREAK-LOSS-OF-COOLANT ANALYSES (OI Report No. 1-95-050)

A. VIOLATION RELATING TO INABILITY TO ANALYZE ENTIRE BREAK SPECTRUM FOR CYCLE 14

10 C.F.R. § 50.46(a)(1) requires, in part, that emergency core cooling system (ECCS) performance must be calculated with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.

10 C.F.R. Part 50, Appendix K, Section II.4, requires that to the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

Contrary to the above, from October 14, 1993, through January 25, 1995 (during Cycle 14 operations), and in the Cycle 14 Core Performance Analysis Report (CPAR) submitted August 25, 1993, Maine Yankee Atomic Power Company (MYAPCo) used unacceptable models to calculate ECCS performance and failed to calculate a number of postulated loss-of-coolant accidents of different sizes, locations and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents were calculated. Specifically, there was a portion of the small-break spectrum between .35 ft^2 and at least .6 ft^2 for which no acceptable evaluation model was capable of calculating cooling performance or reliably calculating cooling performance. MYAPCo calculated Small-Break Loss-of-Coolant Accident (SBLOCA) ECCS performance with the code described in "YAEC 1300P, RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, Volumes 1, 2 3," dated October 1982 (RELAP5YA) and the plant-specific RELAP5YA SBLOCA evaluation model described in YAEC-1868, "Maine Yankee Small Break LOCA Analysis" (both of which were described as an Appendix K approach to RELAP5YA). MYAPCo calculated SBLOCA ECCS performance only up to the .35 ft^2 break size because the RELAP5YA SBLOCA evaluation model documented in

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YAEC-1868 was incapable of calculating ECCS performance for break sizes of and greater than 0.35 ft^2 as a result of the model's terminating after the safety injection tank actuation due to numerical convergence errors for the break size of .35 ft^2. MYAPCo calculated Large-Break Loss-of-Coolant (LBLOCA) ECCS Performance with the LBLOCA analysis described in YAEC-1160, "Application of Yankee WREM-Based Generic PWR ECCS Evaluation Model to Maine Yankee", dated July 1978 (WREM). Although the WREM LBLOCA evaluation model was subsequently demonstrated in 1996 to be capable of calculating ECCS performance down to the .6ft^2 break size, the WREM LBLOCA evaluation model was not used to calculate ECCS performance in the small-break region for Cycle 14, and would not have been acceptable to calculate ECCS performance for break sizes in the small-break region of 0.6 ft^2 and above because the evaluation model was not compared to applicable experimental data to demonstrate its reliability in calculating ECCS performance in the small-break region. (01012)

B. VIOLATION RELATING TO INABILITY TO ANALYZE ENTIRE BREAK SPECTRUM FOR CYCLE 15

10 C.F.R. § 50.46(a)(1) requires, in part, that emergency core cooling system (ECCS) performance must be calculated with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.

10 C.F.R. Part 50, Appendix K, Section II.4, requires that to the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

Contrary to the above, in the Cycle 15 Core Performance Analysis Report (CPAR) submitted December 1, 1995, Maine Yankee Atomic Power Company (MYAPCo) used unacceptable models to calculate ECCS performance and failed to calculate a number of postulated loss-of-coolant accidents of different sizes, locations and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents were calculated. Specifically, there was a portion of the small-break spectrum between .35 ft^2 and at least .6 ft^2 for which no acceptable evaluation model was capable of calculating cooling performance or reliably calculating cooling performance. MYAPCo calculated Small-Break Loss-of-Coolant Accident (SBLOCA) ECCS performance with the code described in "YAEC 1300P, RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, Volumes 1, 2 3," dated October 1982 (RELAP5YA) and the plant-specific RELAP5YA SBLOCA evaluation model described in YAEC-1868, "Maine Yankee Small Break LOCA Analysis" (both of which were described as an Appendix K approach to RELAP5YA). MYAPCo calculated SBLOCA ECCS performance only up to the .35 ft^2 break size because the RELAP5YA SBLOCA evaluation model documented in

YAEC-1868 was incapable of calculating ECCS performance for break sizes of and greater than 0.35 ft^2 as a result of the model's terminating after the safety injection tank actuation due to numerical convergence errors for the break size of .35 ft^2. MYAPCo calculated Large-Break Loss-of-Coolant (LBLOCA) ECCS Performance with the LBLOCA analysis described in YAEC-1160, "Application of Yankee WREM-Based Generic PWR ECCS Evaluation Model to Maine Yankee", dated July 1978 (WREM). Although the WREM LBLOCA evaluation model was subsequently demonstrated in 1996 to be capable of calculating ECCS performance down to the .6ft^2 break size, the WREM LBLOCA evaluation model was not used to calculate ECCS performance in the small-break region for Cycle 15, and would not have been acceptable to calculate ECCS performance for break sizes in the small-break region of 0.6 ft^2 and above because the evaluation model was not compared to applicable experimental data to demonstrate its reliability in calculating ECCS performance in the small-break region. (01022)

These violations in Section I represent a Severity Level II problem (Supplement I).

II OTHER VIOLATIONS RELATED TO INADEQUATE SMALL-BREAK-LOSS-OF-COOLANT ANALYSES (OI Report No. 1-95-050)

A. VIOLATION RELATING TO OPERATING CYCLE 13

Technical Specification (TS) 5.14.2, "Core Operating Limits Report," for the Maine Yankee Atomic Power Station (MYAPS) requires, in part, that analytical methods used to determine operating limits shall be limited to those previously reviewed and approved by NRC, as listed by TS 3.10. TS.3.10 specifies a Small-Break Loss-of-Coolant (SBLOCA) analysis, "YAEC 1300P, RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, Volumes 1, 2, 3, dated October 1982" (RELAP5YA). TS.3.10. does not specify any SBLOCA analysis produced by Combustion Engineering Corporation (CE).

10 C.F.R. § 50.9(a) requires, in part, that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

Contrary to the above, between April 19, 1992 and July 7, 1993 (during Cycle 13 operations), Maine Yankee Atomic Power Company did not determine operating limits for Cycle 13 operations using the RELAP5YA SBLOCA analysis required by TS 5.14.2. In fact, a Combustion Engineering (CE) SBLOCA code was used to prepare the reload analysis, as stated in the Core Performance Analysis Report for Cycle 13 at Section 5.5.5.3. In addition, on April 7, 1992, Maine Yankee Atomic Power Company (MYAPCo) provided to the Commission MYAPCo's Cycle 13 Core Operating Limits Report (COLR), which contained inaccurate information material to the NRC. The COLR stated that MYAPCo used analytical methods listed in TS 5.14 to determine operating limits. In fact, MYAPCo used a CE SBLOCA analysis, which was not listed in TS 5.14. The SBLOCA analysis listed by TS 5.14 is "YAEC 1300P, RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, Volumes 1, 2, 3, dated October 1982" (RELAP5YA). This inaccurate information was material to the NRC because it was a representation that RELAP5YA, which had been approved for application to MYAPS pursuant to the Three Mile Island Action Plan, Item II.K.3.30 (NUREG 0737), had been used to establish core operating limits for Cycle 13 operations. (02014)

This is a Severity Level IV violation (Supplement 1)

B. VIOLATION RELATED TO IMPROPER APPLICATION OF ALB-CHAMBRE CORRELATION FOR CYCLE 14

10 C.F.R. § 50.46(a)(1) requires, in part, that emergency core cooling system (ECCS) performance must be calculated with an acceptable evaluation model.

Contrary to the above, from October 14, 1993, through January 25, 1995 (during Cycle 14 operations), and in the Cycle 14 Core Performance Analysis Report (CPAR) submitted August 25, 1993, MYAPCo calculated ECCS performance for SBLOCAs with an unacceptable evaluation model. MYAPCo used the ECCS code described in YAEC-1300P, "RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, Volumes 1, 2, 3," dated October 1982 (RELAP5YA), and the plant-specific RELAP5YA SBLOCA evaluation model described in YAEC-1868, "Maine Yankee Small Break LOCA Analysis" (YAEC-1868). RELAP5YA as applied was not an acceptable evaluation model because the nodalization model of YAEC-1868 incorrectly applied the Alb-Chambre correlation, resulting in the unjustified use of large penetration factors and a large cross flow resistance factor in the split downcomer nodalization. (02024)

This is a Severity Level IV violation (Supplement 1)

C. VIOLATION RELATED TO IMPROPER APPLICATION OF ALB-CHAMBRE CORRELATION FOR CYCLE 15

10 C.F.R. § 50.46(a)(1) requires, in part, that emergency core cooling system (ECCS) performance must be calculated with an acceptable evaluation model.

Contrary to the above, in the Cycle 15 Core Performance Analysis Report (CPAR) submitted December 1, 1995, MYAPCo calculated ECCS performance for SBLOCAs with an unacceptable evaluation model. MYAPCo used the ECCS code described in YAEC-1300P, "RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, Volumes 1, 2, 3," dated October 1982 (RELAP5YA), and the plant-specific RELAP5YA SBLOCA evaluation model described in YAEC-1868, "Maine

Yankee Small Break LOCA Analysis" (YAEC-1868). RELAP5YA as applied was not an acceptable evaluation model because the nodalization model of YAEC-1868 incorrectly applied the Alb-Chambre correlation, resulting in the unjustified use of large penetration factors and a large cross flow resistance factor in the split downcomer nodalization. (02034)

This is a Severity Level IV violation.

D. VIOLATION RELATING TO ANALYSIS OF REDUCED STEAM GENERATOR PRESSURE FOR CYCLE 14

10 C.F.R. § 50.46(a)(1) requires, in part, that emergency core cooling system (ECCS) performance must be calculated with an acceptable evaluation model. 10 C.F.R. § 50.46(a)(1)(ii) provides that an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models.

Contrary to the above, in a January 1993 analysis of a decrease in steam generator pressure, performed pursuant to the requirements of 10 C.F.R. § 50.59, MYAPCo used an unacceptable evaluation model to calculate SBLOCA ECCS performance. MYAPCo used a Best Estimate (BE) plant-specific evaluation model (described in an August 1, 1990, report produced by Yankee Atomic Electric Company) to implement the SBLOCA code described in YAEC 1300P, "RELAP5YA: A Computer Program for Light Water Reactor System Thermal-Hydraulic Analysis, Volumes 1, 2, 3," dated October 1982 (RELAP5YA). In January 1989, the NRC transmitted its Safety Evaluation Report approving RELAP5YA for application to Maine Yankee Atomic Power Station as an Appendix K model, not as a BE model. Furthermore, contrary to 10 C.F.R. Part 50, Appendix K, the BE evaluation model calculated decay heat with the 1979 ANS Standard rather than the 1971 ANS Standard plus 20 percent, and calculated the two-phase critical flow with the RELAP5YA mechanistic model rather than the Moody critical flow model. (02044)

This is a Severity Level IV violation.(Supplement 1)

III. VIOLATION ASSOCIATED WITH SAFETY SYSTEM LOGIC TESTING (OI REPORT NO. 1-96-043)

Technical Specification 5.8.2 states, in part, that written procedures be established, implemented, and maintained to control, among other things, activities concerning testing of safety related equipment.

Item 12 of Attachment C to Procedure No. 0-16-3, "Work Order Process," defines a Functional Test Instruction (FTI) as instructions that define the evolutions or operations necessary to prove functionality or operability of a component, system, or structure.

Precaution 3.1 of Work Order 96-02928-00, Attachment A, "Functional Test for P-14A/S on A Train SIAS and Bus 5 Undervoltage," and Work Order 96-02929-00, Attachment A, "Functional Test for P-14 B/S on B Train SIAS and Bus 6 Undervoltage," states that if any step cannot be completed as specified in the FTI, then the Field Engineer must be contacted and any deviation from this FTI must be authorized in accordance with Procedure 0-16-3.

Deviations to FTIs are permitted through the use of Minor Technical Changes (MTC) as described in Item 13 of Attachment C to Procedure No. 0-16-3.

10 C.F.R. § 50.9(a) provides in part that information required by the Commission's regulations to be maintained by the licensee to be complete and accurate in all material respects.

10 C.F.R. Part 50, Appendix B, Criterion XVII, "Quality Assurance Records," requires, in part, that records of tests affecting quality be maintained.

Contrary to the above:

(1) On August 22, 1996, Step 5.3.3 of WO 96-02928-00 and WO 96-02929-00 could not be performed as written, and the licensee failed to resolve the discrepancy by making a Minor Technical Change. Specifically, Step 5.3.3 provided that at Main Control Board (MCB), Section C, open circuit continuity be verified at 86-RASA-2(YAF) using a volt-ohm meter (VOM) across the 5-5C contacts. The field test engineers could not verify the open contacts with a VOM because of resistance in the circuit caused by a bulb and resistor wired into the circuit. Instead of making a MTC to permit visual verification, the field engineers verified open circuit continuity visually and signed Step 5.3.3 as satisfactorily completed .

(2) On August 22, 1996, the licensee created test records that were materially inaccurate. Step 5.3.3 of WO 96-02928-00 and WO 96-02929-00 provided that at MCB, Section C, open circuit continuity be verified at 86-RASA-2(YAF) using a volt-ohm meter (VOM) across the 5-5C contacts. The field test engineers could not verify the open contacts with a VOM because of resistance in the circuit caused by a bulb and resistor wired into the circuit. Instead, the field test engineers verified open circuit continuity visually and signed Step 5.3.3 as satisfactorily completed. These inaccuracies were material because the tests concerned functionality or operability of safety-related components. (03014)

This is a Severity Level IV violation (Supplement 1)

Pursuant to the provisions of 10 CFR 2.201, Maine Yankee Atomic Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I , and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a

"Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at King of Prussia, Pennsylvania
this 8th day of October 1998

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

October 8, 1998

EA 97-387

Mr. T. C. McMeekin
President & Chief Executive Officer
Duke Engineering & Services, Inc.
400 South Tryon Street
P. O. Box 1004
Charlotte, NC 28201-1004

SUBJECT: RESPONSE OF DUKE ENGINEERING & SERVICES, INC. TO NRC DEMAND FOR INFORMATION (OI Report No. 1-95-050)

Dear Mr. McMeekin:

This letter is in response to your letter dated February 27, 1998, fowarding the response of Duke Engineering & Services, Inc. (DE&S) to an NRC Demand for Information (DFI) issued on December 19, 1997, to DE&S and Yankee Atomic Electric Company (YAEC). On December 19, 1997, the NRC staff also issued a related letter to Maine Yankee Atomic Power Company (MYAPCo) identifying apparent violations of NRC requirements by MYAPCo. MYAPCo responded to the apparent violations in writing on April 6, 1998, and in a predecisional enforcement conference on April 23, 1998. YAEC responded to the DFI by letter dated March 11, 1998. Although the DFI did not require a response from the two individuals identified therein as the YAEC Loss-of-Coolant Accident (LOCA) Group Manager and the Lead Engineer, they responded by letter dated March 12, 1998.

The DFI articulated NRC concerns regarding actions of the LOCA Group that may have caused: (1) the use of unacceptable evaluation models by MYAPCO to calculate emergency core cooling system (ECCS) performance for the Maine Yankee Atomic Power Station (MYAPS) because the evaluation models were not capable of calculating ECCS performance over the entire spectrum of postulated break sizes, (2) the maintenance and submission by MYAPCo of information that was not complete and accurate in all material respects, (3) the use by MYAPCo of an unacceptable evaluation model to calculate ECCS performance at MYAPS because of the incorrect application of the Alb-Chambre correlation, and (4) the use of an unacceptable best estimate evaluation model to calculate ECCS performance at MYAPS in an anlaysis of reduced steam generator pressure. The DFI required YAEC and DE&S to explain why they should be permitted to perfom LOCA analyses or any safety-related analyses to meet NRC requirements and why the NRC should not consider the unacceptable analyses to be the result of willfulness on the part of YAEC and/or DE&S personnel. The DFI was addressed to DE&S because shortly before issuance of the DFI, DE&S had acquired the YAEC Nuclear Services Division (NSD), which includes the LOCA Group.

The DE&S response to the DFI states that DE&S was aware of the concerns discussed in the DFI before DE&S acquired the YAEC LOCA Group. DE&S provided the results of reviews and assessments performed by or on behalf of DE&S as part of the acquisition. These assessments included (1) independent reviews of the findings, recommendations, and corrective actions from three teams previously formed by YAEC; (2) reviews of engineering and technical work processes and quality assurance programs at YAEC; (3) independent review of the technical issues related to the small-break LOCA (SBLOCA) analyses for MYAPS; (4) a legal assessment of potential willfulness of personnel actions related to SBLOCA analyses; and (5) review of a sampling of analyses performed for other NRC licensees.

In response to the DFI, DE&S also discussed improvements in the quality of YAEC procedures and work products; strengthened leadership in the Bolton, Massachusetts, office of DE&S (formerly YAEC NSD); and enhanced communication with nuclear clients, as providing the basis for concluding that safety-related analyses, products, and services provided by DE&S to NRC power reactor licensees will meet NRC requirements, including the provision of complete and accurate information.

DE&S further stated that although inadequate analyses may have been performed, DE&S found no willfulness on the part of DE&S (formerly YAEC) employees. In support of that conclusion, DE&S stated that before the DE&S acquisition, the YAEC LOCA Group had operated in isolation from the industry and from LOCA analysis experts on the NRC staff, and that although the analyses of the YAEC LOCA Group were not consistent with industry practice or NRC expectations, the conclusions reached by the YAEC LOCA Group seemed reasonable to them at the time. Therefore, DE&S concluded that the decisions made by the YAEC LOCA Group did not rise to careless disregard or deliberate violation of Commission requirements.

The NRC staff has completed its review of the responses of DE&S, YAEC, and the two individuals to the DFI. In light of the entire record, the staff concludes that the actions taken by the YAEC LOCA Group caused MYAPCo to be in violation of Commission requirements in a number of areas, but that these actions did not result from willfulness on the part of DE&S and/or YAEC employees. The violations are cited in a Notice of Violation issued concurrently on this day to MYAPCo, a copy of which is enclosed for your information. Letters to YAEC and to the two individuals regarding their response to the DFI are being issued concurrently on this day by the staff, and copies are enclosed for your information.

The staff further concludes that the corrective actions accomplished and planned, as discussed in the DE&S response to the DFI, provide a basis for reasonable assurance that in the future, the NRC and licensees can rely upon DE&S to provide complete and accurate information and that DE&S is willing and able to otherwise conduct its activities in accordance with the Commission's requirements. Therefore, the NRC staff has determined that no further enforcement action shall be taken against YAEC or DE&S regarding the actions of the LOCA Group at concern in the DFI. Any further review of DE&S activities in support of NRC licensees will be conducted through the staff's routine inspection and licensing review processes.

The staff, however, notes some ambiguity in the responses of the two individuals regarding the small-break LOCA (SBLOCA) analysis provided by YAEC to MYAPCo for operation of the Maine Yankee Atomic Power Station in Cycle 14. The LOCA Group Manager stated that it is his opinion that the analysis provided a "valid and conservative SBLOCA analysis", and the Lead Engineer stated that she believes that the analysis is "technically defensible." It might be inferred from these statements that the two individuals still believe the analysis that was provided to MYAPCo complies with NRC requirements or dispute the NRC staff's position as to the application of 10 C.F.R. §.50.46(a). The staff, however, relies upon the response of DE&S to the DFI to conclude that there is reasonable assurance that DE&S, including the NSD and the LOCA Group, will conduct its activities in accordance with NRC requirements. The staff expects that, regardless of their personal views as to the validity or technical defensibility of the SBLOCA analysis, the LOCA Group Manager and Lead Engineer in the future will conduct their activities in compliance with NRC requirements and consistent with DE&S policies and procedures fashioned to ensure compliance with NRC requirements. The staff has directed that if either individual has a different understanding, he or she shall notify the undersigned immediately.

In accordance with 10 C.F.R. § 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and any response (although none is required) will be placed in the NRC Public Document Room (PDR).

Sincerely,

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enc: Yankee Atomic Electric Company
Maine Yankee Atomic power Company
Mark R. Robeck, Baker & Botts

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

October 8, 1998

EA 97-387

Mark R. Robeck, Esq.
Baker & Botts
1600 San Jacinto Center
98 San Jacinto Blvd.
Austin, TX 78701-4039

SUBJECT: RESPONSE OF TWO INDIVIDUALS TO U.S. NUCLEAR REGULATORY COMMISSION (NRC) DEMAND FOR INFORMATION (OI Report No. 1-95-050)

Dear Mr. Robeck:

This letter is in response to your letter dated March 12, 1998, forwarding the response of two individuals to an NRC Demand for Information (DFI) issued on December 19, 1997, to Yankee Atomic Electric Company (YAEC) and Duke Engineering and Services, Inc. (DE&S). The DFI did not require a response from the two individuals identified in the DFI as the Loss-of-Coolant Accident (LOCA) Group Manager and the Lead Engineer. On December 19, 1997, the NRC staff also issued a related letter to Maine Yankee Atomic Power Company (MYAPCo) identifying apparent violations of NRC requirements. MYAPCo responded to the apparent violations in writing on April 6, 1998, and in a predecisional enforcement conference on April 23, 1998. DE&S responded to the DFI by letter dated February 27, 1998, and YAEC responded to the DFI by letter dated March 11, 1998.

The DFI articulated NRC concerns regarding actions taken by the LOCA Group that may have caused: (1) the use of unacceptable evaluation models by MYAPCO to calculate emergency core cooling system (ECCS) performance for the Maine Yankee Atomic Power Station (MYAPS) because the evaluation models were not capable of calculating ECCS performance over the entire spectrum of postulated break sizes, (2) the maintenance and submission by MYAPCo of information that was not complete and accurate in all material respects, (3) the use by MYAPCo of an unacceptable evaluation model to calculate ECCS performance at MYAPS because of the incorrect application of the Alb-Chambre correlation, and (4) the use of an unacceptable best estimate evaluation model to calculate ECCS performance at MYAPS in an analysis of reduced steam generator pressure. The DFI required YAEC and DE&S to explain why they should be permitted to perform LOCA analyses or any safety-related analyses to meet NRC requirements and why the NRC should not consider the unacceptable analyses to be the result of willfulness on the part of YAEC and/or DE&S personnel. The DFI was also addressed to DE&S because shortly before issuance of the DFI, DE&S had acquired the YAEC Nuclear Services Division (NSD), which includes the LOCA Group.

The NRC staff has completed its review of the responses of YAEC, DE&S, and the two individuals to the DFI. In light of the entire record, the staff concludes that the actions taken by the YAEC LOCA Group staff caused MYAPCo to be in violation of Commission requirements in a number of areas, but that these actions did not result from willfulness on the part of DE&S and/or YAEC employees. The violations are cited in a Notice of Violation issued concurrently on this day to MYAPCo, a copy of which is enclosed for your information. Letters to DE&S and to YAEC regarding their responses to the DFI are being issued concurrently on this day by the staff, and copies are enclosed for your information.

The response of DE&S discussed the circumstances surrounding the violations, and addressed actions taken by YAEC and DE&S, and DE&S's commitment, to prevent recurrence of the events that gave rise to the DFI. The responses of the two individuals state that they generally agree with the conclusions of the DE&S response, especially that they had not acted with careless disregard for NRC requirements. The staff, however, notes some ambiguity in the responses of the two individuals regarding the small-break LOCA (SBLOCA) analysis provided by YAEC to MYAPCo for operation of the MYAPS in Cycle 14. The LOCA Group Manager stated that it is his opinion that the analysis provided a "valid and conservative SBLOCA analysis", and the Lead Engineer stated that she believes that the analysis is "technically defensible." It might be inferred from these statements that the two individuals still believe the analysis complies with NRC requirements or dispute the NRC staff's position as to the application of 10 C.F.R. § 50.46(a). The staff, however, relies upon the response of DE&S to the DFI to conclude that there is reasonable assurance that DE&S, including the NSD and the LOCA Group, will conduct its activities in accordance with NRC requirements. The staff expects that, regardless of their personal views as to the validity or technical defensibility of the SBLOCA analysis, the LOCA Group Manager and Lead Engineer in the future will conduct their activities in compliance with NRC requirements and consistent with DE&S policies and procedures fashioned to ensure compliance with NRC requirements. If either individual has a different understanding, he or she shall notify the undersigned immediately.

In accordance with 10 C.F.R. § 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and any response (although none is required) will be placed in the NRC Public Document Room (PDR).

Sincerely,

Samuel J. Collins

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enc: Duke Engineering & Services, Inc.
Yankee Atomic Electric Company
Maine Yankee Atomic Power Company