50-245

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

February 20, 1996

Mr. Keith Perzanoski
8 Park Lane
Niantic, Connecticut 06357

Dear Mr. Perzanoski:

I am responding to your letter of October 17, 1995, to President Clinton. As you were informed by a letter from James A. Dorskind, Special Assistant to the President, dated November 6, 1995, your letter was forwarded to the U.S. Nuclear Regulatory Commission (NRC) for review and appropriate action. In your letter, you expressed concern that Northeast Utilities (NU) has been negligent in its public safety practices. Specifically, you expressed concern that NU broke its licensing agreement by removing the full core from the nuclear reactor instead of one-third of the core at refueling. You also raised questions about protection of employees who have expressed concern about NU's practices and the NRC's enforcement policy.

As you know, the NRC held a public meeting on November 8, 1995, in Waterford, Connecticut, regarding Millstone Unit 1. The meeting was arranged on short notice to address increased public interest about the safety of Northeast Nuclear Energy Company's (licensee's) fuel offloading activities at Unit 1. During the meeting, the NRC staff discussed key design features in offloading fuel, the status of NRC activities to date regarding NRC's review of the licensee's related amendment request dated July 28, 1995, and the NRC's inspection activities to verify independently the adequacy of the design and procedures to be used in offloading spent nuclear fuel. The NRC staff who participated in the meeting included those who were principally involved in the review and evaluation of the license amendment request.

On November 9, 1995, the NRC completed its review of the licensee's amendment request and issued an amendment to the facility operating license for Millstone Unit 1. The amendment added a license condition that allows the licensee to conduct refueling operations that include full-core offloading as a normal end-of-cycle event in accordance with the controls proposed in the licensee's application for amendment. The license condition further requires that the licensee update its design basis, described in the Updated Final Safety Analysis Report (UFSAR), to reflect the revised description authorized by the amendment in accordance with NRC's regulations. The staff evaluated the safety issues associated with full-core offloading and determined that the restrictions in the license condition ensure adequate protection of the public health and safety. I have enclosed a copy of the letter (Enclosure 1), which includes the NRC's safety evaluation of this matter. The licensee completed offloading of the Millstone Unit 1 core for refueling on November 19, 1995. The fuel offloading was closely tracked by NRC inspectors to ensure that the licensee complied with its license and procedures.

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As you may also be aware, on August 21, 1995, as supplemented on August 28, 1995, the NRC received a petition pertaining to the spent fuel offloading practices at Millstone Unit 1, which was submitted pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations by Mr. George Galatis and the Citizens Group, "We the People." The petition alleged, in part, that the licensee has knowingly, willingly, and flagrantly operated Millstone Unit 1 in violation of its operating license for approximately 20 years; that it obtained previous licensing amendments through the use of material false statements; and that it proposes to continue operating under unsafe conditions rather than comply with the mandates of its license. The NRC staff is reviewing the issues associated with the petition. The NRC's Office of Investigations is also looking into these matters for possible licensee wrongdoing. In addition, the NRC's Office of the Inspector General (OIG) has investigated the Millstone Unit 1 spent fuel pool issues. At a public meeting on December 5, 1995, in New London, Connecticut, the NRC Acting Inspector General stated that certain of the licensee's activities may have been conducted in violation of Millstone Unit 1 license requirements and that refueling activities may not have been conducted consistent with the Millstone Unit 1 UFSAR. The final OIG report was issued December 21, 1995. The NRC staff will inspect past NU practices and other spent fuel pool issues and will consider enforcement action, as appropriate, after the inspections and investigations are complete. The staff will also hold an informal public hearing on March 7, 1996, at the Waterford Townhall in Waterford, Connecticut to gain as much information on the 10 CFR 2.206 petition as possible. The staff considers an informal public hearing a means to serve not only as a source of potentially valuable information for NRC to evaluate a 2.206 petition, but also affords the petitioner substantive involvement in the review and decision making process through direct discussions with the NRC and the licensee. The Federal Register Notice announcing this meeting is enclosed (Enclosure 2) for your information. A final Director's Decision will not be issued until after the agency procedures for enforcement have been completed.

Regarding your concern about protection of employees who have expressed concern about the licensee's practices, 10 CFR 50.7 states, in part, that discrimination by a licensee against an employee for engaging in certain protected activities is prohibited. Therefore, if the NRC finds that a licensee has discriminated against one of its employees who is engaged in protected activities, the NRC will take appropriate enforcement action. For your information, with respect to Millstone, civil penalties have been issued where discrimination has been found. In addition, in a letter dated December 15, 1995, the NRC staff informed the licensee that the staff had formed a senior-level review group to conduct an independent review and evaluation on how Millstone-related employee concerns and allegations have been handled. A copy of this letter and the enclosure, which outlines the objectives and scope of the review and evaluation, are also enclosed (Enclosure 3) for your information. I want to assure you that the NRC regards the threat of discrimination to be of the utmost importance and will continue to investigate allegations of discrimination and take enforcement action, as appropriate.

Regarding your concern about NRC's identification of allegers or "whistleblowers," the NRC policy and procedures on referring allegations back to a licensee are sensitive to the allegers' identity. In an effort to manage our finite resources, allegations may be referred to licensees for action and response unless the information cannot be released in sufficient detail without compromising the identity of the alleger or confidential source (unless the alleger has no objection to his or her name being released). Before referring an allegation to a licensee, all reasonable efforts are to be made to inform the allegers or confidential sources of the planned referral. The referral is to be made in a manner aimed at protecting the identity of the alleger or confidential source, including rewriting the allegation. Notification of the referral is also to be provided to the alleger or confidential source. The NRC then reviews the adequacy of the licensee's response and independently verifies the licensee's findings, conclusions and corrective actions, as considered appropriate.

Regarding your concern about the NRC's enforcement policy, I assume you are referring to an NRC inspection report when you state that the NRC found the licensee to be in violation of its licensing agreement but did not fine the licensee or take enforcement action. Licensees are required to meet all appropriate codes and regulations, as well as, assure conformance with their license, technical specifications and numerous other commitments. NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," states that the NRC recognizes that there are violations of negligible or minor safety or environmental concern that are below the level of significance of Severity Level IV violations (the lowest level violation). These minor violations are not the subject of formal enforcement action and may or may not be described in inspection reports. To the extent such violations are described, they are noted as noncited violations. The purpose of exercising this discretion is not to minimize NRC findings or reduce the number of enforcement actions. The purpose of this discretion is to encourage licensees to identify and correct violations and, at the same time, reduce the NRC's efforts in addressing less significant issues that have been corrected, so that efforts can be focused on more safety significant issues. However, let me assure you that the NRC thoroughly reviews all potential violations before deciding on a course of action. Whether or not formal enforcement action is taken, the licensee is always required to take appropriate corrective action.

Further, the NRC monitors and assesses the operations at all three Millstone units through many ongoing activities, including, inspections by the five Millstone-based resident inspectors (soon to be increased temporarily to six onsite personnel), inspections by NRC Region I personnel, and evaluations by NRC Headquarters personnel of specific technical and operational issues. These NRC functions provide extensive oversight and assessment of the activities at the Millstone plants and are focused on verifying that these facilities are being operated safely. I also want to assure you that the NRC staff is maintaining close contact with Senators Dodd and Lieberman and Representative Gejdenson to keep them informed of regulatory activities at the Millstone site. If you have any other questions, please contact James Andersen, the NRC Project Manager for Millstone Unit 1, at (301) 415-1437.

Sincerely,
Original Signed By
WILLIAM T. RUSSELL
William T. Russell, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. License Amendment
2. Federal Register Notice
3. Independent Staff Review Letter

bcc: J. Dorskind
Special Assistant to the President
Director of Correspondence and
Presidential Messages

*SEE PREVIOUS CONCURRENCE
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Approved by Comm. 2/16/96.

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OFFICE	*OI	*IG	*OGC	D:PDI-3 *	DD:DRPE *
NAME	RFortuna (phone)	GMulley (phone)	CMarco	PMcKee	JZwolinski
DATE	12/28/95	12/28/95	12/29/95	01/22/96	01/05/96
OFFICE	D:DRPE *	ADPR:NRR	D:NRR	EDO	OCM
NAME	SVarga	RZimmerman	WRussell	JTaylor	02/16/96
DATE	01/16/96	02/1/96	02/2/96	02/ /96	~~01/##/96~~

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Sincerely,

William T. Russell, Director
Office of Nuclear Reactor Regulation

Enclosures: 1. License Amendment
2. Independent Staff Review Letter

bcc: J. Dorskind
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Further, the NRC monitors and assesses the operations at all three Millstone units through many ongoing activities, including, inspections by the five Millstone-based resident inspectors (soon to be increased to seven onsite personnel), inspections by NRC Region I personnel, and evaluations by NRC Headquarters personnel of specific technical and operational issues. These NRC functions provide extensive oversight and assessment of the activities at the Millstone plants and are focused on verifying that these facilities are being operated safely. I also want to assure you that the NRC staff is maintaining close contact with Senators Dodd and Lieberman and Representative Gejdenson to keep them informed of all actions regarding regulatory activities at the Millstone site. If you have any other questions, please contact James Andersen, the NRC Project Manager for Millstone Unit 1, at (301) 415-1437.

Sincerely,

William T. Russell, Director
Office of Nuclear Reactor Regulation

Enclosures: 1. License Amendment
2. Independent Staff Review Letter

bcc: J. Dorskind
Special Assistant to the President
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Sincerely,

William T. Russell, Director
Office of Nuclear Reactor Regulation

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3. Independent Staff Review Letter

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

November 9, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M93080)

Dear Mr. Opeka:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-21 for Millstone Nuclear Power Station, Unit 1, in response to your application dated July 28, 1995, as supplemented September 12, October 18, and October 31, 1995.

The amendment adds License Condition 2.C(6), "Spent Fuel Pool Operations," which allows Northeast Nuclear Energy Company (NNECO) to conduct refueling operations that include full-core offload as a normal end-of-cycle event in accordance with the controls proposed in Attachment 4 of the application for amendment dated July 28, 1995, as supplemented September 12, 1995. The license condition further requires that NNECO update the Updated Final Safety Analysis Report (UFSAR) to reflect the revised description authorized by this amendment in accordance with 10 CFR 50.71(e). The bases information proposed in Attachment 4 of NNECO's submittal dated July 28, 1995, as supplemented September 12, 1995, should be placed in the Millstone Unit 1 UFSAR. The amendment does not specifically approve the ORIGEN2 code, however, the staff has determined through independent calculations that the ORIGEN2 code is conservative for this specific application and the results acceptable. The staff did not credit evaporative cooling (ONEPOOL code) in its independent calculations and, therefore, has not made a determination on its acceptability.

The NRC staff added the new license condition instead of the requested technical specifications (TS) since provisions are already contained in TS 6.8.1 which require that written procedures be established, implemented, and maintained covering certain activities. These activities include the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, "Quality Assurance Program Requirements," (including shutdown cooling and spent fuel pool cooling) and refueling operations. In addition, the TS requested do not conform to those in the Standard Technical Specifications (NUREG-1433) and, as discussed in NNECO's October 18, 1995, letter, do not meet the criteria in 10 CFR 50.36 which describe what limiting conditions are required to be in TS. The staff notes that NNECO's controls are more conservative than NUREG-1433 and industry standards, and put tighter restrictions on refueling activities at Millstone Unit 1. Therefore, based on the above and the safety evaluation which is enclosed, the staff has approved the amendment request.

The NRC staff also notes, as described in the safety evaluation, that it currently has ongoing programs reviewing generic shutdown risk and spent fuel pool issues. When these activities are completed, the NRC will inform the industry of the results and any required actions.

The NRC staff further notes that NRC inspectors are currently reviewing other issues at Millstone Unit 1 related to the spent fuel pool, but not related to the license amendment, and will have verified the satisfactory completion of NNECO procedures and design modifications, which need to be completed prior to the movement of the first fuel assembly.

As noted above a copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Phillip F McKee for

James W. Andersen, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-245

Enclosures: 1. Amendment No. 89 to DPR-21
2. Safety Evaluation

cc w/encls: See next page

cc:

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Donald B. Miller, Jr.
Senior Vice President
Millstone Station
Northeast Nuclear Energy Company
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Waterford, CT 06385

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

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-245

MILLSTONE NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-21

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northeast Nuclear Energy Company (the licensee) dated July 28, 1995, as supplemented September 12, October 18, and October 31, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, Facility Operating License No. DPR-21 is hereby amended to authorize revision of the Updated Final Safety Analysis Report (UFSAR) description to allow a partial or full-core offload during refueling as a normal end-of-cycle event and paragraph 2.C(6) is added to read as follows:

   (6) Spent Fuel Pool Operations

   NNECO shall conduct refueling operations that include full-core offload as a normal end-of-cycle event in accordance with the controls proposed in Attachment 4 of the application for amendment dated July 28, 1995, as supplemented September 12, 1995.

   NNECO shall update the UFSAR to reflect the hardware modifications, heat load analysis, and operational controls consistent with the application for amendment dated July 28, 1995, as supplemented September 12, 1995, in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip F. McKee

Phillip F. McKee, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Pages 4 and 5 of License*

Date of Issuance: November 9, 1995

* Pages 4 and 5 are attached, for convenience, for the composite license to reflect this change.

(5) Integrated Implementation Schedule

a. Northeast Nuclear Energy Company shall implement and maintain in effect the Integrated Implementation Schedule Program Plan (the Program Plan) to be followed for scheduling of plant modifications and engineering studies. The Program Plan shall be followed from and after the effective date of this license condition.

b. This license condition shall be effective for three years from the date of issuance of Amendment No. 80. (Date of Issuance February 23, 1995).

(6) Spent Fuel Pool Operations

NNECO shall conduct refueling operations that include full-core offload as a normal end-of-cycle event in accordance with the controls proposed in Attachment 4 of the application for amendment dated July 28, 1995, as supplemented September 12, 1995.

NNECO shall update the UFSAR to reflect the hardware modifications, heat load analysis, and operational controls consistent with the application for amendment dated July 28, 1995, as supplemented September 12, 1995, in accordance with 10 CFR 50.71(e).

D. The facility has been granted certain exemptions from the requirements of 10 CFR Part 50 as set forth below:

(1) Section III.G. of Appendix R to 10 CFR Part 50 "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. The staff safety evaluation, dated November 6, 1985, concluded that the licensee's existing fire-protection configuration with proposed modifications achieves an equivalent level of safety. Exemption granted November 6, 1985.

(2) Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J relates to containment leakage test requirements, specifically periodic verification by tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate containment. Three exemptions were granted on May 10, 1985, on the basis of the staff safety evaluation. The two which remain in effect relate to testing of expansion bellows at containment penetrations and main steam isolation valves.

(3) Section 50.71 of 10 CFR Part 50, "Maintenance of Records, Making of Reports." Section 50.71(e)(3) relates to the requirement for submittal of an updated FSAR. An exemption for a schedular delay in the submittal of the updated FSAR was granted on November 22, 1985. This exemption requires that the FSAR update be completed and submitted by March 31, 1987.

3. This license is effective as of its date of issuance and shall expire at midnight, October 6, 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Frank J. Miraglia, Director
Division of PWR Licensing - B

Attachment:
Appendix A - Technical Specifications

Date of Issuance: October 31, 1986

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 89

TO FACILITY OPERATING LICENSE NO. DPR-21

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-245

1.0 INTRODUCTION

By letter dated July 28, 1995, as supplemented September 12, October 18, and October 31, 1995, the Northeast Nuclear Energy Company (NNECO) submitted a request for changes to the Millstone Nuclear Power Station, Unit 1 Technical Specifications (TS). NNECO's request would add TS to Section 3.10, Refueling and Spent Fuel Handling. Specifically, the proposed TS (with applicability, action, and surveillance requirements) would require that (1) the reactor be subcritical for at least 100 hours prior to the start of reactor refueling operations, (2) the spent fuel pool (SFP) bulk temperature be maintained less than or equal to 140°F, and (3) two trains of shutdown cooling (SDC) be operable during reactor refueling operations. Under the proposed amendment, NNECO could also perform a full-core offload as a normal end-of-cycle event and use the ORIGEN2 and ONEPOOL codes. To support the changes, NNECO has modified the SDC system by adding a cross-connect line to the spent fuel pool cooling (SFPC) system, which provides the SDC system with the necessary redundancy to credit it for the base case decay heat load analysis. The September 12, October 18, and October 31, 1995, letters provided clarifying information that did not change the staff's initial proposed no significant hazards consideration determination.

On August 21, 1995, as supplemented August 28, 1995, a 10 CFR 2.206 Petition was received by the NRC. The Petition alleged, in part, that NNECO has knowingly, willingly, and flagrantly operated Millstone Unit 1 in violation of its operating license for approximately 20 years; that it obtained previous licensing amendments through the use of material false statements; and that it presently proposes to continue operating under unsafe conditions rather than comply with the mandates of its license. The Petition also raised concerns regarding the proposed license amendment. In a letter to the Petitioners dated October 26, 1995, the NRC acknowledged receipt of the Petition and stated that it will take action with regard to the issues raised within a reasonable time. The NRC also stated the issues associated with any pending license amendment are not within the scope of 10 CFR 2.206 and thus are not appropriate for consideration under 10 CFR 2.206. The NRC further stated that the issues raised in the Petition would be considered by the NRC staff in acting upon the license amendment request. The staff has considered the concerns expressed by the Petitioners and has concluded that there is no information in the Petition that would affect the staff's ability to make a final no significant hazards consideration determination.

2.0 BACKGROUND

The purpose of the SFPC system is to remove decay heat from the spent fuel stored in the pool and to maintain the purity of the water in the pool. The SFPC system consists of two trains, each containing one heat exchanger and one pump, which provides 625 gpm flow when aligned to one heat exchanger. The SFPC pumps take their suction from the SFP skimmer surge tanks and discharge to either the SFP or the reactor well (RXW), with the discharge to the SFP as the normal alignment. Cooling water to the heat exchangers is provided by the reactor building closed cooling water (RBCCW) system. The heat exchangers are each rated to remove 3.4 MBTU/hr at rated flow with a SFP temperature of 125°F and an RBCCW temperature of 85°F. A cross-connect line is provided between the pumps and heat exchangers so that each pump can provide flow to either (or both) heat exchangers. A cleanup loop, consisting of a filter, demineralizer, and a strainer, is provided to remove any impurities and activity from the SFP water. The cleanup loop is sized to pass 625 gpm and is manually isolated when the SFP temperature reaches 125°F to protect the demineralizer resin. The condensate system provides the seismic Category I, safety-related source of makeup for the SFP. Alternate sources of makeup are also available to the operators, including makeup from the fire protection system.

The primary function of the SDC system is to cool the reactor from 280°F to 125°F within 24 hours in a controlled manner and to remove decay heat to maintain reactor temperature below 125°F. The SDC system has two parallel loops, each containing one centrifugal 2900 gpm pump, and one design-rated 22 MBTU/hr heat exchanger. Both SDC trains have a common suction line (normal lineup) from the reactor vessel and return flow to the vessel through the low pressure coolant injection (LPCI) system. After the reactor cooldown is complete, the vessel head, dryer, and separator are removed. The RXW is flooded up to the level of the SFP and the SFP weir gate is removed.

In support of higher heat loads associated with refueling activities, the system design has allowed a portion of the flow from SDC train "A" to be diverted through cross-tie piping to aid the SFPC system in removing decay heat from the SFP. When SDC is cross-tied with the SFPC system, the SFP weir gate in the SFP-RXW connecting channel must be open to prevent overflowing the SFP or draining the RXW. With the modifications to the SDC system complete, both SDC trains have the capability of being cross-connected with the SFPC system. However, in the letter dated October 18, 1995, NNECO noted that the cross-connect isolation valves will be administratively controlled to prevent both valves from being open at the same time.

2.1 SFPC Licensing Basis

In the early 1980s, the NRC staff initiated the Systematic Evaluation Program (SEP) review and a Full-Term Operating License (FTOL) review of Millstone Unit 1 to ensure that the design of selected systems provided an acceptable level of safety. Fuel storage was reviewed under SEP Topic IX-1, and the findings of that review were documented in Section 9.1 of the FTOL safety evaluation report (SER). Although the staff found the spent fuel storage facility at Millstone Unit 1 acceptable with respect to Section 9.1.2, "Spent Fuel Storage," and Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," of the Standard Review Plan (SRP), the evaluations that document staff acceptance were based on discrete criteria, set forth below, rather than the entire set of acceptance criteria included in the respective sections of the SRP.

With regard to the heat removal capability of the SFPC system, the staff stated in the FTOL SER for Millstone Unit 1 that the conclusions documented in the safety evaluation for Amendment 39 to Provisional Operating License DPR-21 (issued June 30, 1977), remain valid. This safety evaluation found that the heat removal capability of the SFPC system was adequate to maintain pool outlet temperature below 125°F for the normal refueling offload of a quarter core. The staff also stated that use of the SDC system is acceptable for fuel pool cooling, including operation in conjunction with the SFPC system for a full-core offload, because the SDC system does not perform a safety function that could not be performed by other redundant reactor systems.

The staff's safety evaluation for Amendment 40 to Facility Operating License DPR-21 (issued November 27, 1989), approved an increase in the licensed storage capacity of the Millstone Unit 1 SFP. This safety evaluation found that the heat removal capacity of the SFPC system was adequate to maintain SFP bulk temperature below 150°F for a defined heat load selected to bound the normal discharge of one third of a core, assuming a single, long-term failure of a SFPC pump. The staff also found that the heat removal capability of the SDC system combined with the SFPC system was adequate to maintain SFP bulk temperature below the acceptance criterion of 212°F for the abnormal heat load, which was defined as a full-core offload 3 months after the last normal discharge with 250 hours decay prior to fuel movement.

2.2 NNECO's Analysis

In order to ensure a bounding thermal-hydraulic analysis in support of the requested license amendment, NNECO assumed a SFP capacity of 4412 fuel assemblies (the SFP maximum capacity), which is more conservative than the 3229 fuel assemblies authorized in Millstone Unit 1 TS. To determine the total decay heat load, an initial inventory of 3832 fuel assemblies accumulated through scheduled discharges from September 1972 to October 2009 was assumed to be present in the SFP providing a constant decay heat load from previously discharged fuel.

NNECO presented transient calculations to evaluate the bulk pool temperature. Under the analyzed base case end-of-cycle refueling scenario, NNECO's analysis assumes that the full-core of 580 assemblies are discharged to the SFP producing a total decay heat load of 22.54 MBTU/hr, which includes the 3832 fuel assemblies from previous refueling discharges. For conservatism, NNECO assumed a failure of one SDC pump and one SFPC pump resulting from an electrical distribution system failure when calculating the bulk SFP temperature for this scenario.

Additionally, NNECO considered a maximum decay heat load scenario where the additional decay heat from one refueling offload with maximum burnup is superimposed on the assumptions for the base case end-of-cycle refueling scenario for a total heat load of 26.14 MBTU/hr. A single failure was not considered by NNECO for this maximum decay head load scenario.

The fuel assemblies in both scenarios were assumed to be transferred at a rate of 10 assemblies per hour following 100 hours of decay in the reactor vessel. Heat removal through the SFPC system and the SDC heat exchangers were credited for both scenarios. NNECO calculated the heat removal rate though the SFPC and SDC heat exchangers based on a temperature effectiveness factor obtained by rating each heat exchanger on a proprietary thermal-hydraulic computer code (STER). NNECO also included comparative heat load calculations that credited the effects of evaporative and convective cooling based on the proprietary ONEPOOL code. In all scenarios, the heat exchangers were assumed to be fouled to their design maximum extent for the calculation of the temperature effectiveness factor. Transient calculations based on other discharge scenarios were also included in the license amendment request, but were not considered bounding.

2.3 Results of NNECO's Analysis

For the base case end-of-cycle refueling scenario, assuming a single active failure, the analysis indicated a maximum bulk pool temperature of 138.7°F would be attained 165 hours after reactor shutdown. This temperature is below the design temperature of 140°F specified in the proposed TS for the SFP under base case conditions.

The most limiting design basis scenario with regard to bulk pool temperature was found to be the maximum decay heat load scenario. This scenario was defined as a full-core offload 36 days after the last fuel discharge (refueling outage). The maximum SFP bulk temperature was calculated to be 144.4°F, 164 hours after reactor shutdown (without crediting the effects of evaporative and convective cooling). This temperature is below the previously accepted licensing basis criterion of no bulk boiling for the maximum decay heat load scenario.

NNECO's report also evaluated the transient response of the SFP following a loss of all forced cooling. The calculated minimum time from the loss of pool cooling to the onset of boiling was determined to be approximately 5 hours, with the maximum rate of inventory loss due to boil-off calculated to be

57 gpm. The following conservative assumptions were inherent in the calculations: (1) forced cooling was lost at the instant of maximum temperature using the maximum design heat load in the SFP, (2) the system response was evaluated assuming no makeup water addition to the SFP, and (3) the time-to-boil calculation was based on an isolated SFP, which is inconsistent with the operating requirements of the SDC system prior to the loss of all forced cooling (i.e., SDC system providing cooling to SFPC system with the reactor well flooded and the SFP gate open). The staff did not credit the effects of evaporative cooling in their review of NNECO's analysis.

3.0 EVALUATION

3.1 SDC Cross-Connect to SFPC (Mechanical)

NNECO designed a piping system modification for Millstone Unit 1 that provided a redundant supply from the "B" SDC train to the SFPC system. The modification included a 6" line with a manual isolation gate valve from the "B" SDC heat exchanger discharge line to the existing cross-tie with the SFPC system.

The design conditions for the SDC system are 1250 psig at 350°F. The design conditions for the SFPC system are 200 psig at 150°F. The existing cross-tie from the "A" SDC train to the SFPC system is used during the refueling mode, and is designed to operate at a maximum condition of 150 psig and 140°F. The system modification piping (6" schedule 80), fittings and gate valve exposed to the SDC system are designed for the design conditions of that system. The gate valve was qualified for service in the SDC system. The system modification piping (6" schedule 40), and fittings exposed to the SFPC system are consistent with that system's design conditions. The modification utilized piping design rules and criteria contained in the Millstone Unit 1 Updated Final Safety Analysis Report (UFSAR), and the design and fabrication of the valve meets the requirements of the original plant valve specification requirements.

The modifica n construction and installation is consistent with the design, materials, conditions and components used in the existing systems, and ASME Code Section XI requirements. In addition, NRC inspectors have conducted walk-downs of the SDC cross-tie modification. This included visual observations of the SDC heat exchangers, pump rooms, and the actual cross-tie piping. The inspectors further verified all physical modification work is complete and the system testing was acceptable.

3.2 SDC Cross-Connect to SFPC (Operational)

Because the SDC system interfaces with both the reactor coolant system and the SFPC system, there is a potential for an interfacing-system loss-of-coolant accident (ISLOCA). The SDC system is designed for full reactor coolant system operating pressure and is separated from the reactor coolant system at the suction and return interfaces by redundant, motor-operated containment isolation valves. The SFPC system is designed for a pressure of 200 psig and is separated from the SDC system at the suction and return interfaces by manually-operated valves. These manual valves are designed to withstand full reactor coolant system operating pressure on the SDC system side and their position is administratively controlled.

In the letter dated October 31, 1995, NNECO described the design features, operating procedures, and administrative controls provided to prevent and mitigate ISLOCA via the SDC system and SFPC system piping at Millstone Unit 1. The SDC system containment isolation valves (SDC-1, 2A/B, 4A/B and 5) receive a signal to automatically close on low reactor water level (see Figure 4.1 in NNECO's submittal dated July 26, 1995 for a diagram of the systems). The group III isolation bypass switch, which will bypass the isolation signal, is key-locked and under administrative controls. The isolation valve logic includes a temperature permissive requiring that the reactor coolant temperature be less than 350°F for valve opening. The valves also will close if the reactor coolant temperature exceeds 350°F. There is no bypass switch to bypass the high temperature interlock. Even assuming an isolation failure at the reactor coolant pressure boundary, there are no adverse consequences without a misalignment of the SDC/SFPC system interface valves (SD-34A/B and SD-35), because the SDC system and the interface valves are designed for full reactor coolant system operating pressure. Inspectors verified that procedure OP-305A for operation of the SDC/SFPC cross-connect, which NNECO approved on November 8, 1995, specifies that valves SD-34A/B and SD-35 are normally locked shut, and valves FP-34 and FP-35 are normally locked open. Steps 4.4.1 and 4.5.1 of OP-305A require verification that the gates between the SFP and the reactor cavity are removed for alignment of SDC train A and B, respectively. In this configuration, no coolant inventory loss through the cross-connect can occur.

Certain valve alignments with the SDC in operation have the potential to cause a transfer of coolant inventory from the reactor vessel to the SFP. Based on operating procedures and administrative controls the staff considers the misalignment of manual valves necessary to initiate an inadvertent transfer of coolant inventory from the reactor vessel to the SFP to be unlikely. However, should such a valve misalignment occur, several indications and alarms monitoring reactor water level and the feedwater system would alert the operators to the event. Transfer of coolant to the SFP could be detected by high skimmer surge tank level alarm followed by a high SFP level alarm. Although reactor vessel level decrease during such an event could be temporarily masked by an operating feedwater system or other injection system, if reactor water level continues to decrease, a reactor water low level alarm will be annunciated in the control room. Alarm response procedures will direct the operators to take proper actions to restore the reactor water level. A group III containment isolation signal would be initiated at a lower reactor water level. Millstone Unit 1 off-normal procedures require the operators to verify that the isolation has occurred. The group III isolation would isolate the SDC isolation valves and terminate the decrease in reactor water level.

The staff considered the potential consequences of water released to the reactor building floors due to SFP overflow or valve misalignment. Water leakage from the SFP or misaligned valves in the SFPC system piping could be detected by reactor building sump alarms. Reactor building corner rooms, which contain LPCI and core spray (CS) system equipment, would remain operable for leakage of up to 15,000 gallons to each room. The expected leakage due to the postulated coolant transfer is less than that value. Due to the physical locations of the systems, no more than one corner room would be significantly affected by the leakage. This is due in part to the physical separation of the corner rooms and the existence of dikes or curbs at various locations which would limit the spread of water from the leak. Since the LPCI and CS systems are separated and located at different corner rooms, (LPCI pumps A,C and CS pump A are in one corner room and LPCI pumps B,D and CS pump B are in another corner room) at least one set of LPCI pumps and one CS pump would remain available to achieve and/or maintain a safe shutdown condition.

The staff concludes that the Millstone Unit 1 design features, operator actions, and the administrative controls described above provide reasonable assurance of protection against ISLOCA events in the SDC and SFPC systems. In addition, NRC inspectors have reviewed the Millstone Unit 1 Plant Operations Review Committee approved procedures governing the control of the SDC to SFPC cross-connect isolation valves. The inspectors found the procedures adequate.

3.3 SFP Thermal-Hydraulics

The staff performed calculations to independently verify the SFP heat load using the methodology described in Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling." For additional conservatism, the staff also performed the heat load calculations assuming the SFP was filled to its maximum capacity (3832 fuel assemblies) prior to the beginning of the transient and compared their results with NNECO's calculations that did not credit evaporative cooling. Based on the results of the staff's calculations, the staff determined that the maximum decay heat loads calculated in the NNECO analyses were conservative. The staff also reviewed the impact of changes made by NNECO to the assumptions in the Millstone Unit 1 UFSAR's "Core Off Load Scenarios," regarding the operating conditions for these scenarios (e.g., hold time of 100 hours versus 250 hours, operating time prior to offload of 36 days versus 3 months - for the maximum decay heat load scenario), and found them adequately represented in the decay heat load calculations.

The proposed limiting condition for operation for TS 3.10.H specifies that two trains of SDC must be operable during normal reactor refueling operations. The use of SDC for cooling the SFP when a full-core is discharged is consistent with previous staff positions (i.e., Amendment No. 40). The assumption of a failure of both a SDC pump and a SFPC pump is more conservative than previously accepted cooling system failure assumptions. The remaining operable SDC pump is capable of maintaining SFP temperature below 140°F under limiting design conditions for the base case end-of-cycle heat load. In addition, NRC inspectors reviewed the procedure for operating the

SDC/SFPC cross-correct valves. The procedure requires that while SDC is required by the figure in Attachment 4 to NNECO's July 28, 1995, license amendment request the gate between the SFP and RXW be removed.

The proposed limiting condition for operation for TS 3.10.H also specifies acceptable power source configurations when SDC system operability is required. The acceptable configurations include, as a minimum, either two sources of offsite power and one emergency onsite power source, or one source of offsite power and two emergency onsite power sources. Although the first minimum power source configuration would not support maintenance of SFPC following a failure of an onsite power source coincident with a total loss of offsite power, this configuration is consistent with previously approved electrical distribution system configurations during refueling and the improved Standard Technical Specifications (NUREG-1433) for residual heat removal with a flooded reactor cavity (which has a coolant inventory equivalent to the coolant inventory in the SFP). During fuel assembly transfer operations, the reactor cavity must be flooded. These electrical power source configurations are acceptable based on the large thermal capacity of the SFP coolant inventory, which provides an extended period of time for recovery of cooling for this particular postulated event.

The staff found that the seismic Category I condensate system and the back-up fire protection system provide a reliable makeup capability to the SFP should a complete loss of forced cooling occur. Either system provides adequate capacity to make-up for coolant inventory losses caused by pool boiling for the maximum decay heat load case. The bounding time to reach boiling for these cases provides adequate time to align a source of coolant make-up.

Based on the above factors, the staff concluded that NNECO's analyses are more conservative than previously accepted analyses. The capacity of available systems, including postulated single failures, provides assurance that adequate decay heat removal will be maintained. Therefore, the proposed changes to the evaluated end-of-cycle full-core discharge are acceptable with regard to providing adequate cooling and makeup to the SFP.

3.4 Technical Specifications

NNECO proposed to establish 100 hours as the time the reactor must be subcritical prior to movement of fuel, establish a normal maximum SFP bulk temperature of 140°F, and require two trains of SDC be operable during normal reactor refueling operations (proposed TS 3/4.10.F, 3/4.10.G, and 3/4.10.H, respectively). The proposed changes are consistent with the thermal-hydraulic analysis of the SFP performed by NNECO, more restrictive and conservative than the existing TS, and exceed the requirements for improved Standard Technical Specifications (NUREG-1433). Therefore, the staff finds NNECO's proposed changes technically adequate. However, TS currently exist (TS 6.8.1) which require that written procedures be established, implemented, and maintained covering certain activities. These activities include the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, "Quality Assurance Program Requirements," (including shutdown cooling and spent fuel pool cooling) and refueling operations. In addition, the TS requested do not conform to those in NUREG-1433 and, as discussed in NNECO's October 18, 1995, letter, do not meet the criteria in 10 CFR 50.36 which describe what limiting conditions are required to be in the TS. Therefore, in this instance, the staff has determined to incorporate the requirements listed above in License Condition 2.C(6) of Facility Operating License DPR-21. The staff notes that the addition of this license condition does not incorporate SFPC or SDC procedures into the license. These procedures may be changed pursuant to 10 CFR 50.59.

3.5 Full-Core Offload

The staff has reviewed NNECO's submittal with regard to thermal-hydraulic concerns. NNECO's analysis, as described above, demonstrated the adequacy of the SFP cooling and makeup water systems (including the SDC system) to support the heat load generated by offloading a full-core to the SFP as a normal end-of-cycle event. The staff found NNECO's analysis to be acceptable in addressing potential SFP thermal-hydraulic concerns. Therefore, the NRC staff concludes that the offloading of a full-core to the SFP as a normal end-of-cycle event, with the controls requested by NNECO, is acceptable.

3.6 ORIGEN2 and ONEPOOL Codes

As discussed in Section 3.3 above, the staff used the guidance in BTP ASB 9-2, to independently verify the SFP heat load. Based on the results of its calculations, the staff determined that the maximum decay heat loads calculated in NNECO's analyses were comparable to the staff's calculation and, therefore, were acceptable. The staff notes that this safety evaluation does not specifically approve the ORIGEN2 code, however, the staff has determined through independent calculations that the ORIGEN2 code is conservative for this specific application and the results acceptable. The staff did not credit evaporative cooling (ONEPOOL code) in its independent calculations and, therefore, has not made a determination on its acceptability.

4.0 Generic SFP issues

The NRC staff has initiated an action plan that addresses SFP issues identified through a 1994 special inspection at Dresden 1, the staff's review of loss of SFPC concerns at Susquehanna Steam Electric Station, and other SFP concerns identified as part of this plan. Specific review areas identified through implementation of this action plan include plant design features and administrative controls that affect the probability of the following events: SFP boiling, adverse environmental effects on essential equipment due to boiling, significant loss of SFP coolant inventory, adverse radiological conditions, unplanned SFP reactivity changes, undetected SFP events, and adverse control system actuations. Based on findings from these review areas and their risk significance, the NRC intends to develop criteria for specific SFP operations for potential use in formulating generic communications, revisions of regulatory guidance, and other appropriate regulatory actions.

The staff has initiated this action plan to identify potential, cost-beneficial safety enhancements to the design and operation of the SFP and related systems. The action plan is not intended to address compliance concerns because the staff addresses such concerns directly through its inspection and enforcement activities. The staff has concluded that existing design features of SFPs and related systems, that are consistent with NRC guidance and requirements, provide an adequate level of protection for spent fuel in the SFP, and essential reactor safety systems to prevent an adverse radiological condition from developing (for spent fuel in reactor vessel). In addition, existing design features provide protection for people, both on and offsite, from adverse radiological conditions in the event that these conditions develop. These design features include (1) an acceptably reliable means of cooling, (2) anti-syphon protection on piping within the SFP, (3) multiple sources of SFP make-up water, (4) instrumentation with control room annunciation, and (5) SFP water purification systems. Therefore, additional regulatory actions developed through this action plan to enhance safety can be addressed separately from this amendment request.

The NRC staff has performed an assessment of the safety issues included in the scope of the staff's action plan that are germane to SFPC at Millstone Unit 1. This assessment included an evaluation of the licensing basis and design features for Millstone Unit 1 to better understand the safety significance of the Millstone Unit 1 situation. For Millstone Unit 1, the staff concluded that the probability of reaching boiling conditions in the SFP would be low. Additionally, the safety significance of such conditions, which are most likely during periods when the full-core is transferred to the spent fuel pool, would also be low based on certain design features.

At Millstone Unit 1, the systems that have a SFP cooling capability (SFPC and SDC) are designed to receive power from two separate and independent emergency buses that can receive power from either of two onsite power supplies following a loss-of-normal power. The independence of the systems reduces the probability of an event capable of causing a sustained loss of the cooling function for the SFP. Additionally, the SDC system at Millstone Unit 1 is readily available for SFP cooling because it does not have a design function other than providing core cooling for routine shutdowns and supplementing SFPC.

Assuming that the cooling function for the SFP is lost for an extended period despite these features, the safety systems subject to adverse environmental conditions from SFP boiling would not have a necessary function when all irradiated fuel has been transferred to the SFP and the time to reach boiling conditions is shortest. When the reactor cavity is flooded and the SFP gates are removed during the transfer of irradiated fuel, the time to recover cooling prior to experiencing adverse effects is longer yet and still more systems are available for cooling the irradiated fuel in the SFP and the reactor vessel. When the reactor cavity is not flooded to a level equivalent to the level of the SFP, the decay heat load in the isolated SFP is relatively small in comparison with the design cases evaluated for the proposed amendment and the time available to recover cooling before adverse conditions develop is very long. Therefore, the concerns with respect to a loss of SFPC and the practice of conducting a full-core offload at Millstone Unit 1 are not safety significant, and the design of the SFP and related systems at Millstone Unit 1 are safe.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 REQUEST FOR HEARING

A request for a hearing was received from We the People, the Seacoast Anti-Pollution League, the New England Coalition on Nuclear Pollution, and Donald Del Core of Uncasville, Connecticut.

7.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has made a final determination that the proposed amendment involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92(c), this means that the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission has evaluated the proposed changes against the above standards as required by 10 CFR 50.92(a) and has concluded that the proposed changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed license amendment specifies operating conditions and system configurations during the movement of spent fuel into the SFP. Specifically, the proposed amendment requires that (1) the reactor be subcritical for at least 100 hours prior to the start of reactor refueling operations, (2) the spent fuel pool bulk temperature be maintained less than or equal to 140°F, and (3) two trains of SDC be operable during reactor refueling operations. The new controls are more restrictive than the previous requirements and, as such, would not significantly increase the probability or consequences of an accident previously evaluated.

   In accordance with the proposed license amendment, NNECO may perform a full-core offload to the SFP as a normal end-of-cycle event. With a full-core offload, two moves per assembly are required, one to remove the assembly from the core and one to replace the assembly in the core.

The number of moves necessary for a third core offload can vary due to shuffling which would need to take place within the reactor vessel. However, NNECO stated for the upcoming refueling outage the difference is approximately 20 percent. While more fuel assemblies will be transferred, the refueling procedures and equipment used to move the assemblies have not been significantly modified. In addition, NNECO now requires that the reactor be subcritical for at least 100 hours prior to the start of reactor refueling operations. This time constraint is longer than the 24 hours assumed in the analyzed fuel handling accident. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed license amendment credits a second train of SDC to assist the SFPC system in removing decay heat from the SFP. The use of the SDC system for the cooling of a full-core discharge is consistent with previous staff safety evaluations, and the assumption of a failure of both a SDC pump and a SFPC pump is more conservative than previously accepted cooling system failure assumptions. In addition, Millstone Unit 1 design features, operator actions, and the administrative controls governing the SDC/SFPC interfacing valves provide reasonable assurance of protection against ISLOCA events. Therefore, the crediting of a second train of SDC to assist the SFPC system in removing decay heat from the SFP does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed license amendment requires that at least either two sources of offsite power and one emergency power source, or one source of offsite power and two emergency power sources be operable. Although the first minimum power source configuration would not support maintenance of SFPC following a failure of an onsite power source coincident with a total loss of offsite power, this configuration is consistent with previously approved electrical distribution system configurations during refueling and NUREG-1433 for residual heat removal with a flooded reactor cavity. During fuel assembly transfer operations, the reactor cavity must be flooded. These electrical power source configurations are acceptable based on the large thermal capacity of the SFP coolant inventory, which provides an extended period of time for recovery of cooling for this particular postulated event. The staff has concluded that adequate electrical reliability for cooling the SFP exists and, therefore, not having all electrical power sources available would not significantly increase the probability or consequences of an accident previously evaluated.

Based on the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated.

As stated above, the proposed license amendment puts controls on the movement of spent fuel into the SFP. Therefore, the controls added do not create the possibility of a new or different kind of accident from any previously evaluated.

Since a fuel handling accident was previously reviewed and approved, and the refueling equipment and fuel assembly offload procedures have not been significantly modified, a full-core offload does not create the possibility of a new or different kind of accident from any previously evaluated.

Prior to the cross-connect modification for SDC train "B" and the proposed license amendment, SDC train "A" was cross-tied to the SFPC system and found acceptable. The design basis for the SDC "B" cross-connect is the same as the original licensing design basis for the SDC "A" cross-connect. Therefore, the addition of a cross-connect between SDC train "B" and the SFPC system does not create the possibility of a new or different kind of accident from any previously evaluated.

Thus, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in the margin of safety.

The proposed license amendment puts controls on the movement of spent fuel into the SFP. These controls help ensure that the margin of safety will not be reduced during refueling operations. Therefore, the controls added do not involve a significant reduction in the margin of safety.

The proposed license amendment credits a second train of SDC to assist the SFPC system in removing decay heat from the SFP. NNECO analysis demonstrates that this cooling configuration will maintain the SFP bulk temperature below the pool design limit of 140°F with a postulated single failure while providing adequate cooling for irradiated fuel that may be stored in the reactor vessel. Therefore, the full-core offload and the crediting of SDC do not involve a significant reduction in the margin of safety.

Thus, the proposed changes do not involve a significant reduction in the margin of safety.

8.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration determination with respect to the amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

9.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: November 9, 1995

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHEAST UTILITIES

NOTICE OF INFORMAL 10 CFR 2.206 PUBLIC HEARING

The U.S. Nuclear Regulatory Commission (NRC) will hold an informal public hearing regarding a petition submitted pursuant to 10 CFR 2.206 involving Millstone Units 1, 2, and 3 and Seabrook Unit 1 of Northeast Utilities (NU/the licensee). The hearing will be held on March 7, 1996, at the Waterford Townhall in Waterford, Connecticut. The hearing will be open to public attendance and will be transcribed. The NRC has elected to hold such a hearing because of the complexity of the issues involved and the public's interest.

The structure of the hearing shall be as follows:

Thursday, March 7, 1996:	6:00 p.m.	- NRC opening remarks
	6:15 p.m.	- Petitioner's presentation
	7:00 p.m.	- NRC questions
	7:15 p.m.	- Licensee's presentation
	8:00 p.m.	- NRC questions
	8:15 p.m.	- Public comments
	9:45 p.m.	- Licensee/Petitioners' final statements
	10:00 p.m.	- Meeting concludes

A Petition pursuant to 10 CFR 2.206 was submitted to the NRC by Mr. Ernest C. Hadley on behalf of Mr. George Galatis and the Citizens Group, We the People (Petitioners) on August 21, 1995, as supplemented August 28, 1995. The Petitioners allege that NU has knowingly, willingly, and flagrantly operated Millstone Unit 1 in violation of its operating license for approximately 20 years; that it obtained previous licensing amendments through the use of material false statements; and that it presently proposes to continue operating under unsafe conditions rather than comply with the mandates of its license. Specifically, the Petitioners allege that NU has offloaded more fuel assemblies into the spent fuel pool than permitted under License Amendment No. 39 to the Millstone Unit 1 Provisional Operating License and License Amendment No. 40 to the Millstone Unit 1 Full-Term Operating License. The Petitioners further allege that License Amendments Nos. 39 and 40 were based upon material false statements made by NU in documents submitted to the NRC. The Petitioners refer to certain NU submittals allegedly containing the false information, such as NU Safety Assessment Reports associated with License Amendments Nos. 39 and 40 and with Systematic Evaluation Program Topics IX-1 (fuel storage), IX-5 (ventilation systems), and III-7.B (design codes, design criteria, load combinations, and reactor cavity design criteria).

In the Supplement, Mr. Galatis alleges that NU also committed violations by offloading more than one-third of a core of fuel at Millstone Units 2 and 3 and Seabrook Unit 1. In addition, Mr. Galatis alleged with regard to Millstone Unit 3 that NU submitted a material false statement to the NRC associated with a license amendment and that an unanalyzed condition exists with regard to system piping for full-core offload events. With regard to Seabrook Unit 1, Mr. Galatis alleged technical specification violations associated with a criticality analysis.

The purpose of this informal public hearing is to obtain additional information from the Petitioners, NU, and the public for NRC staff use in evaluating the Petition. Therefore, this informal public hearing will be limited to information relevant to issues raised in the Petition and its Supplement. The staff will not offer any preliminary views on its evaluation of the Petition. The informal public hearing will be chaired by a senior NRC official who will limit presentations to the above subject.

The format of the informal public hearing will be as follows: opening remarks by the NRC regarding the general 10 CFR 2.206 process, the purpose of the informal public hearing, and a brief summary of the Petition and its Supplement (15 minutes); time for the Petitioners to articulate the basis for the Petition (45 minutes); time for the NRC to ask the Petitioners questions for purposes of clarification (15 minutes); time for NU to address the issues raised in the Petition (45 minutes); time for the NRC to ask NU questions for purposes of clarification (15 minutes); time for public comments relative to the Petition (90 Minutes); and time for licensee and Petitioners' final statements (15 minutes).

Members of the public who are interested in presenting information relative to the Petition should notify the NRC official, named below, 5 working days prior to the hearing. A brief summary of the information to be presented and the time requested should be provided in order to make appropriate arrangements. Time allotted for presentations by members of the public will be determined based upon the number of requests received and will be announced at the beginning of the hearing. The order for public presentations will be on a first received first to speak basis. Written statements will also be accepted and included in the record of the hearing. Written statements should be mailed to the U.S. Nuclear Regulatory Commission, Mailstop O-14H3, Attn: Stephen Dembek, Washington, DC 20555.

Requests for the opportunity to present information can be made by contacting Stephen Dembek, Project Manager, Division of Reactor Projects-I/II (telephone 301-415-1455) between 8:30 a.m. to 5:00 p.m. (EST). Persons planning to attend this informal public hearing are urged to contact the above 1 or 2 days prior to the informal public hearing to be advised of any changes that may have occurred.

Dated at Rockville, Maryland this 1st day of February, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Deputy Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation