Template NRR-058

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

March 7, 2000

Mr. Raymond P. Necci
Vice President-Nuclear Oversight
and Regulatory Affairs
c/o David A. Smith, Manager
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SUBJECT: MILLSTONE NUCLEAR POWER STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS (TAC NO. MA5326)

Dear Mr. Necci:

The Commission has issued the enclosed Amendment No.107 to Facility Operating License No. DPR-21 for the Millstone Nuclear Power Station, Unit 1 (MP1), in response to your application dated April 19, 1999, as supplemented by letters dated August 25, October 14, November 3, December 20, 1999, and February 29, 2000.

This amendment is the second and final response to your application for a complete set of new Technical Specifications (TS) to reflect the permanently shutdown status of the plant. Portions of your April 19, 1999, application, as supplemented, specifically Sections 3.10.C, D and E, of the current TS (Fuel Storage Pool Water Level, Crane Operability, and Crane Travel With a Spent Fuel Cask), were still being evaluated by the staff when Amendment 106 was issued on November 9, 1999. Amendment 106 approved a substantial portion of your permanently defueled TS (PDTS). The staff has now completed the evaluation of Sections 3.10.C, D and E, and the enclosed Amendment 107 is issued approving the remaining requested changes to the MP1 PDTS. Action on your April 19, 1999, license amendment application, as supplemented, is complete with the issuance of the enclosed amendment.

DF01

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Louis L. Wheeler

/Louis L. Wheeler, Senior Project Manager
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Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-245

Enclosures: 1. Amendment No. 107 to DPR-21
2. Safety Evaluation

cc w/encls: See next page

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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. 107
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 April 19, 1999, as supplemented by letters dated August 25, October 14, November 3, December 20, 1999, and February 29, 2000, 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.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-21 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No.107 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 90 days of issuance (including the relocation of Technical Specifications to the Final Safety Analysis Report). In addition, the licensee shall include the relocated information in the Final Safety Analysis Report submitted to the NRC, pursuant to 10 CFR 50.71(e), as described in the licensee's application dated April 19, 1999, as supplemented by letters dated August 25, October 14, November 3, December 20, 1999, and February 29, 2000, and evaluated in the staff's Safety Evaluation dated

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Masnik

Michael T. Masnik, Chief
Decommissioning Section
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 107

FACILITY OPERATING LICENSE NO. DPR-21

DOCKET NO. 50-245

Replace the following pages of Appendix A, Technical Specifications, with the attached pages.

Remove	Insert
i	i
3/4 10-3	3.1-1
	3.2-1
	3.2-2

Replace the following Technical Specifications Bases pages with the attached pages.

Remove	Insert
i	i
B 3/4 10-2	B 3.1-1
	B 3.1-2
	B 3.1-3
	B 3.2-1
	B 3.2-2
	B 3.2-3
	B 3.2-4

TABLE OF CONTENTS

3.1 DEFUELED SYSTEMS

3.1.1 Fuel Storage Pool Water Level

LCO 3.1.1 The Fuel Storage Pool Water Level shall be greater than or equal to 33 feet.

APPLICABILITY Whenever irradiated fuel is stored in the Fuel Storage Pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel Storage Pool Water Level not within limit.	A.1 Suspend all Fuel Handling Operations.	Immediately
	<u>AND</u>	
	A.2 Restore Fuel Storage Pool Water Level to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1 Verify the Fuel Storage Pool Water Level is greater than or equal to 33 feet. <u>AND</u> Record the Fuel Storage Pool Water Level.	24 hours

3.2 SPENT FUEL HANDLING

3.2.1 Reactor Building Crane Operability

LCO 3.2.1 The Reactor Building crane shall be OPERABLE.

APPLICABILITY When the Reactor Building crane is used for handling of a spent fuel cask.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor Building crane is INOPERABLE.	A.1 Suspend all Spent Fuel Cask handling and place the load in a safe condition.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1	Conduct a visual inspection of crane cables, sheaves, hook, yoke, and cask lifting trunnions. Conduct no-load mechanical and electrical tests to verify proper operation of crane controls, brakes, and lifting speeds. Conduct a load test by lifting the empty cask out of the pivot cradle. The above inspections and pre-lifting procedure shall meet the requirements of ANSI Standard B30.2, 1967.	Within 4 days prior to Spent Fuel Cask handling operations and every 4 days thereafter during spent fuel cask handling

3.2 SPENT FUEL HANDLING

3.2.2 Reactor Building Crane Travel with a Spent Fuel Cask

LCO 3.2.2 The Reactor Building crane loaded with a Spent Fuel Cask shall be prohibited from travel over irradiated fuel assemblies. The Reactor Building crane mode switch shall be in a "Mode 2" position and the mode switch key removed.

APPLICABILITY When the Reactor Building crane is used for handling of a spent fuel cask.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor Building Crane mode switch not in "Mode 2" position and mode switch key not removed.	A.1 Suspend all Spent Fuel Cask handling and place the load in a safe condition.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2 Demonstrate OPERABILITY of Reactor Building crane interlocks and limit switches which prevent crane travel over irradiated fuel assemblies.	Within 7 days prior to Spent Fuel Cask handling operations Every 7 days thereafter during Spent Fuel Cask handling

TABLE OF CONTENTS

B 3.1 DEFUELED SYSTEMS

B 3.1.1 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. A general description of the spent fuel storage pool design is found in Chapter 3 of the DSAR, (Ref. 1). The assumptions of the fuel handling accident are found in Chapter 5 of the DSAR (Ref. 2).

APPLICABLE SAFETY ANALYSIS

Although the unit is permanently shutdown and defueled, fuel handling accidents in the fuel storage pool are still possible.

A bounding calculation of the radiological consequences of such an accident in the spent fuel pool was performed, based on the following:

- Actual source term - radioactive decay since shutdown credited
- Failure of four assemblies - 248 fuel rods in four 8 x 8 assemblies
- Unfiltered ground release - no credit for secondary containment or standby gas treatment

The analysis concluded that 1) calculated doses at the exclusion area boundary and the low population zone are within 10CFR100 limits; and 2) calculated doses to the operating units and Unit 1 Control Rooms are within the limits set in GDC-19.

(continued)

BASES

LCO

The fuel storage pool water level is required to be greater than or equal to 33 feet above the bottom of the pool. The bottom of the fuel storage pool is located at an elevation of 69 feet, 9 inches above mean sea level (MSL). Therefore, the 33 feet limit corresponds to an elevation of 102 feet, 9 inches above MSL.

This water level preserves the assumptions of the fuel handling accident analysis and provides shielding to minimize the general area dose when irradiated fuel is being moved.

APPLICABILITY

This LCO applies whenever irradiated fuel assemblies are stored in the fuel storage pool.

ACTIONS

A.1

When the initial conditions for an accident cannot be met, action should be taken to preclude the accident from occurring. When the fuel storage pool level is lower than the required level, fuel handling activities should be suspended immediately. This does not preclude movement of items to a safe position.

Fuel handling activities as described in this specification include the movement of spent fuel, or other loads suspended from the fuel building crane or refueling machine, over irradiated fuel assemblies.

This effectively precludes a fuel handling accident from occurring.

A.2

This action is intended to restore the fuel storage pool level as soon as possible to minimize the time that the water level assumed in the accident analysis is not being met.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1

This SR ensures that the water level is within the established limit. The water level in the fuel storage pool must be checked periodically. The 24 hour Frequency is based on engineering judgement and is considered adequate because of available indication of level changes and the large volume of water in the pool. Water level changes are controlled by facility procedures and level changes are unlikely based on operating experience.

References

1. DSAR Chapter 3
2. DSAR Chapter 5

B 3.2 SPENT FUEL HANDLING

B 3.2.1 Reactor Building Crane Operability

BASES

BACKGROUND

The purpose of this specification is to preclude the possibility of dropping a spent fuel cask over irradiated fuel in the fuel storage pool.

A description of the Reactor Building crane design improvements was provided by NNECO to the NRC on June 29, 1973. The modification improvements were described as a "Cask Drop Prevention System." By letter dated December 30, 1975, the NRC informed NNECO that the proposed improvements were acceptable. However, the NRC also requested NNECO to submit proposed Technical Specifications to assure safe operation and continued surveillance of the Reactor Building crane. NNECO submitted the proposed Technical Specifications on April 1, 1976, and the NRC approved new Technical Specifications, including the "Crane Operability" LCO, as Amendment 27 to License No. DPR-21.

APPLICABLE SAFETY ANALYSIS

The "Cask Drop Prevention System" utilizes a redundant hoist system rated at 110 tons for the main hoist. This redundant system ensures that a load will not be dropped for all postulated credible single-component failures. The range of component failure examined extends over the total load path from the cask trunnions through the cask lifting yoke and redundant hoist system to the crane bridge structure. In addition, once the crane is set into the cask handling mode, its travel over the fuel pool will be limited to the cask storage area of the spent fuel pool. The operability requirements of the Reactor Building crane ensure that all redundant features of the crane have been adequately inspected.

Spent fuel cask drop over irradiated fuel in the fuel storage pool is precluded by these features as well as the features described in LCO and Surveillance Requirement 3.2.2 of these Technical Specifications.

(continued)

B 3.2 SPENT FUEL HANDLING

B 3.2.1 Reactor Building Crane Operability

BASES

LCO

The Reactor Building crane is required to be OPERABLE. The operability is established by:

- a visual inspection of the crane cables, sheaves, hook, yoke, and cask lifting trunnions,
- conducting no-load mechanical and electrical tests to verify proper operation of crane controls, brakes and lifting speeds,
- conducting a load test by lifting an empty cask out of the pivot cradle.

Maintaining the Reactor Building crane OPERABLE preserves the assumption of preventing a cask drop accident.

APPLICABILITY

This LCO applies whenever the Reactor Building crane is used for handling of a spent fuel cask.

ACTIONS

A.1

When the operability requirements for the Reactor Building crane cannot be met, steps should be taken to preclude a Spent Fuel Cask drop accident from occurring. Fuel cask handling activities should be suspended immediately and the load placed in a safe condition. This will effectively preclude a spent fuel cask drop accident from occurring.

SURVEILLANCE REQUIREMENTS

SR 3.2.1

This SR verifies operability of the Reactor Building crane and ensures that the redundant features of the crane have been adequately inspected. The redundant hoist system ensures that a load will not be dropped for all postulated credible single-component failures. The Frequency is appropriate because operability is required to be established before Spent Fuel Cask handling operations commence.

(continued)

B 3.2 SPENT FUEL HANDLING

B 3.2.2 Reactor Building Crane Travel with a Spent Fuel Cask

BASES

BACKGROUND

The purpose of this specification is to preclude the possibility of dropping a spent fuel cask over irradiated fuel in the fuel storage pool. The Reactor Building crane has a 2-position mode switch which is designed to restrict crane motion, when in "Mode 2," as follows:

- It prevents a spent fuel cask height above the refueling floor not greater than 6 inches, and
- It establishes a predetermined path which specifically excludes the area above irradiated fuel by interlocks and limit switches.

This specification, in conjunction with LCO 3.2.1, ensures that a fuel cask drop over irradiated fuel in the fuel storage pool is prevented from occurring.

APPLICABLE SAFETY ANALYSIS

The "Cask Drop Prevention System" features a single-failure proof design that prevents a spent fuel cask drop over the fuel storage pool with resultant damage to irradiated fuel and/or plant equipment and structures. Once the Reactor Building crane mode switch is set into the cask handling mode, its travel over the fuel storage pool will be limited to the cask storage area of the fuel pool. This design feature as well as associated crane interlocks and limit switches ensure that a spent fuel cask drop will not occur over the irradiated fuel in the fuel storage pool.

An event initiated by a spent fuel cask drop over the irradiated fuel in the fuel storage pool is precluded by these features as well as the features described in LCO and Surveillance Requirement 3.2.1 of these Technical Specifications.

(continued)

B 3.2 SPENT FUEL HANDLING

B 3.2.2 Reactor Building Crane Travel with a Spent Fuel Cask

BASES

LCO	The Reactor Building crane mode switch is required to be in the "Mode 2" position with its key removed. This mode switch position is an engineered control which restricts crane travel to a path which excludes the area above the irradiated fuel in the fuel storage pool. Also, the height of a spent fuel cask loaded on the crane is restricted to a height of no greater than 6 inches above the refueling floor. Maintaining the Reactor Building crane mode switch, associated crane limit switches, and interlocks preserves the assumption of preventing a cask drop accident.
APPLICABILITY	This LCO applies whenever the Reactor Building crane is used for handling of a spent fuel cask.
ACTIONS	<u>A.1</u> When mode switch requirements for the Reactor Building crane cannot be met, steps should be taken to preclude a spent fuel cask drop accident from occurring. Fuel cask handling activities should be suspended immediately and the load placed in a safe condition. This will effectively preclude a spent fuel cask drop accident from occurring.
SURVEILLANCE REQUIREMENTS	<u>SR 3.2.2</u> This SR demonstrates operability of the Reactor Building crane interlocks and limit switches which restricts the height of the crane load (i.e., the spent fuel cask bottom) to no more than 6 inches above the refueling floor and restricts crane path from traveling over the irradiated fuel assemblies. The Frequency is appropriate because operability is established before spent fuel cask handling operations start and operability is periodically assured during spent fuel cask handling.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 107

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 April 19, 1999, as supplemented by letters dated August 25, October 14, November 3, December 20, 1999, and February 29, 2000, the Northeast Nuclear Energy Company (NNECO, the licensee) submitted a request for changes to the Millstone Nuclear Power Station, Unit 1 (MP1) Technical Specifications (TS). The requested changes would replace the entire set of current TS (CTS) with a completely new set of TS to reflect the permanently shutdown and defueled status of the plant. The staff issued Amendment 106 on November 9, 1999, when a substantial portion of its evaluation of the licensee's application had been completed. At the time Amendment 106 was issued, the evaluation of the proposed changes to CTS Sections 3.10.C, D, and E was still in progress. The evaluation of these sections has now been completed, and is documented in this safety evaluation. The supplemental letters dated August 25, October 14, November 3, December 20, 1999, and February 29, 2000, provided clarifying information that did not change the scope of the April 19, 1999, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

By letter dated July 21, 1998, NNECO certified to the Nuclear Regulatory Commission (NRC), under the provisions of Section 50.82(a) of Title 10 of the *Code of Federal Regulations* (10 CFR), that MP1 had permanently ceased operations and that the fuel had been permanently removed from the reactor vessel. NNECO is therefore prohibited by 10 CFR 50.82(a)(2) from operating the plant or placing fuel in the reactor vessel. NNECO has determined that major changes to the CTS are necessary to reflect the permanently shutdown and defueled status of the plant. Therefore, by letter dated April 19, as supplemented by letters dated August 25, October 14, November 3, December 20, 1999, and February 29, 2000, NNECO submitted a proposed license amendment that would replace the entire CTS, which is designed primarily to support power operations, with a new set of TS to reflect the permanently shutdown and defueled status of the plant.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "...those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." (Statement of Consideration, "Technical Specification for Facility Licenses; Safety Analysis Reports," 33 FR 18610 (December 17, 1968).) Pursuant to 10 CFR 50.36, TS are required to include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

On July 22, 1993, the Commission issued its Final Policy Statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated surveillances should remain in the TS. The Final Policy Statement established four criteria to define the scope of equipment and parameters to be included in the improved STS LCOs. TS LCOs that do not satisfy any of these four criteria may be removed from the TS and relocated to licensee controlled documents. These criteria were developed for licenses authorizing operation; nevertheless, these criteria, now codified by 10 CFR 50.36, are the source of the TS requirements for safe storage of spent fuel. A general discussion of these considerations is provided below.

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for "[i]nstalled instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure [RCP] boundary." Since no fuel is present in the RCP at the MP1 facility, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for "[a] process variable, design feature, or operating restriction that is an initial condition of a design basis accident [DBA] or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. While this criterion was developed for operating reactors, there are some DBAs that continue to apply to a plant authorized only to handle, store, and possess nuclear fuel. The scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for "[a] structure, system, or component [SSC] that is part of the primary success path and which functions or actuates to mitigate a [DBA] or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The intent of this criterion is to capture into the TS only those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of a combination and sequence of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. While there are no transients that continue to apply to MP1, there are some DBAs that continue to apply to a plant authorized only to handle, store, and possess nuclear fuel. As stated above, the scope of DBAs applicable to a plant with a reactor that is permanently shut down and defueled is markedly reduced from those postulated for an operating plant.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states the TS LCOs must be established for SSCs "...which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs.

Addressing administrative controls, 10 CFR 50.36(c)(5) states that they "...are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner...." The particular administrative controls to be included in the TS, therefore, are the provisions that the Commission deems essential for the safe operation of the facility that are not already covered by other regulations. Accordingly, the staff has determined that administrative control requirements that are not specifically required under Section 50.36(c)(5), and which are not otherwise necessary to obviate the possibility of an abnormal situation or an event giving rise to an immediate threat to the public health and safety, may be relocated to more appropriate documents (e.g., Quality Assurance Program, Security Plan, or Emergency Plan), which are subject to regulatory controls. Similarly, while the required content of TS administrative controls is specified in 10 CFR 50.36(c)(5), particular details may be relocated to licensee-controlled documents, where other regulations provide adequate regulatory control.

In April of 1995, the Commission issued NUREG-1433, "Standard Technical Specifications, General Electric Plants (BWR/4)." This NUREG provides a set of standardized TS evaluated by the staff and found to be acceptable for an operating boiling water reactor (BWR/4) to meet the criteria provided in the Commission's Final Policy Statement for Nuclear Power Reactors, dated July 22, 1993.

Coincident with the change to reflect the change in plant status, the requested amendment incorporates Standard TS (STS) guidance provided by the NRC. In some cases, the licensee proposed deleting certain CTS items in order to bring MP1 more in line with current regulatory positions on TS content as described in NUREG-1433.

3.0 EVALUATION

3.1 CTS 3/4.10.C (Fuel Storage Pool Water Level)

CTS Subsection 3.10.C has been retained and reformatted to be consistent with the STS in NUREG-1433 for General Electric plants. The fuel storage pool water level requirement provides adequate water shielding when irradiated fuel is being moved and preserves the assumptions of the fuel handling accident analysis. The proposed TS, renumbered as PDTS 3.1.1, requires that water level be maintained greater than or equal to 33 feet whenever irradiated fuel is stored in the pool. The proposed TS LCO requires that whenever the water level is not within this limit, all fuel handling operations be suspended and the required water level be restored immediately. The proposed surveillance requirement is to verify the water level is greater than or equal to 33 feet, and record the level every 24 hours. The proposed PDTS has the same requirements as the CTS and exceeds the current requirement in the STS. Therefore, we find this PDTS acceptable.

3.2 CTS 3/4.10.D (Crane Operability), and
CTS 3/4.10.E (Crane Travel with a Spent Fuel Cask/Crane Interlocks and Switches)

The licensee proposes to retain the requirements in the CTS pertaining to the operability and surveillance of the containment refueling crane and the system used to transfer fuel between the containment and the fuel storage pool. The CTS requirements, with no changes, will be retained in PDTS 3.2.1, "Reactor Building Crane Operability," and PDTS 3.2.2, "Reactor Building Crane Travel with a Spent Fuel Cask."

3.2.1 Heavy Loads Background

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides guidelines and recommendations for licensees to assure safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over spent fuel assemblies, over the core, and over safety-related equipment. NUREG-0612 defines a heavy load as any load carried in a given area during the operation of the plant that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool.

Phase I of NUREG-0612 implementation provides guidelines for reducing the likelihood of dropping heavy loads and limiting the resulting potential consequences of a drop. The guidelines are focused on establishing safe load paths, procedures for load handling operations, training of crane operators, the design of lifting devices, and the design, testing, inspection, and maintenance of cranes. Phase II provides guidelines for mitigating the consequences of dropped loads, including the use of a single-failure-proof crane, the use of electrical interlocks and mechanical stops to restrict crane travel, and the performance of load drop and consequence analyses to assess the impact of dropped loads on plant safety. Generic Letter (GL) 85-11, "Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612," dated June 28, 1985, made it voluntary for licensees to implement the requirements of NUREG-0612, Phase II. However, via GL 85-11, licensees are encouraged to implement actions they perceive to be appropriate to maintain safety.

3.2.2 Heavy Loads Evaluation

By retaining the requirements of CTS 3/4.10.D and E in the PDTS, the movement of heavy loads over the spent fuel pool is restricted to help limit the potential onsite and offsite consequences of heavy load drops. This is within the guidance of NUREG-0612 and is, therefore, acceptable. The licensee retained the requirements of these CTS sections in their entirety in PDTS LCO 3.2.1, "Reactor Building Crane Operability," and LCO 3.2.2, "Reactor Building Crane Travel with a Spent Fuel Cask." Additionally, the licensee adopted the improved STS format for these PDTS sections.

The purpose of these specifications is to preclude the possibility of dropping a spent fuel cask over irradiated fuel in the fuel storage pool. This is accomplished by the LCOs and SRs that the licensee is retaining in the PDTS.

LCO 3.2.1 establishes the operability requirements for the Reactor Building Crane. The operability is established by a visual inspection of the crane cables, sheaves, hook, yoke, and cask lifting trunnions; conducting a no load mechanical and electrical tests to verify proper operation of crane controls, brakes and lifting speeds; and conducting a load test by lifting an empty cask out of the pivot cradle.

The purpose of LCO 3.2.2 is to preclude the possibility of dropping a spent fuel cask over irradiated fuel in the fuel storage pool. This is accomplished by restricting the movement of heavy loads by the use of interlock switches and by imposing the restriction that the reactor building crane mode switch be in the "Mode 2" position with its key removed when the crane is used for handling a spent fuel cask. This mode switch position is an engineered control feature which restricts crane travel in two ways: it prevents a spent fuel cask from being at a height above the refueling floor greater than 6 inches, and it establishes a predetermined path which specifically excludes the area above irradiated fuel from crane travel by interlocks and limit switches. These specifications help ensure that a fuel cask drop over irradiated fuel in the fuel storage pool is prevented.

As part of the licensee's heavy load handling system, the licensee has a Cask Drop Prevention System which utilizes a redundant hoist system rated at 110 tons for the main hoist, and features a single failure proof design that prevents a spent fuel cask drop over the fuel storage pool with resultant damage to the irradiated fuel and/or plant equipment and structures. This redundant system ensures that a load will not be dropped for all postulated credible single-component failures. The range of component failure examined by the licensee extends over the total load path from the cask trunnions through the cask lifting yoke and redundant hoist system to the crane bridge structure. In addition, once the crane is in the cask handling mode, its travel over the fuel pool will be limited to the cask storage area of the pool. This design feature and associated crane interlocks and limit switches help ensure that a spent fuel cask drop will not occur over the irradiated fuel in the fuel storage pool. The operability requirements of the Reactor Building crane ensure that all redundant features of the crane have been adequately inspected. A spent fuel cask drop over irradiated fuel in the fuel storage pool is precluded by these features.

CTS 4.10.D (Crane Operability) and CTS 4.10.E (Crane Interlocks and Switches) are retained in the PDTS as SR 3.2.1, and 3.2.2 respectively. SR 3.2.1 and SR 3.2.2 specify the minimum frequency and type of surveillance to be applied to the refueling and spent fuel handling systems. These specifications were designed to assure the safe handling of spent fuel casks.

Based on the LCOs and SRs in PDTS Sections 3.2.1 and 3.2.2 which restrict the movement of heavy loads over spent fuel, and the analysis performed by the licensee to evaluate the consequences of dropping several loads and components of differing weights and sizes that showed the resulting dose to be within 10CFR100 and GDC19 limits, the staff finds that the licensee's analysis has adequately addressed the handling of heavy loads to limit the potential consequences of a dropped load and are consistent with NUREG-0612. PDTS 3.2.1 and 3.2.2 are, therefore, acceptable.

4.0 <u>STATE CONSULTATION</u>

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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 35208 dated June 30, 1999). 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.

6.0 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.

Principal Contributors: J. Guo
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Date: March 7, 2000

TEMPLATE = NRR-058
Needs to Be SCANNED

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Louis L. Wheeler, Senior Project Manager
Decommissioning Section
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-245

Enclosures: 1. Amendment No. 107 to DPR-21
2. Safety Evaluation

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* SE incorporated w/no major revisions ** See prior concurrence

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