January 28, 1977

Docket No.: 50-271

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JRBuchanan

Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Rodd
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications in response to your application dated November 8, 1976, and staff discussions.

This amendment modifies the Technical Specifications to provide for specific surveillance and testing of the reactor building crane prior to fuel cask handling.

However, so that there is no misunderstanding, prior to any cask movement you should submit for our review and approval an evaluation and description of the cask lifting devices exclusive of the above. The evaluation should provide assurance that the designed redundancy of the crane will be maintained and that there will be nil displacement of the load in the event of single failure.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 29
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

*See previous Yellow for concurrences.

OFFICE➤	ORB#4:DOR *	ORB#4:DOR *	~~C-ORB#4~~:DOR	APCSB:DSS *	OELD	
SURNAME➤	RIngram	PDiBenedetto	RReid	ETomlinson		
DATE➤	1/ /77	1/ /77	1/28/77	1/ /77	1/27/77	

Form AEC-318 (Rev. 9-53) AECM 0240

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Robert W. Reid, Chief
Operating Reactors Branch #4
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1. Amendment No.
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OFFICE	ORB#4:DOR	ORB#4:DOR	C-ORB#4:DOR	APCSB:DSS	OELD	
SURNAME	RIngram	PDiBenedetto	RReid	E[illegible]inson		
DATE	1/3/77	1/3/77	12/ /76	1/03/77	12/ /76	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Yankee Atomic Electric
  Company

U. S. Environmental Protection
  Agency
Region I Office
ATTN: EIS COORDINATOR
JFK Federal Building
Boston, Massachusetts 02203

cc w/enclosures and cy of VY's
  filing dtd.: 11/8/76
Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated November 8, 1976, 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
181	181
185	185
-	185a
187a	187a

The changed areas on the revised pages are shown by marginal lines.

3.12 LIMITING CONDITION FOR OPERATION

3.12 REFUELING AND SPENT FUEL HANDLING

Applicability:

Applies to fuel handling, core reactivity limitations, and spent fuel handling.

Objective:

To assure core reactivity is within capability of the control rods, to prevent criticality during refueling, and to assure safe handling of spent fuel casks.

Specification:

A. Refueling Interlocks

The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks, listed below, shall be operable except as specified in Specifications 3.12.D and 3.12.E.

1. Control Rod Blocks

a. Mode switch in Startup/Hot Standby and refueling platform over the reactor.

4.12 SURVEILLANCE REQUIREMENT

4.12 REFUELING AND SPENT FUEL HANDLING

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling and to the testing of the reactor building crane.

Objective:

To verify the operability of instrumentation and interlocks used in refueling and the operability of the reactor building crane.

Specification:

A. Refueling Interlocks

Prior to any fuel handling, with the Head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

.12 LIMITING CONDITION FOR OPERATION

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

2. SRMs shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification 3.12.B.

F. Fuel Movement

Fuel shall not be moved or handled in the reactor core for 24 hours following reactor shutdown to cold shutdown conditions.

G. Crane Operability

1. The reactor building crane shall be operable when the crane is used for handling of a spent fuel cask.

4.12 SURVEILLANCE REQUIREMENT

1. This surveillance requirement is the same as that given in Specification 4.12.A.

2. This surveillance requirement is the same as that given in Specification 4.12.B.

F. Fuel Movement

Prior to any fuel handling or movement in the reactor core, the licensed operator shall verify that the reactor has been in the cold shutdown condition for a minimum of 24 hours.

G. Crane Operability

1.a. Within one month prior to spent fuel cask handling operations, an inspection of crane cables, sheaves, hook, yoke and cask lifting trunnions will be made. These inspections shall meet the requirements of ANSI Standard B30.2, 1967. A crane rope shall be replaced if any of the replacement criteria given in ANSI B30.2.0-1967 are met.

3.12 LIMITING CONDITIONS FOR OPERATION

2. Crane Travel

Spent fuel casks shall be prohibited from travel over irradiated fuel assemblies.

4.12 SURVEILLANCE REQUIREMENT

b. No-load mechanical and electrical tests will be conducted prior to lifting the empty cask from its transport vehicle to verify proper operation of crane controls, brakes and lifting speeds. A functional test of the crane brakes will be conducted each time an empty cask is lifted clear of its transport vehicle.

2. Crane Travel

Crane travel limiting mechanical stops shall be installed on the crane trolley rails prior to cask handling operations to prohibit cask travel over irradiated fuel assemblies.

3.12 & 4.12 (Continued)

G. The operability requirements of the reactor building crane ensures that the redundant features of the crane have been adequately inspected just prior to using it for handling of a spent fuel cask. The redundant hoist system ensures that a load will not be dropped for any postulated credible single component failures. Details of the design of the redundant features of the crane and specific testing requirements for the crane are delineated in the Vermont Yankee document entitled "Reactor Building Crane Modification" (December, 1975).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

Introduction

Our safety evaluation dated June 1, 1971, for Vermont Yankee Nuclear Power Station (VYNPS) expressed concern about the accidental dropping of a fuel cask into the spent fuel pool, the subsequent rupture of the floor of the pool and the eventual dewatering of the stored spent fuel. Since this concern did not affect reactor operation, resolution was not required prior to issuance of an operating license for VYNPS.

Background

On April 5, 1973, Vermont Yankee Nuclear Power Corporation (VYNPC) proposed installing certain protective devices which would prevent accidental dropping of a spent fuel cask. On June 14, 1973, a meeting was held between VYNPC and the NRC staff for the purpose of discussing the proposed resolution to the cask drop accident. At this meeting we expressed certain concerns and areas were pointed out that needed further investigation by VYNPC.

On July 23, 1973, VYNPC, as a result of further investigation, proposed to install a substitute device to prevent the fuel cask from dropping under postulated conditions. This proposal was to modify the existing crane trolley to provide redundancy in the load-carrying path from the cask to crane trolley itself so that no single failure would allow the cask to drop.

On December 30, 1975, VYNPC submitted for our review a report entitled "Reactor Building Crane Modification." This report contained a detailed description of the modification proposed by VYNPC on July 23, 1973. We requested additional information relating to the modification on June 2, 1976. VYNPC responded to our request on July 2, 1976.

On November 8, 1976, VYNPC requested a change to the Appendix A Technical Specifications for VYNPS. The proposed change would incorporate into the Technical Specifications surveillance and operational requirements to provide increased assurance that the reactor building crane will be operable prior to using it for lifting a spent fuel cask.

Discussion

The overhead crane handling system for VYNPS consists of an overhead, bridge-type crane, spent fuel cask lifting devices, and controls. The overhead crane handling system is used during plant operation for lifting and transporting the spent fuel shipping cask between the spent fuel pool and the cask decontamination/shipping area. The overhead crane is located indoors in a controlled environment and has a main hoist rated at 110 tons. The overhead crane handling system has been designed to minimize the potential of a spent fuel cask drop accident which could result in an unacceptable release of radioactive materials by (1) replacing the trolley on the overhead crane with a new trolley designed to single failure criteria pursuant to NRC Branch Technical Position APCSB 9-1 and (2) restricting the path of travel of the crane and spent fuel cask so that the cask passes over the minimum amount of safety related equipment.

Evaluation

We have reviewed the control system for limiting the crane/cask travel path as described by the VYNPC in their letter dated December 30, 1975. We conclude there is adequate component redundancy to preclude the crane traveling outside the prescribed path of travel, and the control system for this purpose is acceptable.

The overhead crane has redundancy in the areas of brakes, gear trains, reeving system, load attaching points, and cask lifting devices, as well as crane control components and systems which are designed fail safe. Based on our review of data provided by the VYNPC, we conclude the integrated design of crane, controls, and cask lifting devices meets the intent of Branch Technical Position APCSB 9-1 regarding single failure criteria except in the specific areas of the crane reeving system, protection against "two blocking" fatigue analysis, load brakes, and crane hooks.

The crane reeving system which was designed and constructed prior to the development of the NRC Branch Technical Position, does not meet the recommended criteria for wire rope safety factors and fleet angles. The purpose of these criteria is to ensure a design which minimizes wire rope stress and thereby provides maximum assurance of crane safety under all operating and maintenance conditions. Because the crane reeving system does not meet these recommended criteria there is a possibility of an accelerated rate of wire rope wear occurring. Accordingly to compensate in these design areas, the VYNPC, by letter dated July 2, 1976, has committed to a specific program of wire rope inspection and replacement, the purpose of which is to ensure that the entire length of the wire rope will be maintained as close as practicable to original design safety factors at all times. This inspection/replacement program provides an equivalent level of protection to the methods suggested in our wire rope safety and crane fleet angle criteria and will assure that accelerated wire rope wear will be detected before crane use and satisfies our concerns, and we conclude the crane reeving system is acceptable.

The crane control system does not provide adequate protection against "two blocking" in the event of a fused contactor in the main hoist control circuitry. However, the crane has dual upper travel limit switches which control a single electric circuit to stop all hoisting motion in the event the upper travel limit is reached. To compensate for possible single failure in the circuit, the VYNPC has installed a second control circuit in parallel with the original one. We conclude the proposed addition will provide adequate protection against two-blocking, and the control system is acceptable.

VYNPC has provided an analysis which shows that fatigue will not be a consideration for this particular crane due to the low stress levels and limited number of loading cycles the crane can be expected to see during its anticipated 40 year life. We conclude the crane design provides more than adequate margin for fatigue considerations and is, therefore, acceptable.

Initially, VYNPC had proposed to utilize a crane hook not in conformance with BTP APCSB 9-1 criteria for certain limited operations. We found this proposed method of operation unacceptable, and the VYNPC, in their July 2, 1976 letter, agreed to discard this proposed operational procedure. The new crane hook with dual load attaching points and dual paths for load transfer to the load block will be used for all crane operations. Since the new hook meets our criteria and satisfies our concerns, we conclude it is acceptable.

Initially, VYNPC did not provide adequate data to show that the load would be stopped by the load brakes within 3 inches of vertical travel in the event of failure of one wire rope. Additional data were, however, included in VYNPC's July 2, 1976 letter. We reviewed the data provided and found it to be adequate, and we conclude the main hoist load brakes are acceptable.

Based on our evaluation of the data provided, and the commitments made by the VYNPC in the areas of crane reeving and two blocking protection, we conclude that the modified overhead handling system for VYNPS is acceptable.

VYNPC has agreed to install crane travel limiting mechanical stops on the crane trolley rails prior to cask handling operations to preclude the cask from traveling over irradiated fuel assemblies. We find that the mechanical stops are adequate to prohibit cask travel over irradiated fuel and therefore conclude that this modification is acceptable.

Technical Specifications

By letter dated November 8, 1976, VYNPC submitted proposed Technical Specifications relating to the operability and surveillance of the reactor building crane prior to fuel cask handling. During our review of the proposed change, we found it necessary to make changes to the VYNPC submittal. We have discussed these changes with VYNPC and they have concurred with the changes.

The modified Technical Specifications provide for specific surveillance and testing of the reactor building crane prior to fuel cask handling and will enhance the level of reliability associated with crane operation during cask handling. Based on the above, we conclude that the Technical Specifications as modified are acceptable.

Based on our review of the crane modifications, the Technical Specifications, and the additional information provided by VYNPC, we find that the cask drop accident has been adequately resolved by VYNPC. Thus, we conclude that fuel cask handling, as proposed with the modified reactor building crane under the surveillance and testing requirements of the proposed Technical Specifications, is acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 28, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 29 to Facility Operating License No. DPR-28, issued to Vermont Yankee Nuclear Power Corporation (the licensee), which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (the facility) located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to provide for specific surveillance and testing of the reactor building crane prior to fuel cask handling.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 8, 1976, (2) Amendment No. 29 to License No. DPR-28, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 28th day of January 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors