

Maine Yankee

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October 14, 2003
MN-03-064 RA-03-153

UNITED STATES NUCLEAR REGULATORY COMMISSION
Attention: Document Control Desk
Washington, DC 20555

- References:
- (1) License No. DPR-36 (Docket No. 50-309)
 - (2) Maine Yankee Letter to NRC, Certifications of Permanent Cessation of Power Operation and Permanent Removal of Fuel from the Reactor, (MN-97-89), dated August 7, 1997.
 - (3) NRC Administrative Letter (AL) 95-06: "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995.

Subject: Proposed Change No. 217: License and Technical Specifications

Pursuant to 10 CFR 50.90, Maine Yankee hereby requests Nuclear Regulatory Commission (NRC) review and approval of a modification to the License and to the Technical Specifications.

The proposed changes to the License would eliminate License information that no longer applies to a licensee that has permanently ceased operation. Maine Yankee recommends deletion of the paragraph under License title "Maximum Power Level". Pursuant to 10 CFR 50.82, licensees who have docketed certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel (Reference 2) are no longer authorized to operate the reactor. Maine Yankee also recommends the deletion of License paragraph 2.B.(8) since with permanent cessation of operations and pursuant to 10 CFR 50.51, the License continues in effect beyond the expiration date until the Commission notifies the licensee in writing that the License is terminated.

The proposed changes to the Technical Specifications would simplify the Technical Specifications by removing certain design and administrative requirements and relocating them to the Defueled Safety Analysis Report or the Quality Assurance Program. This relocation is being proposed pursuant to the criteria contained in 10 CFR 50.36 and in accordance with NRC Administrative Letter 95-06 (Reference No. 2). Additionally, Maine Yankee proposes to eliminate technical specifications that will no longer be applicable following the transfer of the last fuel assembly from the spent fuel pool to the Independent Spent Fuel Storage Installation (ISFSI).

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The Technical Specification proposed changes consist, in part, of the following:

Deletion of Technical Specification, Section 1.0, USE AND APPLICATION since this section does not apply to a completely defueled plant.

Deletion of Technical Specification, Section 3.0, LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY since this section does not apply to a completely defueled plant.

Deletion of Technical Specification, Section 3.1, DEFUELED SYSTEMS since this section only applied during the storage of spent fuel assemblies in the fuel storage pool or during movement of irradiated fuel assemblies in the fuel storage pool.

Deletion of the specified minimum distance to the boundary of the exclusion area from Technical Specification Design Features Section 4.1 and relocation of the requirement to the Defueled Safety Analysis Report.

Deletion of Technical Specification, Section 4.2, Fuel Storage and associated Figure 4.1-1, Spent Fuel Pool Assembly Placement since this section only applied during the storage of irradiated fuel in the fuel storage pool.

Deletion of the following Technical Specification, ADMINISTRATIVE CONTROLS, and relocation of these administrative controls to the Maine Yankee Quality Assurance Program:

Section 5.1, "Responsibility"

Section 5.2, "Organization"

Section 5.3, "Unit Staff Qualifications"

Section 5.4, "Training"

Section 5.5, "Procedures"

Subsection 5.6.1, "Radiation Protection Program"

Subsection 5.6.2, "Offsite Dose Calculation Manual (ODCM)"

Subsection 5.6.3, "Radioactive Effluent Controls Program"

Deletion of Technical Specification, Subsection 5.6.4, Technical Specifications (TS) Bases Control Program since as stated below the Technical Specification Bases Section will cease to exist.

Deletion of Technical Specification, SubSection 5.6.5, Spent Fuel Water Chemistry Program since irradiated fuel will no longer be stored in the Fuel Storage Pool.

Deletion of Technical Specification, SubSection 5.6.6, Inservice Testing Program since safety related ASME Code Class 1, 2, and 3 components, including associated supports, no longer exist at the Maine Yankee plant.

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Deletion of Technical Specification, Section 5.7, Reporting Requirements, and relocation of these administrative controls to the Maine Yankee Quality Assurance Program.

Deletion of Technical Specification, Section 5.8, High Radiation Area, and relocation of these administrative controls to the Maine Yankee Quality Assurance Program.

Deletion of Technical Specification - Bases Section since the Section is not applicable to a completely defueled plant.

The License and Technical Specification Proposed Changes, Attachment I, provides a description section, identifies the proposed changes and bases for changes, provides a no significant hazards consideration determination and an environmental impact consideration determination. Attachment II provides marked-up pages showing the proposed changes to the License. Attachment III provides a copy of the affected Technical Specification page with changes annotated. Attachment IV provides an updated version of the revised Technical Specification pages.

Maine Yankee is requesting approval of the proposed change prior to completion of transfer of spent fuel assemblies to the ISFSI. Transfer of spent fuel assemblies is expected to be complete by February, 2004.

Maine Yankee requests implementation of the approved license amendment to be within 30 days after transfer of the last spent nuclear fuel assembly from the spent fuel pool to the Independent Spent Fuel Storage Installation (ISFSI).

Also, as evident in the revised Technical Specification, Attachment IV, the Technical Specification when approved will be reduced to a single paragraph in Section 4 that provides a description of the plant location. This remaining information is not germane to Technical Specification requirements under 10 CFR 50.36. Maine Yankee therefore requests NRC consider the deletion of Technical Specifications from the Maine Yankee Part 50 License.

These changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in the margin of safety. These proposed changes have undergone an Independent Safety Review. The Independent Review and Audit Committee has also reviewed these proposed changes. A representative of the State of Maine is being informed of this request by a copy of this letter.

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If you have any questions, please contact us.

Sincerely,



Thomas L. Williamson, Director
Nuclear Safety and Regulatory Affairs

Attachments: (1) Evaluation of Proposed Technical Specification Change
(2) Proposed Technical Specification Changes (mark-up)
(3) Proposed Technical Specification pages (retyped)

cc: Dr. R. R. Bellamy, NRC Region I
Mr. J. T. Buckley, NRC NMSS Project Manager, Decommissioning
P. Craighead, Esq., State of Maine, Nuclear Safety Advisor
Mr. P. J. Dostie, State of Maine, Division of Health Engineering
D. R. Lewis, Esq., Shaw Pittman
Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. M. C. Roberts, NRC, Region I
Mr. R. Shadis, Friends of the Coast

AFFIDAVIT

STATE OF MAINE

Then personally appeared before me, Thomas L. Williamson, who being duly sworn did state that he is the director, Nuclear Safety and Regulatory Affairs of Maine Yankee Atomic Power Company, that he is duly authorized to execute and file the foregoing request in the name and on the behalf of Maine Yankee Atomic Power Company, and that the statements therein are true to the best of his knowledge and belief.

Donna L. Pelletier
Notary Public

My commission expires Dec. 11, 2006

ATTACHMENT I

PROPOSED CHANGE NO. 217

DESCRIPTION AND EVALUATION OF CHANGES

**ATTACHMENT I
DESCRIPTION AND EVALUATION OF CHANGES
PROPOSED CHANGE NO. 217**

Subject: Proposed Changes: License and Technical Specifications

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DESCRIPTION AND EVALUATION OF CHANGES
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DESCRIPTION

This submittal is a request to amend Operating License No. DPR-36 for Maine Yankee. Maine Yankee proposes changes to its License and to the associated Technical Specifications. The proposed changes would eliminate License information that no longer applies to a license that has permanently ceased operation. The proposed changes would also simplify the Technical Specifications. Maine Yankee proposes to remove certain design and administrative requirements, relocate them to the to the Defueled Safety Analysis Report or the Quality Assurance Program and make other minor administrative changes. The Technical Specification relocation is being proposed pursuant to the criteria contained in 10 CFR 50.36 and in accordance with NRC Administrative Letter 95-06. Additionally, Maine Yankee proposes to eliminate technical specifications which will no longer be applicable following the transfer of the last fuel assembly from the spent fuel pool to spent fuel storage cask. The proposed changes to the Technical Specification are similar to the NRC approved changes provided to the Yankee Atomic Electric Company Technical Specifications on April 18, 2003 (TAC No. L52086).

PROPOSED CHANGES & BASES FOR CHANGES

The proposed changes consist of the following:

Deletion of License paragraph under the title "Maximum Power Level".

Pursuant to 10 CFR 50.82, licensees who have docketed certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are no longer authorized to operate the reactor. The paragraph under "Maximum Power Level" does not apply to the Maine Yankee plant since it has ceased from operations.

Deletion of License paragraph 2.B.(8).

Pursuant to 10 CFR 50.51, the License for a plant that has permanently ceased operation continues in effect beyond the expiration date until the Commission notifies the licensee in writing that the License is terminated.

Deletion of Technical Specification, Section 1.0, USE AND APPLICATION.

Deletion of this Section is proposed since the section does not apply to a completely defueled facility. Technical Specification use and application of terms, such as, Required Actions, Logical Connectors, Completion Times, and Surveillance Requirements are associated with Limiting Conditions for Operation (LCO's) minimum requirements for ensuring the safe storage of

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irradiated fuel. With complete transfer of irradiated fuel to the ISFSI, Spent Fuel Pool Storage LCO's cease to exist and Section 1.0 is no longer required.

The definition for CERTIFIED FUEL HANDLER has been deleted. With completion of spent fuel transfer to the ISFSI, the handling of spent fuel will no longer be required and the Certified Fuel Handler position is also no longer required.

Minor modification to Technical Specification, Section 2.0, SAFETY LIMITS.

Replace the words "defueled facilities" with "a completely defueled plant". Note: Throughout this document and associated Attachments the term "plant" has been intentionally used to replace the various terms describing the plant, such as, unit and facility. As applied in this document and associated Attachments the term "plant" is used to denote the former nuclear plant and associated buildings, systems, and components used for the production of electric power and does not include the Independent Spent Fuel Storage Installation (ISFSI).

Deletion of Technical Specification, Section 3.0, LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY.

Deletion of this Section is proposed since the section does not apply to a completely defueled plant. As stated above, with complete transfer of irradiated fuel to the ISFSI, Spent Fuel Pool Storage LCO's cease to exist.

Deletion of Technical Specification, Section 3.1, DEFUELED SYSTEMS.

Deletion of this Section is proposed since the section only applied during the storage of spent fuel assemblies in the fuel storage pool or during movement of irradiated fuel assemblies in the fuel storage pool. With complete transfer of irradiated fuel to the ISFSI, spent fuel assemblies will no longer be located in the fuel storage pool.

Minor modification to Section 4.1.1 in the first paragraph.

Minor modification to the first paragraph of Section 4.1.1 to better described the location of the plant. The plant is located on Bailey Point and not on any areas surrounding Bailey Point. Therefore reference to the area surrounding Bailey Point is removed.

Deletion of exclusion area from Technical Specification, Section 4.1.1.

Deletion of the specified minimum distance to the boundary of the exclusion area from Technical Specification Design Features Section 4.1 and relocation of the requirement to the Defueled

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Safety Analysis Report. Relocating this design element to the Defueled Safety Analysis Report is consistent with 10 CFR 50.36 regarding the required contents of the design features section of Technical Specification.

10CFR50.36(c)(4) - states that design features to be included are those features of the facility such as materials of construction and geometric arrangement, which if altered or modified, would have a significant effect on safety and are not covered in categories (c)(1), (2) and (3) of this section.

Maine Yankee Technical Specification Section 4.0 describes the design features of the Maine Yankee plant. The design features of Section 4.1 "Site Description" include a simple description of the site and a specified minimum distance to the Exclusion Area Boundary. As described below, the required distance to the Exclusion Area Boundary is controlled by 10 CFR Part 100 and is demonstrated to be within appropriate regulatory limits by accident analysis. The specification of a minimum distance to the Exclusion Area Boundary does not qualify as a Technical Specification design feature under 10CFR50.36(c)(4) since the distance is already controlled by 10 CFR Part 100 and can not, by regulation, be altered or modified in a manner which would have a significant effect on safety. Therefore, in accordance with 10 CFR 50.36(c)(4), this specification should be relocated to the DSAR.

Deletion of Technical Specification, Section 4.2.

Deletion of Section 4.2 and associated Figure 4.1-1, Spent Fuel Pool Assembly Placement Limitations is being proposed since this section only applied during the storage of irradiated fuel in the fuel storage pool. Fuel storage design features that apply to irradiated fuel stored in dry casks at the ISFSI are contained in the NAC UMS Technical Specification.

Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.1, Responsibility.

Maine Yankee is proposing relocating this administrative control (with changes) to the Maine Yankee Quality Assurance Program.

Relocating organizational responsibilities from the technical specifications to the Quality Assurance Program is consistent with the criteria contained in 10 CFR 50.36 and the guidance in NRC Administrative Letter 95-06.

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The following paragraph is being entered to the Organization Section of the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.1.1:

The ISFSI Manager shall be responsible for overall site operation and shall delegate in writing the succession of this responsibility during his absence. The ISFSI Manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that are important to safety as defined in 10 CFR 72.3.

The above paragraph is being revised to address site organization after completion of spent fuel transfer to ISFSI. Responsibilities of ISFSI Manager are similar to previous responsibilities assigned to the replaced Plant Manager. The words "that affect the safe storage of irradiated fuel" have been replaced with "that are important to safety as defined in 10 CFR 72.3" since all irradiated fuel will now be stored at the ISFSI governed by Part 72.

The following paragraph is being entered to the Organization Section of the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.1.2:

The ISFSI Shift Lead reports to the ISFSI Manager and is the senior person onsite during backshifts and weekends.

Changes to the above paragraph are being made to address site organization after completion of spent fuel transfer to ISFSI. The Shift Lead responsibilities will change from control room command functions to senior oversight functions, after the transfer of spent fuel to the ISFSI is completed. Control Room command of the controls for rapid response to abnormal or accident situation will cease to exist at the plant after the removal of the spent fuel.

Section 5.1.3 has been deleted. ANSI Standard N18.1 provides job titles and organizational structures for organizations operating power reactors. The Standard does not apply to an ISFSI facility.

Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.2, Organization.

Maine Yankee is proposing relocating this administrative control (with changes) to the Maine Yankee Quality Assurance Program.

Relocating organizational responsibilities from the technical specifications to the Quality Assurance Program is consistent with the criteria contained in 10 CFR 50.36 and the guidance in NRC Administrative Letter 95-06.

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The following paragraphs are being entered to the Organization Section of the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.2.1:

Site organizations shall be established for the ISFSI operation and support management, respectively. The organizations shall include the positions for activities affecting the safe storage of irradiated fuel.

- a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation;*
- b. The ISFSI Manager shall have control over those onsite activities necessary for maintenance and storage of irradiated fuel in a safe condition;*
- c. The President shall have corporate responsibility for overall site nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to ensure the safe storage of irradiated fuel; and*
- d. The individuals who carry out radiation protection functions or perform quality assurance functions may report to the appropriate line manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.*

Changes to the above paragraphs were made to replace “unit organization” with “site organization”, “unit operation” with “ISFSI operation”, “Plant Manager” with “ISFSI Manager”, and “plant nuclear safety” with “site nuclear safety”. These changes focus organization requirements to the irradiated fuel storage area (ISFSI) after completion of fuel transfer. The revised text for paragraph (a) removes reference to the FSAR or QA program since it will be self-evident that the requirements are located in the QA program. Reference to CERTIFIED FUEL HANDLER in paragraph (d) has been deleted. With completion of spent fuel transfer to the ISFSI, the handling of spent fuel will no longer be required and the Certified Fuel Handler position is also no longer required.

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The following paragraphs are being entered to the Organization Section of the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.2.2:

The ISFSI staff organization shall include the following:

- a. *Each on duty shift shall be composed of at least one ISFSI Shift Lead person. Any unexpected absence of the on-duty Shift Lead shall be restored within 2 hours. This does not permit the Shift Lead position to be unmanned upon shift change due to an oncoming Shift Lead being late or absent.*
- b. *Administrative procedures shall be developed and implemented to limit the working hours of the ISFSI staff who perform functions that are important to safety.*

Adequate shift coverage shall be maintained without routine heavy use of overtime. The baseline for determining overtime use will be a 40 hour week. However, in the event that unforeseen problems require substantial amounts of overtime to be used; or during major maintenance or modifications the following guidelines shall be followed on a temporary basis:

1. *An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;*
2. *An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;*
3. *A break of at least 8 hours should be allowed between work periods, including shift turnover time;*
4. *The use of overtime should be considered on an individual basis and not for the entire staff on a shift.*

Any deviation from the above guidelines shall be authorized in advance by the ISFSI Manager or his designee, in accordance with established procedures and with documentation of the basis for granting the deviation. Routine deviation from the above guidelines is not authorized.

Changes to the above paragraphs were made to replace "unit staff" with "ISFSI staff", "facility" with "ISFSI", "Plant Manager" with "ISFSI Manager", and "Shift Manager" with "Shift Lead". These changes focus staff requirements to the irradiated fuel storage area (ISFSI) after

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completion of fuel transfer. After completion of irradiated fuel transfer to the ISFSI, the ISFSI staff organization will comprise of at least one ISFSI Shift Lead and paragraph (a) has been revised accordingly. In addition, paragraph (a) allows for an unexpected 2 hour absence of the Shift Lead. The 2 hour absence is acceptable since with all the spent fuel assemblies in ISFSI storage no actions will be required during this period to avoid departure from license conditions or technical specifications. Paragraph (b), (d), and (e) of the Technical Specification have been deleted since with completion of spent fuel transfer to the ISFSI, the handling of spent fuel will no longer be required and the Certified Fuel Handler position is no longer required. In addition, with respect to 10 CFR 50.54x, with all spent fuel assemblies in ISFSI storage no actions will be required that depart from license conditions or technical specifications. Paragraph (c) of the Technical Specification has been revised to require administrative procedures to limit working hours to staff personnel who perform functions important to safety at the ISFSI.

Since the staff for the defueled plant is no longer involved in functions that are safety related, minimum shift crew composition are no longer required in the plant Technical Specifications and therefore table 5.2.2.1 is being deleted.

Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.3, Unit Staff Qualifications.

Maine Yankee is proposing to relocate this administrative control (with changes) to the Maine Yankee Quality Assurance Program.

Relocating organizational responsibilities from the technical specifications to the Quality Assurance Program is consistent with the criteria contained in 10 CFR 50.36 and the guidance in NRC Administrative Letter 95-06.

The following paragraph is being entered to Section II of the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.3.1:

Each member of the ISFSI Staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8 – September 1975 for comparable positions.

The above paragraph was changed to remove the phrase “unless otherwise noted in the Technical Specifications” since the Technical Specifications will no longer provide the administrative controls for staff qualification requirements. The words “unit staff” has been replaced with “ISFSI staff” since important to safety functions now exist at the ISFSI only.

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Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.4, Training.

Technical Specification, Section 5.4.1 has been deleted since with completion of spent fuel transfer to the ISFSI, the handling of spent fuel will no longer be required and the Certified Fuel Handler training and retraining program is no longer required.

Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.5, Procedures.

Maine Yankee is proposing relocating this administrative control (with changes) to the Maine Yankee Quality Assurance Program.

Relocating organizational responsibilities from the technical specifications to the Quality Assurance Program is consistent with the criteria contained in 10 CFR 50.36 and the guidance in NRC Administrative Letter 95-06.

The following paragraph is being entered to Section V of the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.5.1:

Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The procedures applicable to the safe storage of irradiated fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;*
- b. Emergency Plan implementation;*
- c. Quality assurance for radiological effluent and environmental monitoring;*
- d. Fire Protection Program implementation; and*
- e. Radiation Protection, Offsite Dose Calculation Manual, and Radioactive Effluent Programs.*

Technical Specification 5.5.1 (c) "Cold Weather Operations" has been deleted in its entirety since with completion of transfer of irradiated fuel to the ISFSI, the plant no longer contains any safety related systems, structures, or components. Therefore cold weather operation is not required for safe nuclear operation. Spent fuel stored at the ISFSI is stored in sealed canisters with a dry and inert environment and, as such, do not require cold weather operation or

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rotection. Replaced subparagraph (f) of the Technical Specifications with new subparagraph (e) specifically identify the remaining programs subject to written procedures. As stated later in Technical Specification 5.6, the Technical Specification (TS) Bases Control Program has been deleted since with the complete removal of spent fuel assemblies from the spent fuel pool the pool will no longer contain any LCO's associated with the plant, therefore statements of bases Control Program Technical Specification LCO's will not be required. The Spent Fuel Pool Water Chemistry Program has been deleted since irradiated fuel will no longer be stored in the Fuel Storage Pool. In addition, the Inservice Testing Program also has been deleted in its entirety since the plant no longer will contain any safety related ASME Code Class 1, 2, or 3 components, including any associated supports. Therefore written procedures associated with these programs are no longer required and are not included in the above paragraph (e).

Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.6, Programs and Manuals.

Maine Yankee is proposing relocating this administrative control (with changes) to the Maine Yankee Quality Assurance Program.

Relocating organizational responsibilities from the technical specifications to the Quality Assurance Program is consistent with the criteria contained in 10 CFR 50.36 and the guidance in NRC Administrative Letter 95-06.

The following paragraph is being entered to the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.6.1:

Radiation Protection Program:

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

There are no changes proposed to the above paragraph from Technical Specification 5.6.1.

The following paragraph is being entered to the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.6.2:

Offsite Dose Calculation Manual (ODCM):

- a. *The ODCM shall contain the methodology and parameters used in the calculation of off-site doses resulting from radioactive gaseous and liquid effluents, in the calculation of*

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gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

- b. The ODCM shall also contain the radioactive effluent controls and the radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by the ODCM.*

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:*
- 1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);*
 - 2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, and 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;*
- b. Shall become effective after approval by the ISFSI Manager or designee; and*
- c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.*

The above paragraph (b) under ODCM was changed to replace the reference to Specification 5.7.2 and Specification 5.7.3 with reference to ODCM instead since Specification Sections 5.7.2 and 5.7.3 will be deleted. Under the heading "Licensing initiated changes to the ODCM", changed subparagraph (b) from "Plant Manager" to "ISFSI Manager" since the Plant Manager position will cease to exist and approval of the ODCM will be assigned to the ISFSI Manager.

The MY ISFSI does not create any radioactive effluents or have any radioactive waste treatment systems. Therefore, specific operating procedures for control of radioactive effluents in accordance with 10 CFR 72.44(d) are not required. NAC-UMS Technical Specification, Section A.3.1.5 CANISTER Helium Leak Rate, provides assurance that there are essentially no

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measurable radioactive effluents from the ISFSI. As such, an ODCM and environmental monitoring program for the ISFSI are not required.

The following paragraph is being entered to the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.6.3:

Radioactive Effluent Controls Program:

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;*
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in 10 CFR 20, Appendix B; Table 2, Column 2;*
- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;*
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the plant to unrestricted areas, conforming to 10 CFR 50, Appendix I;*
- e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;*
- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the estimated doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;*

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- g. *Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:*
- (1) *For noble gases: Less than or equal to dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and*
 - (2) *For Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to a dose rate of 1500 mrems/yr to any organ.*
- h. *Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;*
- i. *Limitations on the annual and quarterly doses to a member of the public from tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and*
- j. *Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.*

The only change proposed to the above paragraph from Technical Specification, Section 5.6.3 is to replace the term "unit" in paragraphs (d), (h), and (i) with "plant" to provide consistency in use of terms as described in Section 2.0 of this Attachment.

The MY ISFSI does not create any radioactive effluents or have any radioactive waste treatment systems. Therefore, specific operating procedures for control of radioactive effluents in accordance with 10 CFR 72.44(d) are not required. NAC-UMS Technical Specification, Section A.3.1.5 CANISTER Helium Leak Rate, provides assurance that there are essentially no measurable radioactive effluents from the ISFSI. As such, a Radioactive Effluent Controls Program for the ISFSI is not required.

Section 5.6.4, Technical Specifications (TS) Bases Control Program, has been deleted in its entirety since with the removal of spent fuel assemblies from the spent fuel pool the plant will no longer contain any LCO's and therefore statements of bases for Technical Specification LCO's will no longer be required.

Section 5.6.5, Spent Fuel Pool Water Chemistry Program, has been deleted in its entirety since irradiated fuel is no longer in the Fuel Storage Pool.

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Section 5.6.6, Inservice Testing Program, has been deleted in its entirety since with completion of transfer of irradiated fuel to the ISFSI, the plant no longer contains any safety related ASME Code Class 1,2, or 3 components, including associated supports.

Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.7, Reporting Requirements.

Maine Yankee is proposing relocating this administrative control (without changes) to the Maine Yankee Quality Assurance Program.

Relocating organizational responsibilities from the technical specifications to the Quality Assurance Program is consistent with the criteria contained in 10 CFR 50.36 and the guidance in NRC Administrative Letter 95-06.

The following paragraph is being entered to the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.7.1:

Occupational Radiation Exposure Report:

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem/yr and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., fuel handling, surveillance, routine maintenance, special maintenance [describe maintenance] and waste processing). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

There are no changes proposed to the above paragraph from Section 5.7.1.

The following paragraphs are being entered to the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.7.2:

Annual Radiological Environmental Operating Report:

The Annual Radiological Environmental Operating Report covering the plant activities during the previous calendar year shall be submitted by May 15 of each year. The report shall include

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summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

The only change proposed to the above paragraph from Technical Specification, Section 5.7.2 is to replace the term “unit” in the first paragraph with “plant” to provide consistency in use of terms as described in Section 2.0 of this Attachment.

NAC-UMS Technical Specification, Section A.3.1.5 CANISTER Helium Leak Rate, provides assurance that there are essentially no measurable radioactive effluents from the ISFSI. As such, a Radiological Environmental Operating Report is not required for the ISFSI.

The following paragraph is being entered to the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.7.3:

Radioactive Effluent Release Report:

The Radioactive Effluent Release Report covering the activities of the plant in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

The only change proposed to the above paragraph from Technical Specification, Section 5.7.3 is to replace the terms “unit” in the paragraph with “plant” to provide consistency in use of terms as described in Section 2.0 of this Attachment.

NAC-UMS Technical Specification, Section A.3.1.5 CANISTER Helium Leak Rate, provides assurance that there are essentially no measurable radioactive effluents from the ISFSI. As such, an annual report of liquid or gaseous releases from the ISFSI is not required.

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Deletion of Technical Specification, ADMINISTRATIVE CONTROLS, Sections 5.8, High Radiation Area.

Maine Yankee is proposing relocating this administrative control to the Maine Yankee Quality Assurance Program.

Relocating organizational responsibilities from the technical specifications to the Quality Assurance Program is consistent with the criteria contained in 10 CFR 50.36 and the guidance in NRC Administrative Letter 95-06.

The following paragraphs are being entered to the Maine Yankee Quality Assurance Program in place of the existing Technical Specification Section 5.8.1, 5.8.2, and 5.8.3:

High Radiation Area:

Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures or personnel continuously

escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates < 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.*
- b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.*
- c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities*

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within the area and shall perform periodic radiation surveillance at the frequency specified by Radiation Protection in the RWP.

In addition to the above requirements, each high radiation area, as defined in 10CFR 20, with radiation levels > 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Lead on duty or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas, as defined in 10 CFR 20, with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

The only change proposed is to replace the term "Shift Manager" in third paragraph with "Shift Lead. With spent fuel removed from the plant, the ISFSI Shift Lead or radiation protection supervisor will now be responsible for providing administrative control of any keys to high radiation areas. There are no additional changes proposed to the above paragraphs from Technical Specifications Sections 5.8.1, 5.8.2, and 5.8.3.

Deletion of Technical Specification, Bases Section.

The current Technical Specification Bases provides bases or reasons for current Limiting Conditions for Operations (LCO's). The proposed Technical Specifications after spent fuel transfer to the ISFSI will no longer contain any LCO's associated with the plant, therefore statements of bases for Technical Specification LCO's will not be required.

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SIGNIFICANT HAZARDS CONSIDERATION

Maine Yankee has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed License changes delete License information that does not apply to a plant that has permanently ceased operation. These changes are in compliance with 10 CFR Part 50 regulations and are not associated with the probability or consequences of accidents previously evaluated.

The proposed Technical Specification changes reflect the complete transfer of all spent nuclear fuel from the Spent Fuel Pool (SFP) to the Independent Spent Fuel Storage Installation (ISFSI). Design basis accidents related to the Spent Fuel Pool are discussed in the MY Defueled Safety Analysis Report (DSAR). These postulated accidents are predicated on spent nuclear fuel being stored in the Spent Fuel Pool. With the removal of the spent fuel from the Spent Fuel Pool, there are no remaining safety related systems required to be monitored and there are no remaining credible design basis accidents related to the SFP.

The proposed relocation of the specified minimum distance to the Exclusion Area Boundary from the Technical Specification to the DSAR has no impact on the probability or consequences of the remaining applicable design basis accidents.

The proposed changes do not affect design functions of structures, systems or components (SSC's) associated with the safe storage of fuel or radioactive material. Nor do any of these changes increase the likelihood of the malfunction of an SSC. The proposed changes do not affect operating procedures or administrative controls that have the function of preventing or mitigating any design basis accidents.

The MY DSAR provides a discussion of radiological events postulated to occur as a result of decommissioning with the bounding consequence resulting from a materials handling event. The proposed changes do not have an adverse impact on decommissioning activities or any of their postulated consequences.

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In addition, the proposed Technical Specification changes are consistent with the guidance provided in NRC Administrative Letter 95-06. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed License changes delete License information that does not apply to a plant that has permanently ceased operation. These changes are in compliance with 10 CFR Part 50 regulations and are not associated with any accidents previously evaluated.

These proposed Technical Specification changes relocate requirements from the Technical Specifications to the Defueled Safety Analysis Report, eliminate Technical Specifications associated with the storage of spent fuel in the SFP, and relocate Technical Administrative Controls to the MY Quality Assurance Program. With the complete removal of spent fuel assemblies from the plant there are no safety related SSC's that remain at the plant. Thus, these proposed changes will not have any affect on the operation or design function of safety related SSC's. These changes do not create new component failure mechanisms, malfunctions or accident initiators. Therefore, these proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed License changes delete License information that does not apply to a plant that has permanently ceased operation. These changes are in compliance with 10 CFR Part 50 regulations and do not involve a reduction in a margin of safety.

The design basis and accident assumptions within the MY DSAR and the Defueled Technical Specifications relating to spent fuel are no longer applicable. The proposed Technical Specification changes do not affect remaining plant operations, systems, or components supporting decommissioning activities. In addition, the proposed changes do not result in a change in initial conditions, system response time, or in any other parameter affecting the course of a decommissioning activity accident analysis.

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The relocation of the specified minimum distance to the Exclusion Area Boundary from the Technical Specifications to the Defueled Safety Analysis Report is consistent with the criterion set forth in 10 CFR 50.36 (c)(4). This criterion states that design features to be included in the Technical Specifications are those features of the facility such as materials of construction and geometric arrangement, which if altered or modified, would have a significant effect on safety and are not covered in other Technical Specification categories. The minimum distance to the Exclusion Area Boundary is established to maintain compliance within the limits specified in 10 CFR Part 100. The relocation of the specified minimum distance to the Exclusion Area Boundary to the DSAR continues to provide the safety analysis controls to assure compliance with 10 CFR Part 100 regulation.

Therefore, the proposed changes will not involve a significant reduction in the margin of safety.

Based on the above, Maine Yankee concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

In conclusion, based on the considerations discussed above, (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.

ENVIRONMENTAL CONSIDERATION

This amendment request meets the criteria specified in 10 CFR 51.22(c)(9) for categorical exclusion or otherwise not requiring environmental review. Specific criteria contained in this section of the regulations are discussed below:

1. The above amendment involves no significant hazards consideration. As demonstrated above, this requested amendment does not involve any significant hazards considerations.
2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The amendment removes operational and administrative requirements for systems that are no longer functionally required to support the safe storage of spent nuclear fuel within the Spent Fuel Pool. In addition, Technical Specification administrative controls are being relocated to the Quality Assurance Program. These changes are administrative in

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nature and do not affect any systems such that there may be an increase or change in type of effluents discharged offsite.

3. The elimination of non-applicable operational and administrative requirements from the Technical Specifications will not result in a significant increase in individual or cumulative occupational radiation exposure.

Based on the foregoing, it is concluded that the proposed amendment meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9) and therefore, no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

ATTACHMENT II
PROPOSED CHANGE PAGES
TO LICENSE

2.B.(7) This amended license is subject to the following conditions for protection of the environment:

(a) Deleted.

(b) Deleted.

2.B.(8) ~~This amended license is effective as of the date of issuance and shall expire at midnight October 21, 2008.~~

2.B.(9) Lands Released from the Jurisdiction of Facility Operating License No. DPR-36

The lands described in the following correspondence have been released from the jurisdiction of Facility Operating License No. DPR-36. The NRC may require additional surveys and/or decontamination only if, based upon new information, it determines that the criteria of 10 CFR Part 20, Subpart E were not met and residual activity remaining at the site could result in a significant threat to public health and safety.

(a) MYAPC Letter to USNRC dated August 16, 2001 "Early Release of Backlands, Proposed Change No. 211 as supplemented and as approved in Amendment No. 167.

2.B.(10) License Termination

(i) The Maine Yankee License Termination Plan describes an acceptable approach for demonstrating compliance with the radiological criteria for unrestricted use, as defined by 10 CFR 20.1402, by meeting a site release criteria of 10 millirem TEDE per year over background (all pathways) and 4 millirem (as distinguishable from background) TEDE per year for groundwater sources of drinking water using appropriate dose modeling methods, pathways and parameters and acceptable final radiation survey methods.

The licensee shall implement and maintain in effect all the provisions of the approved License Termination Plan submitted on August 13, 2001, as supplemented and as approved in the SER dated February 28, 2003, subject to and as amended under the following stipulations:

The licensee may make changes to the License Termination Plan without prior approval, provided the proposed changes do not:

- (a) Require Commission approval pursuant to 10 CFR 50.59;
- (b) Violate the requirements of 10 CFR 50.82(a)(6);
- (c) Reduce the coverage requirements for scan measurements;

ISSUED
Date: 4.8.03

- AM.14
11.13.75
- (2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as supplemented and amended.
- AM.14
11.13.75
- (3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.
- AM.14
11.13.75
- (4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration, or associated with radioactive apparatus or components.
- AM.14
11.13.75
- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear material as may be produced by the operation of the facility.
- AM.14
11.13.75
- (6) This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Parts 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:
- AM.113
07.10.89
- (a) Maximum Power Level
~~The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts thermal.~~
- AM. 164
05.05.99
- (b) Technical Specifications
- AM. 169
07.27.03
- The Technical Specifications contained in Appendix A, as revised through Amendment 169, are hereby incorporated in the license. The licensee will maintain the facility in accordance with the Technical Specifications.
- (c) Deleted.

ISSUED DATE: 7-24-03

Amendment No. 156, 164, 165, 166, 167, 169

ATTACHMENT III
PROPOSED CHANGE PAGES
TO TECHNICAL SPECIFICATIONS

Technical Specifications

Maine Yankee Nuclear Plant

Specifications

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SPECIFICATIONS**

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1.0 USE AND APPLICATION

This section is not applicable to a completely defueled plant.

1.1 Definitions

----- NOTE -----

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Technical Specification 5.4.1.

~~1.0 USE AND APPLICATION~~

~~1.2 Logical Connectors~~

~~PURPOSE~~ ————— ~~The purpose of this section is to explain the meaning of logical connectors.~~

————— ~~Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that may appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.~~

~~BACKGROUND~~ ————— ~~Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.~~

————— ~~If logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.~~

(continued)

1.2 Logical Connectors (continued)

EXAMPLES The following examples illustrate the use of logical connectors:

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met:	A.1 Verify ... <u>AND</u> A.2 Restore ...	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed:

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met:	A.1 Trip ... <u>OR</u> A.2.1 Verify ... <u>—AND</u> A.2.2 Reduce ...	

This example represents a more complicated use of logical connectors. Required Actions A.1, and A.2 are alternative choices; only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Either of the Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND:

4.0 USE AND APPLICATION

4.3 Completion Times

PURPOSE — The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND — Limiting Conditions for Operation (LCO's) specify minimum requirements for ensuring the safe storage of irradiated fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION — The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.

EXAMPLES — The following examples illustrate the use of Completion Times with different types of Conditions:

(continued)

1.3 Completion Times

EXAMPLES — EXAMPLE 1.3-1
(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Action and associated Completion Time not met.	A.1 Verify ...	6 hours
	<u>AND</u> A.2 Restore ...	36 hours

Condition A has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition A is entered.

The Required Actions of Condition A are to perform the verification required by ACTION A.1 within 6 hours AND to perform the restoration required by ACTION A.2 within 36 hours. A total of 6 hours is allowed for performing ACTION A.1 and a total of 36 hours (not 42 hours) is allowed for performing ACTION A.2 from the time that Condition A was entered. If ACTION A.1 is completed within 3 hours, the time allowed for completing ACTION A.2 is the next 33 hours because the total time allowed for completing ACTION A.2 is 36 hours.

(continued)

1.3 Completion Times

EXAMPLES — **EXAMPLE 1.3-2**
~~(continued)~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO limit exceeded:	A.1 Verify ...	1-hour <u>AND</u> 8 hours thereafter
	<u>AND</u> A.2 Restore ...	72 hours

~~Required Action A.1 has two Completion Times. The 1-hour Completion Time begins at the time the Condition is entered and each "8 hours thereafter" interval begins upon performance of Required Action A.1. The 72-hour completion time for Required Action A.2 also begins at the time the Condition is entered and runs concurrently.~~

IMMEDIATE — When "Immediately" is used as a Completion Time, the
COMPLETION — Required Action should be pursued without delay and in a
TIME — controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE — The purpose of this section is to define the proper use and application of Frequency requirements:

DESCRIPTION — Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LGO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR:

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0; Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements:

EXAMPLES — The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LGO (LGO not shown) is when irradiated fuel is stored in the spent fuel pool:

(continued)

1.4 Frequency

EXAMPLES — **EXAMPLE 1.4-1**
 —(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify parameter is within limits.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in the specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified, then SR 3.0.3 becomes applicable.

(continued)

1.4 Frequency

EXAMPLES — **EXAMPLE 1.4-2**
 (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify parameter is within limits.	Within 24 hours prior to moving irradiated fuel <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time Performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. The use of "prior to" indicates that the surveillance must be performed once before the initiation of fuel handling activities. This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "prior to" performance in this example).

2.0 SAFETY LIMITS (SLs)

This section is not applicable to a completely defueled plant.

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

This section is not applicable to a completely defueled plant.

~~LCO 3.0.1 LCO's shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.~~

~~LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.~~

~~If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.~~

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 — ~~SRs shall be met during specified conditions in the Applicability for individual LCO's, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.~~

SR 3.0.2 — ~~The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.~~

SR 3.0.3 — ~~If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever, is less. This delay period is permitted to allow performance of the Surveillance:~~

~~If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon expiration of the delay period.~~

~~When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. The Completion Times of the Required Actions begin immediately upon failure to meet the Surveillance.~~

~~3.1 DEFUELED SYSTEMS~~

~~3.1.1 Fuel Storage Pool Water Level~~

~~LCO 3.1.1 The fuel storage pool water level shall be \geq 21 ft over the top of irradiated fuel assemblies seated in the storage racks.~~

~~APPLICABILITY: During movement of irradiated fuel assemblies in the fuel storage pool.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel Storage pool water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in the fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify the fuel storage pool water level is \geq 21 ft above the top of irradiated fuel assemblies seated in the storage racks.	24 hours

~~3.1 DEFUELED SYSTEMS~~

~~3.1.2 Fuel Storage Pool Boron Concentration~~

~~LCO 3.1.2 — The fuel storage pool boron concentration shall be
≥ 1000 ppm.~~

~~APPLICABILITY: — When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Fuel storage pool boron concentration not within limit.</p>	<p>A.1 Suspend movement of fuel assemblies in the fuel storage pool.</p> <p style="text-align: center;">AND</p> <p>A.2.1 Initiate action to restore fuel storage pool boron concentration to within limit.</p> <p style="text-align: center;">OR</p> <p>A.2.2 Initiate action to perform a fuel storage pool verification.</p>	<p>Immediately</p> <p>Immediately</p> <p>Immediately</p>

(Continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.1 — Verify the fuel storage pool boron concentration is within limit.	Prior to movement of fuel if not performed within the past 7 days. <u>AND</u> 7 days thereafter

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Description

The plant shall be located on property owned by Maine Yankee Atomic Power Company on and surrounding Bailey Point in the Town of Wiscasset, Lincoln County, Maine.

The minimum distance to the boundary of the exclusion area, as defined in 10CFR100.3, shall be 2000 feet.

4.0 DESIGN FEATURES

4.2 Fuel Storage

~~Applicability: These design features apply only when irradiated fuel is stored in the fuel storage pool:~~

4.2.1 Criticality

4.2.1.1 The spent fuel storage racks are designed and shall be maintained with:

- ~~a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent;~~
- ~~b. $k_{\text{eff}} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in the FSAR~~
- ~~c. A nominal 9.085 inch center to center distance between fuel assemblies placed in Region II racks;~~
- ~~d. A nominal 10.5 inch center to center distance between fuel assemblies placed in Region I racks;~~

4.2.2 Drainage

~~The fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 40 ft:~~

4.2.3 Capacity

- ~~a. No more than 2019 fuel assemblies shall be stored in the fuel storage pool. These assemblies will be stored in either Region I or Region II of the fuel storage pool in accordance with the limitations of Figure 4.1-1. Unirradiated fuel assemblies shall be stored in either the New Fuel Storage Area or Region I of the fuel storage pool. Consolidated fuel shall be stored in Region II only:~~
- ~~b. No more than 20 standard fuel assemblies may be in consolidated form. These are included in the 2019 fuel assemblies of specification a:~~

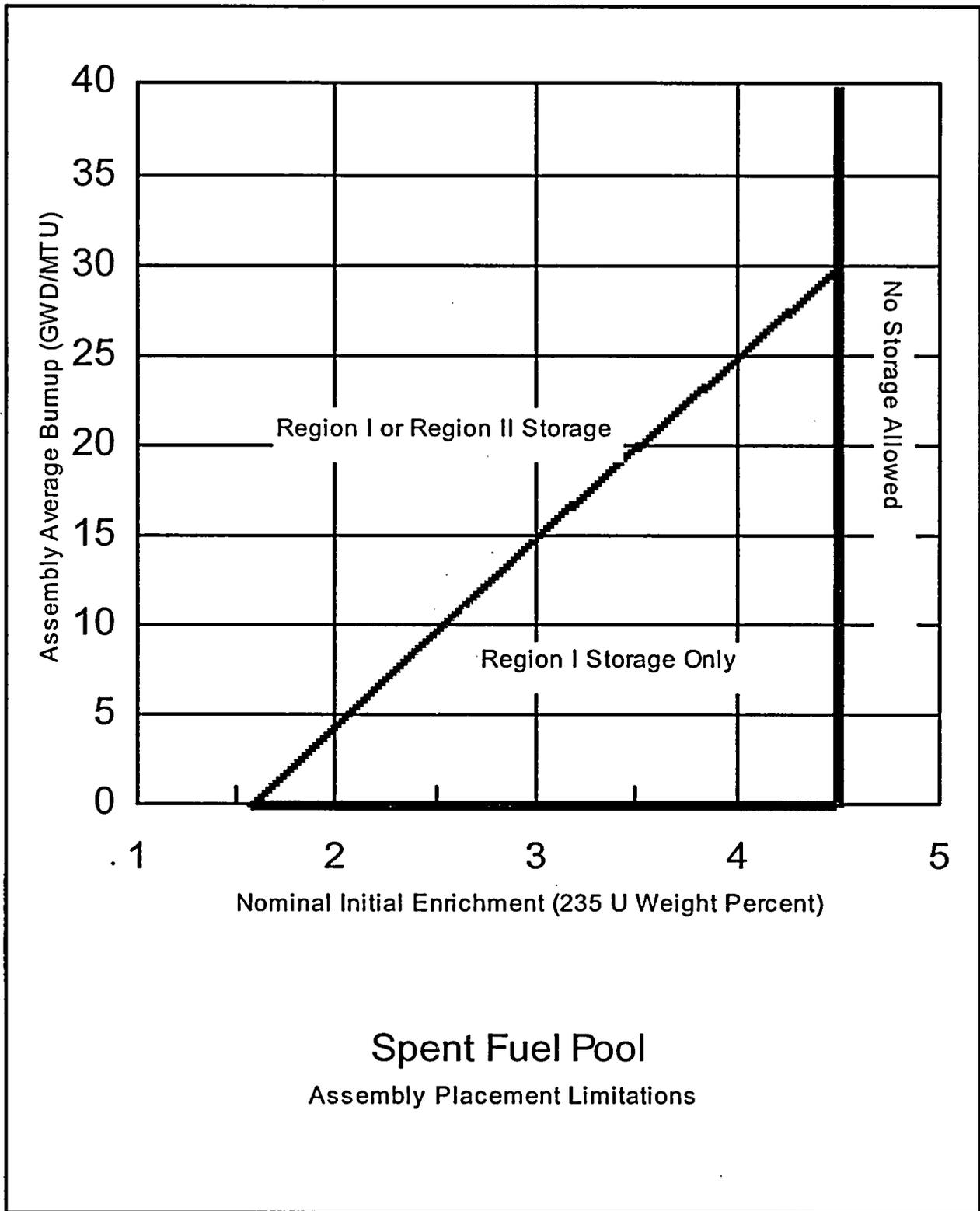


Figure 4.1-1 (DELETED)

5.0 ADMINISTRATIVE CONTROLS

Appropriate administrative controls are located in the Maine Yankee Quality Assurance Program.

~~5.1 Responsibility~~

~~5.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence:~~

~~The Plant Manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect the safe storage of irradiated fuel.~~

~~5.1.2 The Shift Manager on-site shall be responsible for the command function.~~

~~5.1.3 Unless otherwise defined, the technical specification titles for members of the staff are generic titles as provided in Regulatory Guide 1.8 - 1975 and/or ANSI N18.1 - 1971. Plant-specific, Maine Yankee titles for the functions and responsibilities associated with these generic titles are identified in the FSAR and/or QA program.~~

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 General Organizational Requirements

Unit organizations shall be established for the unit operation and support management, respectively. The organizations shall include the positions for activities affecting the safe storage of irradiated fuel.

- a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR or QA program;
 - b. The Plant Manager shall have overall responsibility for the unit and shall have control over those onsite activities necessary for maintenance and storage of irradiated fuel in a safe condition;
 - c. The President shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to ensure the safe storage of irradiated fuel; and
 - d. The individuals who train the CERTIFIED FUEL HANDLERS, carry out radiation protection functions, or perform quality assurance functions may report to the appropriate line manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.
-

5.2 Organization (continued)

5.2.2 Unit Staff

The facility staff organization shall include the following:

- a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.
- b. All fuel handling operations shall be directly supervised by a CERTIFIED FUEL HANDLER.
- c. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform functions important to the safe storage of irradiated fuel assemblies (e.g., CERTIFIED FUEL HANDLERS, non-certified operators, radiation protection personnel, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The baseline for determining overtime use will be a 40-hour week. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during major maintenance or modifications (including decommissioning activities) the following guidelines shall be followed on a temporary basis:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;
2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time;

(continued)

~~5.2 Organization~~

~~5.2.2 Unit Staff (continued)~~

- ~~3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;~~
- ~~4. The use of overtime should be considered on an individual basis and not for the entire staff on a shift.~~
- ~~Any deviation from the above guidelines shall be authorized in advance by the Plant Manager or his designee, in accordance with established procedures and with documentation of the basis for granting the deviation. Routine deviation from the above guidelines is not authorized.~~
- ~~d. The Shift Manager shall be a CERTIFIED FUEL HANDLER.~~
- ~~e. The Shift Managers shall report to an individual who is a CERTIFIED FUEL HANDLER.~~

**Table 5.2.2-1
Minimum Shift Crew Composition^(a)**

Position	Minimum Crew Number
Shift Manager	4
non-certified operator	4
Total	2

^(a) The shift crew composition may be one less than the minimum requirements of Table 5.2.2-1 for not more than 2 hours to accommodate unexpected absences of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crew member being late or absent.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8 - September 1975 for comparable positions unless otherwise noted in the Technical Specifications.

5.0 ADMINISTRATIVE CONTROLS

5.4 Training

5.4.1 A NRC-approved training and retraining program for the CERTIFIED FUEL HANDLERS shall be maintained under the direction of the Plant Manager or his designee:

5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures

5.5.1 ~~Written procedures shall be established, implemented, and maintained covering the following activities:~~

- ~~a. The procedures applicable to the safe storage of irradiated fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;~~
 - ~~b. Emergency plan implementation;~~
 - ~~c. Cold Weather Operations;~~
 - ~~d. Quality assurance for radiological effluent and environmental monitoring;~~
 - ~~e. Fire Protection Program implementation; and~~
 - ~~f. All programs specified in Specification 5.6:~~
-
-

~~5.0 ADMINISTRATIVE CONTROLS~~

~~5.6 Programs and Manuals~~

~~The following programs shall be established, implemented, and maintained:~~

~~5.6.1 Radiation Protection Program~~

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.~~

~~5.6.2 Offsite Dose Calculation Manual (ODCM)~~

~~a. The ODCM shall contain the methodology and parameters used in the calculation of off-site doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and~~

~~b. The ODCM shall also contain the radioactive effluent controls and the radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specification 5.7.2 and Specification 5.7.3.~~

~~Licensee initiated changes to the ODCM:~~

~~a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:~~

~~1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s);~~

(continued)

5.6 Programs, and Manuals

5.6.2 Offsite Dose Calculation Manual (ODCM) (continued)

2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, and 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the Plant Manager or designee; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.6.3 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

(continued)

5.6 Programs, and Manuals

5.6.3 Radioactive Effluent Controls Program (continued)

- ~~b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;~~
- ~~c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODGM;~~
- ~~d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;~~
- ~~e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODGM at least every 31 days;~~
- ~~f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the estimated doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;~~
- ~~g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be limited to the following:
 - ~~(1) For noble gases: Less than or equal to dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and~~
 - ~~(2) For Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ.~~~~

~~(continued)~~

5.6 Programs, and Manuals

5.6.3 ~~Radioactive Effluent Controls Program (continued)~~

- ~~h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;~~
- ~~i. Limitations on the annual and quarterly doses to a member of the public from tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and~~
- ~~j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190;~~

(continued)

5.6 Programs, and Manuals

5.6.4 Technical Specifications (TS) Bases Control Program

~~This program provides a means for processing changes to the Bases of these Technical Specifications:~~

- ~~a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews:~~
- ~~b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 - ~~A change in the TS incorporated in the license; or~~
 - ~~A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59:~~~~
- ~~c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR:~~
- ~~d. Proposed changes that meet the criteria of 5.6.4.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e):~~

5.6.5 Spent Fuel Pool Water Chemistry Program

~~The spent Fuel Pool Water Chemistry Program shall be established, implemented, and maintained, whenever irradiated fuel is stored in the Fuel Storage Pool.~~

~~This program provides controls for monitoring spent fuel pool water chemistry to minimize the potential effects of corrosion which could affect the safe storage of irradiated fuel. The program shall include identification of critical variables and control points for these variables. The program shall also include sampling frequencies and define corrective actions to be taken for off control point chemistry conditions. The NRC will be notified prior to elimination or changes to the acceptance criteria for critical variables monitored.~~

(continued)

5.6 Programs, and Manuals

5.6.5 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:**

<u>ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;**
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and**
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.**
-
-

5.0 ADMINISTRATIVE CONTROLS

5.7 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4:

5.7.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem/yr and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., fuel handling, surveillance, routine maintenance, special maintenance [describe maintenance] and waste processing). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totaling $< 20\%$ of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report shall be submitted by April 30 of each year.

5.7.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the unit activities during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.7 Reporting Requirements

5.7.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.7.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the activities of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.8 High Radiation Area

~~5.8.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates < 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.~~

~~Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:~~

- ~~a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.~~
- ~~b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.~~
- ~~c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by Radiation Protection in the RWP.~~

(continued)

5.8 High Radiation Area (continued)

- 5.8.2 In addition to the requirements of Specification 5.8.1, each high radiation area, as defined in 10 CFR 20, with radiation levels > 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Manager on duty or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.
- 5.8.3 For individual high radiation areas, as defined in 10 CFR 20, with radiation levels of > 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.
-

Technical Specifications

Maine Yankee Nuclear Plant

Bases

This Section is not applicable to a completely defueled plant.

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B-3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

~~LCO's~~ ~~LCO 3.0.1 and 3.0.2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated:~~

~~LCO 3.0.1~~ ~~LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification):~~

~~LCO 3.0.2~~ ~~LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:~~

~~a.~~ ~~Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and~~

~~b.~~ ~~Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified:~~

~~Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications:~~

(continued)

BASES

~~LCO 3.0.2~~
~~(continued)~~ ~~The Completion Times of the Required Actions are also applicable when a specified condition in the Applicability is entered intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise the safe storage of irradiated fuel. Intentional entry into ACTIONS should not be made for convenience.~~

B-3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs — SR 3.0.1 through SR 3.0.3 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated:

SR 3.0.1 — SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LGO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LGO:

Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LGO are not applicable, unless otherwise specified:

SR 3.0.2 — SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., other ongoing Surveillance or maintenance activities):

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. Any exceptions to SR 3.0.2 are stated in the individual Specifications:

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified:

(continued)

BASES

~~SR 3.0.3~~ ~~SR 3.0.3 establishes the flexibility to defer declaring an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.~~

~~This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.~~

~~The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.~~

~~Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals.~~

~~(continue)~~

BASES

~~SR 3.0.3~~ If a Surveillance is not completed within the allowed delay period, then the
~~(continued)~~ variable is considered outside the specified limits and the Completion Times of the
Required Actions for the applicable LCO Conditions begin immediately upon
expiration of the delay period. If a Surveillance is failed within the delay period;
then the variable is outside the specified limits and the Completion Times of the
Required Actions for the applicable LCO Conditions begin immediately upon the
failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this
Specification, or within the Completion Time of the ACTIONS, restores compliance
with SR 3.0.4.

B-3.1 DEFUELED SYSTEMS

B-3.1.1 Fuel Storage Pool Water Level

BASES

-
- BACKGROUND** — The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level provides shielding during the movement of spent fuel and minimizes the general area dose.
-
- A general description of the fuel storage pool design and the fuel storage pool cooling and purification system is given in the SAR. The assumptions of the fuel handling accident are given in the SAR.
-
- APPLICABLE SAFETY ANALYSES** — The Maine Yankee facility is permanently shutdown and the time since the last operation of the reactor has allowed significant decay of fission products (especially the daughter products with short half-lives) contained in the spent fuel. The spent fuel pool water level is adequate to provide necessary shielding for the fuel handlers to minimize occupational dose during normal spent fuel handling activities.
-
- The accident analysis for the fuel handling accident, as described in the FSAR, assumes that the accident occurs one year after shutdown from operations and a decontamination factor for iodine of 75 (19 feet of water.) No credit is taken for ventilation air filtration or isolation. For the purpose of establishing an upper limit on the amount of fuel damage resulting from a fuel handling accident, it is assumed that the fuel assembly is dropped during handling. The number of ruptured fuel rods which would result depends on several variables including the kinetic energy at impact and fuel assembly orientation during impact. The analysis assumes that all rods in the dropped assembly fail upon impact. The resulting doses calculated for the fuel handling accident are below the NRC acceptance limit for this accident, which is 10% of 10 CFR Part 100 limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued) — The normal operating water level of the fuel storage pool is established at approximately 36.5 (44 foot elevation) feet above the floor of the pool. The Technical Specification value of 24 feet over the top of irradiated fuel assemblies seated in the storage racks is equivalent to 34.7 feet of water above the fuel storage pool floor (42.2 foot elevation). The top of the fuel assembly is defined as the bottom of the flow plate, which is the top of the fuel rods. In the case of a fuel assembly dropped to the bottom of the fuel storage pool and standing upright, the Technical Specification Water Level provides more than 19 feet of water for iodine decontamination. In the case of a fuel assembly dropped and lying horizontally on top of the spent fuel racks, the Technical Specification Water Level provides more than 19 feet of water above the top of the bundle.

— A reduction in margin of safety for the fuel handling accident occurs when the acceptance limit (10% of 10 CFR Part 100 limits) is no longer met. The margin that exists between the technical specification limit for the fuel storage pool water level and the fuel handling accident acceptance limit represents operation margin.

— Radiological shielding analysis has determined that the Technical Specification Water Level provides a sufficient amount of water above the top of a fuel assembly (bottom of the flow plate) raised to its maximum height to maintain the radiation dose rates less than 80 mrem/hr at the surface of the water and less than 50 mrem/hr at the fuel handling hoist platform or the walkway around the pool.

— A reduction in margin of safety for the radiological shielding analysis occurs when the acceptance limit is no longer met. The acceptance limit is defined as that combination of occupancy time and dose rate such that no station personnel receive in excess of 5 rem per year (10 CFR 20.1201).

(Continued)

BASES

LGO The fuel storage pool water level is required to be ≥ 21 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified 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. As such, it is the minimum level required for movement of irradiated fuel within the fuel storage pool.

APPLICABILITY This LGO applies whenever irradiated fuel assemblies are being moved in the spent fuel pool, since the potential for a release of fission products exists and increased water level shielding is needed.

ACTIONS A-1
When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes a spent fuel handling accident from occurring. This does not preclude movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS SR 3.1.1.1
This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident and to provide shielding to minimize the general area dose during the movement of irradiated fuel. The water level in the spent fuel pool must be checked periodically when moving fuel. The 24-hour Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by facility procedures and are acceptable, based on operating experience.

BASES

LGO — The fuel storage pool boron concentration is required to be ≥ 1000 ppm when the LGO applies. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of a potential criticality accident resulting from the misplacement of a fuel assembly in the fuel storage pool. This concentration of dissolved boron is the minimum required concentration when fuel assemblies are stored in the fuel storage pool and a pool verification has not been performed.

APPLICABILITY — This LGO applies whenever fuel assemblies are stored in the spent fuel pool and a fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool. This LGO does not apply following the verification since the verification would confirm that there are no misplaced fuel assemblies. This verification includes a demonstration that fuel assemblies are properly located in the correct fuel storage rack in accordance with Figure 4.1.1 and that there are no fuel assemblies outside of or on top of the fuel storage racks. With no further fuel assembly movements in progress, there is no potential for a misplaced or a dropped fuel assembly.

ACTIONS — A.1, A.2 and A.3

— When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude movement of a fuel assembly or load to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit. Alternately, beginning a verification of the fuel storage locations, to ensure proper locations of the fuel can be performed.

SURVEILLANCE REQUIREMENTS — SR 3.1.2.1

— This SR verifies that the concentration of boron in the fuel storage pool is within the required limit during the movement of fuel assemblies. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water which could result in a dilution of the boron concentration is expected to take place over a short period of time.

ATTACHMENT IV
PROPOSED UPDATED VERSION
OF REVISED TECHNICAL SPECIFICATIONS

Technical Specifications

Maine Yankee Nuclear Plant

Specifications

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SPECIFICATIONS**

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1.0 USE AND APPLICATION

This section is not applicable to a completely defueled plant.

SLs
2.0

2.0 SAFETY LIMITS (SLs)

This section is not applicable to a completely defueled plant.

LCO Applicability
3.0

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

This section is not applicable to a completely defueled plant.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Description

The plant shall be located on property owned by Maine Yankee Atomic Power Company on Bailey Point in the Town of Wiscasset, Lincoln County, Maine.

5.0 ADMINISTRATIVE CONTROLS

Appropriate administrative controls are located in the Maine Yankee
Quality Assurance Program.

Technical Specifications

Maine Yankee Nuclear Plant

Bases

This Section is not applicable to a completely defueled plant.