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

HOLTEC DECOMMISSIONING INTERNATIONAL, LLC AND

HOLTEC INDIAN POINT 3, LLC

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO RENEWED FACILITY LICENSE

Renewed License No. DPR-64

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

   A. The application for a renewed license filed by Entergy Nuclear Indian Point 3, LLC (ENIP3) and Entergy Nuclear Operations, Inc. (ENO) for Indian Point Nuclear Generating Unit No. 3 (IP3 at the Indian Point Energy Center (IPEC) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

   D. Holtec Indian Point 3, LLC (Holtec IP3) and HDI are financially and technically qualified to engage in the activities authorized by this amendment;

   E. Holtec IP3 and HDI have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements" of the Commission's regulations;

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

   G. The receipt, possession and use of source, byproduct and special nuclear material as authorized by this renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70 including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31;

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

I. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1); and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the facility, and that any changes made to the facility's current licensing basis in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

2. Accordingly, Renewed Facility License No. DPR-64 is hereby issued to Holtec IP3 and HDI to read as follows:

A. This renewed license applies to the Indian Point Nuclear Generating Unit No. 3, a pressurized water nuclear reactor and associated equipment (the facility), owned by Holtec IP3 and maintained by HDI. The facility is located in Westchester County, New York, on the east bank of the Hudson River in the Village of Buchanan, and is described in the "Defueled Safety Analysis Report" as supplemented and amended, and the Environmental Report, as amended.

B. Subject to the conditions and requirements incorporated herein, the Commission licenses:

(1) Pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) Holtec IP3 to possess and use, and (b) HDI to possess and use the facility at the designated location in Westchester County, New York, in accordance with the procedures and limitations set forth in this renewed license;

(2) HDI pursuant to the Act and 10 CFR Part 70, to possess, at any time, special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Defueled Safety Analysis Report, as supplemented and amended;

(3) HDI 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 that were used for reactor startup, sealed sources that were used for calibration of reactor instrumentation and are used in the calibration of radiation monitoring equipment, and that were used as fission detectors in amounts as required;

(4) HDI 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;

(5) HDI pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials that were produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and 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:

(1) Deleted per Amendment No. 270

(2) Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 271, are hereby incorporated in the renewed license. HDI shall maintain the facility in accordance with the Technical Specifications.

D. (DELETED)

E. (DELETED)

F. This renewed license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

G. HDI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and to the authority of 10 CFR 50.90 and CFR 50.54(p). The combined set of plans[1] for the Indian Point Energy Center, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: “Physical Security, Training and Qualification, and Safeguards Contingency Plan, Revision 0,” and was submitted by letter dated October 14, 2004, as supplemented by letter dated May 18, 2006.

[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

HDI shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The HDI CSP was approved by License Amendment No. 243, as supplemented by changes approved by License Amendment Nos. 254, 260, and 263.

HDI has been granted Commission authorization to use “stand alone preemption authority” under Section 161A of the Atomic Energy Act, 42 U.S.C. 2201a with respect to the weapons described in Section II supplemented with Section III of Attachment 1 to its application submitted by letter dated August 20, 2013, as supplemented by letters dated November 21, 2013, and July 24, 2014, and citing letters dated April 27, 2011, and January 4, 2012. HDI shall fully implement and maintain in effect the provisions of the Commission-approved authorization.

H. Deleted per Amendment No. 270

I. <u>DELETED</u>

J. <u>DELETED</u>

K. <u>DELETED</u>

L. <u>DELETED</u>

M. <u>DELETED</u>

N. <u>DELETED</u>

O. Deleted per Amendment No. 270

P. Deleted

Q. <u>DELETED</u>

R. <u>DELETED</u>

S. <u>DELETED</u>

T. <u>DELETED</u>

U. <u>DELETED</u>

V. <u>DELETED</u>

W. Deleted

X. Deleted

AA. Deleted per Amendment No. 270

AB. Deleted per Amendment No. 270

AC. Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

AD. Deleted per Amendment No. 270

AE. HDI may transfer IP3 spent fuel to the IP2 spent fuel pit subject to the conditions listed in Appendix C. HDI is further authorized to transfer IP3 spent fuel into NRC approved storage casks for onsite storage by HDI and Holtec IP3.

AF. License Renewal License Conditions

(1) The information in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d) and as revised during the license renewal application review process, and licensee commitments as listed in Appendix A of the "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Units 2 and 3," (SER) and supplements to the SER, are collectively the "License Renewal UFSAR Supplement." The UFSAR Supplement is henceforth part of the UFSAR, which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs, activities, and commitments described in the UFSAR Supplement, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59, "Changes, Tests, and Experiments," and otherwise complies with the requirements in that section.

(2) The License Renewal UFSAR Supplement, as defined in license condition AF(1) above, describes certain programs to be implemented and activities to be completed prior to the period of extended operation (PEO).

a. The licensee shall implement those new programs and enhancements to existing programs no later than the date specified in the License Renewal UFSAR Supplement.

b. The licensee shall complete those activities no later than the date specified in the License Renewal UFSAR Supplement.

3. This renewed license is effective as of the date of issuance, and until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ho K. Nieh, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A - Permanently Defueled Technical Specifications
Appendix B - Environmental Technical Specification Requirements
Appendix C - Inter-Unit Fuel Transfer Technical Specifications

Date of Issuance: September 17, 2018

APPENDIX A

TO

FACILITY LICENSE DPR-64

PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS AND BASES

FOR THE

INDIAN POINT 3 NUCLEAR GENERATING STATION UNIT NO. 3

WESTCHESTER COUNTY, NEW YORK

HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3)

AND HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)

DOCKET NO. 50-286

Date of Issuance:
April 15, 1976

TABLE OF CONTENTS

1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
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 (CFH)	A CERTIFIED FUEL HANDLER is an individual who complies with the provisions of the CERTIFIED FUEL HANDLER training and retraining program required by TS 5.3.2.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the qualification requirements of Specification 5.3.1, but is not a CERTIFIED FUEL HANDLER.

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 Required Actions and Surveillances. The only logical connectors that 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.

When logical connectors are used to state a Surveillance, only the first level of logic is used, and the logical connector is left justified with the statement of the Surveillance.

EXAMPLE

The following example illustrates the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met…	A.1 Verify… AND 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.

1.0 USE AND APPLICATION

1.3 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 (LCOs) specify minimum requirements for ensuring safe handling and storage of spent nuclear 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, providing the facility is in a specified condition stated in the Applicability of the LCO. 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.
EXAMPLE	The following example illustrates the use of Completion Times with different Required Actions.

1.3 Completion Time

EXAMPLE (continued)

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pit boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pit.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pit boron concentration to within limit.	Immediately

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 immediately suspend movement of fuel assemblies in the spent fuel pit and initiate action to restore spent fuel pit boron concentration within limit.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a 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 LCO. 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.

EXAMPLE

The following example illustrates the type of Frequency statement that appears in the Technical Specifications (TS).

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify level is within limits.	12 hours

Example 1.4-1 contains the type of SR 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 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 a specified condition in the Applicability of the LCO, then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the specified condition or the LCO is considered not met (in accordance with SR 3.0.1).

2.0 DELETED

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs 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 the specified conditions in the Applicability for individual LCOs, 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 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.
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 greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. 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. 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.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3.

3.7 SPENT FUEL PIT REQUIREMENTS

3.7.14 Spent Fuel Pit Water Level

LCO 3.7.14 The spent fuel pit water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pit water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in the spent fuel pit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.14.1	Verify the spent fuel pit water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 SPENT FUEL PIT REQUIREMENTS

3.7.15 Spent Fuel Pit Boron Concentration

LCO 3.7.15 The spent fuel pit boron concentration shall be ≥ 1000 ppm.

---NOTE---
During inter-unit transfer of fuel the spent fuel pit boron concentration must also meet Appendix C LCO 3.1.1, "Boron Concentration."

APPLICABILITY: When fuel assemblies are stored in the spent fuel pit and a spent fuel pit verification has not been performed since the last movement of fuel assemblies in the spent fuel pit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pit boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pit.	Immediately
	AND	
	A.2.1 Initiate action to restore spent fuel pit boron concentration to within limit.	Immediately
	OR	
	A.2.2 Initiate action to perform a spent fuel pit verification.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify the spent fuel pit boron concentration is within limit.	31 days

3.7 SPENT FUEL PIT REQUIREMENTS

3.7.16 Spent Fuel Assembly Storage

LCO 3.7.16 Fuel assemblies stored in the spent fuel pit shall be classified in accordance with Figure 3.7.16-1 based on initial enrichment and burnup; and,

Fuel assembly storage location within the spent fuel pit shall be restricted based on the Figure 3.7.16-1 classification as follows:

a. Fuel assemblies classified as Type 2 may be stored in any location in either Region 1 or Region 2;

b. Fuel assemblies classified as Type 1A, 1B or 1C shall be stored in Region 1;

c. Fuel assembly storage location within Region 1 shall be restricted as follows:

1. Type 1A assemblies may be stored anywhere in Region 1;

2. Type 1B assemblies may be stored anywhere in Region 1, except a Type 1B assembly shall not be stored face-adjacent to a Type 1C assembly;

3. Type 1C assemblies shall not be stored in Row 64 or in Column ZZ; and

4. Type 1C assemblies shall be stored in Region 1 locations where all face-adjacent locations are as follows:

   a) occupied by Type 2 or Type 1A assemblies, or

   b) occupied by non-fuel components, or

   c) empty.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move fuel to restore compliance with LCO 3.7.16.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify by administrative means the initial enrichment and burnup of each fuel assembly and that the storage location meets LCO 3.7.16 requirements.	Prior to storing the fuel assembly in the spent fuel pit

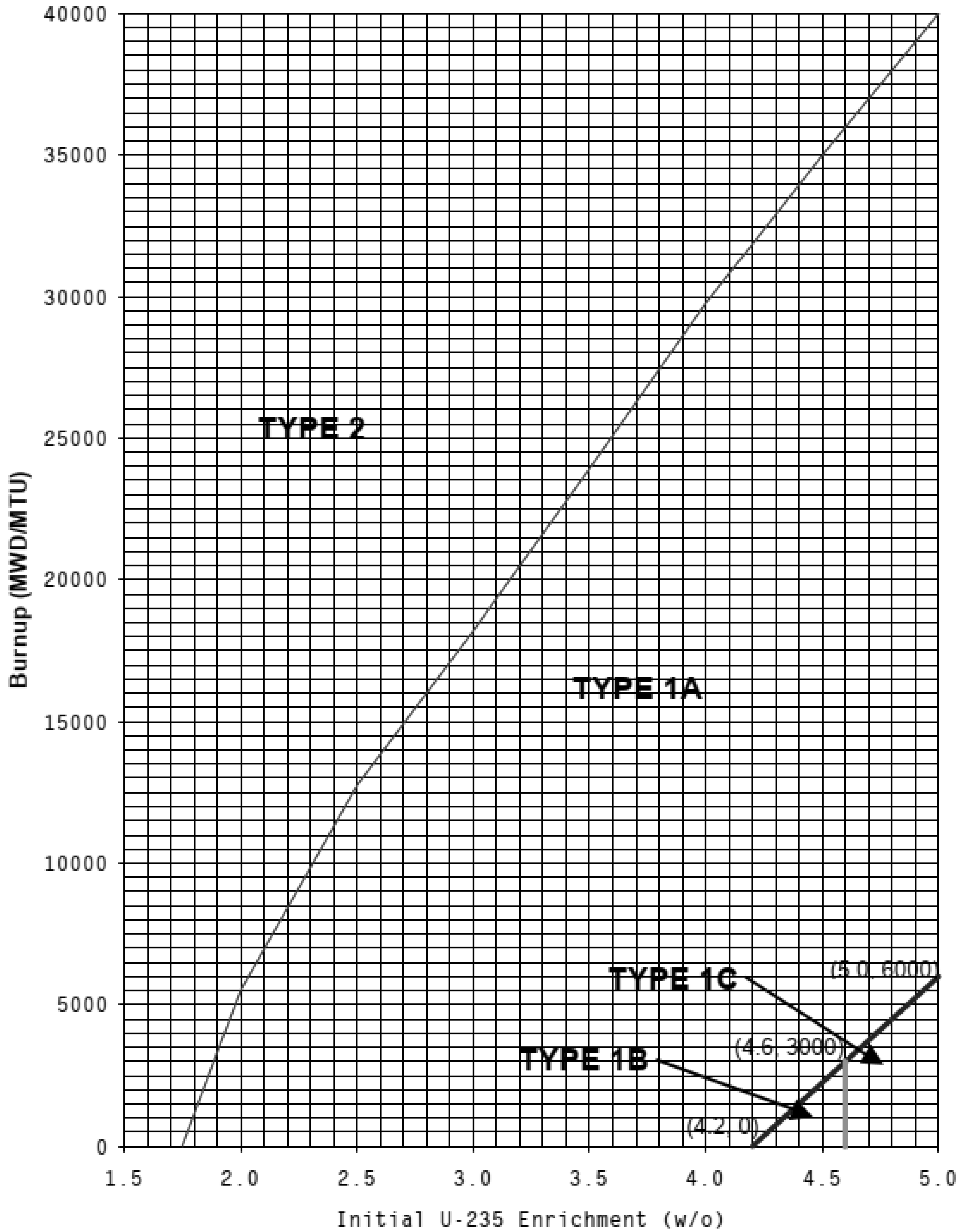

Figure 3.7.16-1 (Page 1 of 1)

Fuel Assembly Classification for Storage in the Spent Fuel Pit

4.0 DESIGN FEATURES

4.1 Site Location

Indian Point 3 is located on the east bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York. The site is approximately 24 miles north of the New York City boundary line. The nearest city is Peekskill which is 2.5 miles northeast of Indian Point.

The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone as defined in 10 CFR 100.3 is 350 meters and 1100 meters, respectively.

4.2 Deleted

4.3 Fuel Storage

4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. keff ≤ 0.95 if assemblies are inserted in accordance with Technical Specification 3.7.16, Spent Fuel Assembly Storage;

c. A nominal 9.075 inch center to center distance between fuel assemblies placed in the high density fuel storage racks (Region II);

d. A nominal 10.76 inch center to center distance between fuel assemblies placed in low density fuel storage racks (Region I).

4.3.2 Drainage

The spent fuel pit is designed and shall be maintained to prevent inadvertent draining of the pool below a nominal elevation of 88 ft.

4.3.3 Capacity

The spent fuel pit is designed and shall be maintained with a storage capacity limited to no more than 1345 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall facility 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 nuclear safety.

5.1.2 The shift manager (SM) shall be responsible for the shift command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility staff and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear fuel.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all decommissioning 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, including the facility-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the DSAR and Quality Assurance Plan, as appropriate;

b. The plant manager shall be responsible for overall safe maintenance of the facility and shall have control over those onsite activities necessary for safe storage and maintenance of nuclear fuel;

c. The corporate officer with direct responsibility for IP3 shall have corporate responsibility for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in maintaining and providing technical support to the facility to ensure safe management of nuclear fuel; and

d. The individuals who train the CERTIFIED FUEL HANDLERS, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2 Organization

5.2.2 Facility Staff

The facility staff organization shall include the following:

a. Each duty shift shall be composed of at least one shift manager and one NON-CERTIFIED OPERATOR. The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.

   At least one person qualified to stand watch in the control room (NON-CERTIFIED OPERATOR or CERTIFIED FUEL HANDLER) shall be present in the control room when nuclear fuel is stored in the spent fuel pool.

b. Shift crew composition may be less than the minimum requirement of 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements and all of the following conditions are met:

   1) No fuel movements are in progress;
   2) No movement of loads over fuel are in progress; and
   3) No unmanned shift positions during shift turnover shall be permitted while the shift crew is less than the minimum.

c. An individual qualified in radiation protection procedures shall be on site during fuel handling operations and during movement of heavy loads over the fuel storage racks. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Not Used.

e. The shift manager shall be a CERTIFIED FUEL HANDLER.

f. Deleted.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions with exceptions specified in the HDI Quality Assurance Program Manual (QAPM).

5.3.2 An NRC approved training and retraining program for CERTIFIED FUEL HANDLERS shall be maintained.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

a. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 except as provided in the quality assurance program described or referenced in the DSAR;

b. Deleted;

c. Quality assurance for effluent and environmental monitoring;

d. Deleted; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite 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 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 5.6.2 and Specification 5.6.3.

c. Licensee initiated changes to the ODCM:

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

(a) Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

(b) A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 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;

2. Shall become effective after the approval of the plant manager; and

3. 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.

5.5.2 Deleted

5.5 Programs and Manuals

5.5.3 Not Used

5.5.4 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 each unit/facility 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. Determination of projected dose contributions from radioactive effluents in accordance with the methodology 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 projected 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;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

   a. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

   b. For iodine-131, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit/facility 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 iodine-131, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit/facility 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, beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Controls Program surveillance frequency.

5.5.5 through 5.5.10 Deleted

5.5 Programs and Manuals

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, “Postulated Radioactive Release due to Waste Gas System Leak or Failure.” The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, “Postulated Radioactive Release due to Tank Failures.”

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank shall be limited to less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks’ contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks’ contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.12 Deleted

5.5 Programs and Manuals

5.5.13 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:

1. a change in the TS incorporated in the license; or

2. a change to the DSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the DSAR.

d. Proposed changes that do not meet the criteria of Specification 5.5.13.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).

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 Not Used

5.6.2 Annual Radiological Environmental Operating Report

---NOTE---
A single submittal may be made for a multiple unit/facility station. The submittal should combine sections common to all units/facilities at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit/facility 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.

A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.

5.6.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit/facility station. The submittal shall combine sections common to all units/facilities at the station; however, for units/facilities with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit/facility.

The Radioactive Effluent Release Report covering the operation of the unit/facility 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/facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR Part 50.36a and 10 CFR 50, Appendix I, Section IV.B.I.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following facility radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following facility radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u> (continued)

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

APPENDIX B

TO

FACILITY LICENSE

FOR

HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3) AND

HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)

INDIAN POINT 3 NUCLEAR

POWER PLANT

ENVIRONMENTAL TECHNICAL SPECIFICATION

REQUIREMENTS

PART I: NON-RADIOLOGICAL ENVIRONMENTAL PROTECTION PLAN

FACILITY LICENSE NO. DPR-64

DOCKET NUMBER 50-286

INDIAN POINT NUCLEAR GENERATING PLANT

UNIT 3

ENVIRONMENTAL TECHNICAL SPECIFICATION REQUIREMENTS

PART I: NON-RADIOLOGICAL ENVIRONMENTAL PROTECTION PLAN

TABLE OF CONTENTS

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during handling and storage of spent fuel and maintenance of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is maintained in an environmentally acceptable manner, as established by the FES and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of handling and storage of spent fuel and maintenance of the facility and of actions taken to control those effects.

Environmental concerns identified in the FES which relate to water quality matters are regulated by way of the licensee's SPDES permit.

2.0 Environmental Protection Issues

In the FES-OL for Unit 2 dated September 1972 and the FES-OL for Unit 3 dated February 1975, the staff considered the environmental impacts associated with operation of the Indian Point Nuclear Generating Plant. Certain environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment. The Appendix B Environmental Technical Specifications issued with the licenses included monitoring programs and other requirements to protect water quality and aquatic biota during plant operation with once-through cooling. As amended on July 11, 1979, the Appendix B ETS included monitoring and other requirements to address the following non-radiological aquatic protection issues:

(1) Controlled release of thermal discharges (ETS Sections 2.1, 3.1, 2.2.2, 3.2.2, and 4.1.1.a).

(2) Controlled release of non-radioactive chemical discharges (ETS Sections 2.3 and 3.3).

(3) Controlled intake flow velocity to limit impingement of organisms on intake structures (ETS Sections 2.2.1 and 3.2.1.

(4) Monitoring of aquatic biota in the Hudson River to evaluate effects of once-through operation (ETS Section 4.1.2).

Aquatic issues are now addressed by the effluent limitations, monitoring requirements and other conditions in or annexed to the effective SPDES permit issued by the Department of Environmental Conservation of the State of New York (DEC). The NRC will therefore rely on the DEC for regulation of matters involving water quality and aquatic biota and in the case of federally listed sturgeon, decisions made by the National Marine Fisheries Service (NMFS) under authority of the Endangered Species Act, for requirements pertaining to aquatic monitoring.

3.0 Consistency Requirements

3.1 Plant Design and Operation

HDI may make changes in facility design or operations or perform tests or experiments affecting the environment provided such changes, tests or experiments do not involve an unreviewed environmental question, and do not involve a change in the Environmental Protection Plan.* Changes in the facility design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this section.

Before engaging in additional construction or operational activities which may affect the environment, HDI shall prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity involves an unreviewed environmental question, HDI shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation. When such activity involves a change in the Environmental Protection Plan, such activity and change to the Environmental Protection Plan may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental statement (FES) or final supplemental environmental impact statement (FSEIS), as modified by the staff's testimony to the Atomic Safety and Licensing Boards, supplements to the FES or FSEIS, environmental impact appraisals, or in any decision of the Atomic Safety and Licensing Board;

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* This provision does not relieve the HDI of the requirements of 10 CFR 50.59.

or (2) a significant change in effluents or power level; or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

HDI shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include a written evaluation which provides a basis for the determination that the change, test, or experiment does not involve an unreviewed environmental question nor constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. HDI shall include as part of its Annual Environmental Protection Plan Report per Subsection 5.4.1: brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the NPDES Permits and State Certifications

Violations of the NPDES Permit or the State certification (pursuant to Section 4.1 of the Clean Water Act) shall be reported to the NRC by submittal of copies of the reports required by the NPDES Permit or certification.

Changes and additions to the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The NRC shall be notified of changes to the effective NPDES Permit proposed by Holtec IP3 and HDI by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The notification of a licensee-initiated change shall include a copy of the requested revision submitted to the permitting agency. HDI shall provide the NRC a copy of the application for renewal of the NPDES permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in facility design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, or local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to the handling and storage of spent fuel and maintenance of the facility shall be recorded and promptly reported to the NRC within 24 hours by telephone, telegraph, or facsimile transmissions followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, unusual mortality or occurrence of any species protected by the Endangered Species Act of 1973, unusual fish kills, unusual increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

In accordance with Section 7(a) of the Endangered Species Act, the National Marine Fisheries Service (NMFS) issued a Biological Opinion related to the continued operation of IP2 and IP3 that pertains to shortnose sturgeon (*Acipenser brevirostrum*) and Atlantic sturgeon (*Acipenser oxyrinchus oxyrinchus*). The Biological Opinion includes an Incidental Take Statement with Reasonable and Prudent Measures that the NMFS has determined to be necessary or appropriate to minimize the amount or extent of incidental take and associated Terms and Conditions, which are non-discretionary and implement the Reasonable and Prudent Measures. The currently applicable Biological Opinion concludes that continued operation of IP2 and IP3 is not likely to jeopardize the continued existence of the listed species or to adversely affect the designated critical habitat of those species. This Biological Opinion conservatively bounds the conditions that will occur in the permanently shut down and defueled condition.

Holtec Decommissioning International, LLC shall adhere to the requirements within the Incidental Take Statement of the currently applicable Biological Opinion. Changes to the Biological Opinion, including the Incidental Take Statement, Reasonable and Prudent Measures, and Terms and Conditions contained therein, must be preceded by consultation between the NRC, as the authorizing agency, and the NMFS.

5.0 Administrative Procedures

5.1 Review and Audit

HDI shall provide a review and audit of compliance with the Environmental Protection Plan. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure is utilized to achieve the independent review and audit function and results of the audits activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of previous plant operation and the handling and storage of spent fuel and maintenance of the facility shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to the NRC on request.

Records of modifications to facility structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the facility. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Requests for changes in the Environmental Protection Plan shall include an assessment of the environmental impacts of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the Environmental Protection Plan. This EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Protection Plan Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following issuance of the operating license. The period of the first report shall begin with the date of issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this Environmental Protection Plan for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous non-radiological environmental monitoring reports, and an assessment of the observed impacts of the previous plant operation and the handling and storage of spent fuel and maintenance of the facility on the environment. If harmful effects or evidence of trends towards irreversible damage to the environment are observed, HDI shall provide a detailed analysis of the data and a proposed course of action to alleviate the problem.

The Annual Environmental Protection Plan Report shall also include:

(a) A list of EPP noncompliances and the corrective actions taken to remedy them.

(b) A list of all changes in facility design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental issue.

(c) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

(d) A list of all reports submitted in accordance with the NPDES permit or the State certification.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and facility conditions, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State, or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the same time it is submitted to the other agency.

APPENDIX B

TO

FACILITY LICENSE

FOR

HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3) AND

HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)

INDIAN POINT 3 NUCLEAR

POWER PLANT

ENVIRONMENTAL TECHNICAL SPECIFICATION

REQUIREMENTS

PART II: RADIOLOGICAL ENVIRONMENTAL

FACILITY LICENSE NO. DPR-64

DOCKET NUMBER 50-286

DELETED BY AMENDMENT NO. 205

APPENDIX C

TO

FACILITY LICENSE

FOR

HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3) AND

HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART I: SPENT FUEL TRANSFER CANISTER AND TRANSFER CASK SYSTEM

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

SPENT FUEL SHIELDED TRANSFER CANISTER AND TRANSFER CASK SYSTEM

1.0 DESCRIPTION

The spent fuel transfer system consists of the following components: (1) a spent fuel shielded transfer canister (STC), which contains the fuel; (2) a transfer cask (HI-TRAC 100D) (hereafter referred to as HI-TRAC), which contains the STC during transfer operations; and (3) a bottom missile shield.

The STC and HI-TRAC are designed to transfer irradiated nuclear fuel assemblies from the Indian Point 3 (IP3) spent fuel pit to the Indian Point 2 (IP2) spent fuel pit. A fuel basket within the STC holds the fuel assemblies and provides criticality control. The shielded transfer canister provides the confinement boundary, water retention boundary, gamma radiation shielding, and heat rejection capability. The HI-TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The STC contains up to 12 fuel assemblies.

The STC is the confinement system for the fuel. It is a welded, multi-layer steel and lead cylinder with a welded base-plate and bolted lid. The inner shell of the canister forms an internal cylindrical cavity for housing the fuel basket. The outer surface of the canister inner shell is buttressed with lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are ¾ inch steel, 2 ¾ inch lead and ¾ inch steel, respectively. The canister closure incorporates two O-ring seals to ensure its confinement function. The confinement system consists of the canister inner shell, bottom plate, top flange, top lid, top lid O-ring seals, vent port seal and cover plate, and drain port seal and coverplate. The fuel basket, for the transfer of 12 Pressurized Water Reactor (PWR) fuel assemblies, is a fully welded, stainless steel, honeycomb structure with neutron absorber panels attached to the individual storage cell walls under stainless steel sheathing. The maximum gross weight of the fully loaded STC is 40 tons.

The HI-TRAC is a multi-layer steel and lead cylinder with a bolted bottom (or pool) and top lid. For the fuel transfer operation the HI-TRAC is fitted with a solid top lid, an STC centering assembly, and a bottom missile shield. The inner shell of the transfer cask forms an internal cylindrical cavity for housing the STC. The outer surface of the cask inner shell is buttressed with intermediate lead and steel shells for radiation shielding. The minimum thickness of the steel, lead and steel shells relied upon for shielding starting with the innermost shell are ¾ inch steel, 2 ⅞ inch lead and 1 inch steel, respectively. An outside shell called the "water jacket" contains water for neutron shielding, with a minimum thickness of 5". The HI-TRAC bottom and top lids incorporate a gasket seal design to ensure its water confinement function. The water confinement system consists of the HI-TRAC inner shell, bottom lid, top lid, top lid seal, bottom lid seal, vent port seal, vent port cap and bottom drain plug.

The HI-TRAC provides a water retention boundary, protection of the STC, gamma and neutron radiation shielding, and heat rejection capability. The bottom missile shield is attached to the bottom of the HI-TRAC and provides tornado missile protection of the pool lid bolted joint. The HI-TRAC can withstand a tornado missile in other areas without the need for additional shielding. The STC centering assembly provides STC position control within the HI-TRAC and also acts as an internal impact limiter in the event of a non-mechanistic tipover accident.

2.0 CONDITIONS

2.1 OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, maintenance, and recovery from off normal conditions such as crane hang-up. The written operating procedures shall be consistent with the technical basis described in Chapter 10 of the Licensing Report (Holtec International Report HI-2094289).

2.2 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 8 of the Licensing Report (Holtec International Report HI-2094289).

2.3 PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A training exercise of the loading, closure, handling/transfer, and unloading, of the equipment shall be conducted prior to the first transfer. The training exercise shall not be conducted with irradiated fuel. The training exercise may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The training exercise shall include, but is not limited to the following:

a) Moving the STC into the IP3 spent fuel pool.
b) Preparation of the HI-TRAC for STC loading.
c) Selection and verification of specific fuel assemblies and non-fuel hardware to ensure type conformance.
d) Loading specific assemblies and placing assemblies into the STC (using a single dummy fuel assembly), including appropriate independent verification.
e) Remote installation of the STC lid and removal of the STC from the spent fuel pool.
f) Placement of the STC into the HI-TRAC with the STC centering assembly.
g) STC closure, establishment of STC water level with steam, verification of STC water level, STC leakage testing, and operational steps required prior to transfer, as applicable.
h) Establishment and verification of HI-TRAC water level.
i) Installation of the HI-TRAC top lid.
j) HI-TRAC closure, leakage testing, and operational steps required prior to transfer, as applicable.
k) Movement of the HI-TRAC with STC from the IP3 fuel handling building to the IP2 fuel handling building along the haul route with designated devices.
l) Moving the STC into the IP2 spent fuel pool.
m) Manual crane operations for bare STC movements including demonstration of recovery from a crane hang-up with the STC suspended from the crane.

APPENDIX C

TO

FACILITY LICENSE

FOR

HOLTEC INDIAN POINT 3, LLC (HOLTEC IP3) AND

HOLTEC DECOMMISSIONING INTERNATIONAL, LLC (HDI)

INDIAN POINT NUCLEAR

GENERATING UNIT No. 3

INTER-UNIT FUEL TRANSFER TECHNICAL SPECIFICATIONS

PART II: TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. DPR-64

DOCKET NO. 50-286

TABLE OF CONTENTS

1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
INTACT FUEL ASSEMBLIES	INTACT FUEL ASSEMBLIES are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as INTACT FUEL ASSEMBLIES unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on an STC while it is being loaded with fuel assemblies and while the STC is being placed in the HI-TRAC. LOADING OPERATIONS begin when the first fuel assembly is placed in the STC and end when the HI-TRAC is suspended from or secured on the TRANSPORTER.
NON-FUEL HARDWARE (NFH)	NFH is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Neutron Source Assemblies (NSAs), Hafnium Flux Suppressors, and Instrument Tube Tie Rods (ITTRs).
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all licensed activities performed on a HI-TRAC loaded with one or more fuel assemblies when it is being moved after LOADING OPERATIONS or before UNLOADING OPERATIONS. TRANSFER OPERATIONS begin when the HI-TRAC is first suspended from or secured on the TRANSPORTER and end when the TRANSPORTER is at its destination and the HI-TRAC is no longer secured on or suspended from the TRANSPORTER.
TRANSPORTER	TRANSPORTER is the device or vehicle which moves the HI-TRAC. The TRANSPORTER can either support the HI-TRAC from underneath or the HI-TRAC can be suspended from it.

(continued)

1.1 Definitions (continued)

Term	Definition
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an STC or HI-TRAC while it is being unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the HI-TRAC is no longer suspended from or secured on the TRANSPORTER and end when the last fuel assembly is removed from the STC.
ZR	ZR means any zirconium-based fuel cladding authorized for use in a commercial nuclear power plant reactor.

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 appear in TS are <u>AND</u> and <u>OR</u>. 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 indentions of the logical connectors. When 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 . . . AND 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.

(continued)

1.2 Logical Connectors (continued)

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Remove . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 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. Any one of these three 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. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are altemative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 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 (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. 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 Times(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., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the Spent Fuel Shielded Transfer Canister and Transfer Cask System is in a specified condition stated in the Applicability of the LCO. 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 Spent Fuel Shielded Transfer Canister and Transfer Cask System is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1 AND B.2 Perform Action B.2	12 hours 36 hours

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

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit.	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1.	12 hours
	AND	
	B.2 Complete action B.2.	36 hours

When a system is determined not to meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

-----------------------------NOTE-----------------------------
Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1. AND B.2 Complete action B.2.	6 hours 12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a 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 Limiting Condition for Operation (LCO). 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. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

(continued)

1.4 Frequency (continued)

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	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 interval specified in the 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 the equipment or variables are 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 a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 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. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). 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 "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0 NOT USED

This section is intentionally left blank

3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during 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, except as provided in LCO 3.0.5. 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.
LCO 3.0.3	Not applicable.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an STC.
LCO 3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing.

3.1 INTER-UNIT FUEL TRANSFER

3.1.1 Boron Concentration

LCO 3.1.1 The boron concentration of the water in the Spent Fuel Pit and the STC shall be ≥ 2000 ppm.

APPLICABILITY: Whenever one or more fuel assemblies are in the STC.

-----------------------------------NOTE-----------------------------------
Only applicable to the spent fuel pit when the STC is in the spent fuel pit
----

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend LOADING OPERATIONS or UNLOADING OPERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------------------NOTE----------------------------- This surveillance is only required to be performed if the STC is submerged in water in the spent fuel pool or if water is added to, or recirculated through, the STC when the STC is in the HI-TRAC. Any added water must meet the boron concentration requirement of LCO 3.1.1. ---	Once, within 4 hours prior to entering the Applicability of this LCO. AND
SR 3.1.1.1 Verify the boron concentration is within limit using two separate measurements.	Once per 48 hours thereafter.

3.1 INTER-UNIT FUEL TRANSFER

3.1.2 Shielded Transfer Canister (STC) Loading

LCO 3.1.2 INTACT FUEL ASSEMBLIES placed into the Shielded Transfer Canister (STC) shall be classified in accordance with Table 3.1.2-1 based on initial enrichment and burnup and shall be restricted based on the following:

a. INTACT FUEL ASSEMBLIES classified as Type 2 may be placed in the STC basket (see Figure 3.1.2-1) with the following restrictions:

1. Post-irradiation cooling time, initial enrichment, and allowable average burnup shall be within the limits for the cell locations as specified in Table 3.1.2-3;
2. Decay heat including NON FUEL HARDWARE ≤ 1.2 kW (any cell);
3. Total STC Decay heat from all cell locations including NON FUEL HARDWARE ≤9.621 kW;
4. Post-irradiation cooling time and the maximum average burnup of NON FUEL HARDWARE shall be within the cell locations and limits specified in Table 3.1.2-2. In accordance with Table 3.1.2-2 RCCAs and Hafnium Flux Suppressors cannot be placed in locations 5, 6, 7, 8, 9, 10, 11, 12 of the STC basket.

- NOTE -

If one or more Type 1 fuel assemblies are in the STC, cells 1, 2, 3, AND 4 must be empty, with a cell blocker installed that prevents inserting fuel assemblies and/or NON-FUEL HARDWARE.

---

b. INTACT FUEL ASSEMBLIES classified as Type 1 or Type 2 may be placed in locations 5, 6, 7, 8, 9, 10, 11, 12 of the STC basket (see Figure 3.1.2-1) with the following restrictions:

1. Post-irradiation cooling time, initial enrichment, and allowable average burnup shall be within the limits for the cell locations as specified in Table 3.1.2-3;
2. Decay heat including NON FUEL HARDWARE ≤ 1.2 kW;
3. Post-irradiation cooling time and the maximum average burnup of NON FUEL HARDWARE shall be within the cell locations and limits specified in Table 3.1.2-2. In accordance with Table 3.1.2-2 RCCAs and Hafnium Flux Suppressors cannot be placed in locations 5, 6, 7, 8, 9, 10, 11, 12 of the STC basket.

c. Only INTACT FUEL ASSEMBLIES with initial average enrichment ≤ 4.4 wt% U-235 and discharged prior to IP3 Cycle 12 shall be placed in the STC basket. IP3 fuel assemblies V43 and V48 shall not be selected for transfer.

APPLICABILITY: Whenever one or more fuel assemblies are in the STC.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more fuel assemblies or NON FUEL HARDWARE in the STC do not meet the LCO limits.	A.1.1 Initiate action to restore compliance with LCO 3.1.2. OR A.1.2 Initiate action to move fuel to the IP3 spent fuel pit in accordance with IP3 Appendix A Technical Specification LCO 3.7.16.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.1 Verify by administrative means that the fuel assembly and NON FUEL HARDWARE meets the requirements specified in the LCO for placement in the STC.	Prior to placing the fuel assembly in the STC.
SR 3.1.2.2 Verify by visual inspection that a cell blocker which prevents inserting fuel assemblies and/or NON-FUEL HARDWARE into cells 1, 2, 3, and 4 of the STC is installed.	Prior to placing a Type 1 fuel assembly in the STC.

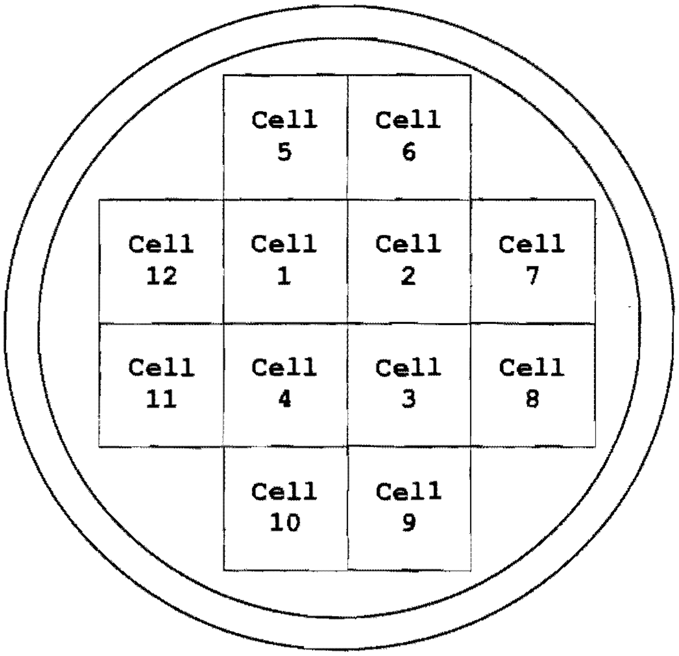

Figure 3.1.2-1
Shielded Transfer Canister Layout
(Top View)

Table 3.1.2-1
Minimum Burnup Requirements at Varying Initial Enrichments[a]

Maximum Assembly Initial Enrichment[f][g] (wt% U235)	Configuration A[c] Minimum Assembly Average Burnup (MWD/MTU)[b]	Configuration B[d] Minimum Assembly Average Burnup (MWD/MTU)[b]
2.0	5,400	6,000
2.5	13,800	18,800
3.0	22,100	28,600
3.5	30,000	37,300
4.0	36,900	44,600
4.5	42,700	52,500
5.0	48,700	Note (e)

(a) Fuel that does not meet the minimum assembly average burnup at a given initial enrichment is classified as Type 1 fuel. Fuel that meets the minimum assembly average burnup at a given initial enrichment is classified as Type 2 fuel.

(b) Linear interpolation between enrichment levels to determine minimum burnup requirements is permitted.

(c) Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation or where it can be shown that the insertion did not exceed 8 inches below the top of the active fuel.

(d) Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation and where the insertion was more than 8 inches below the top of the active fuel. This configuration also applies to fuel assemblies that have contained a Hafnium Flux Suppressor.

(e) Configuration B assemblies with enrichment greater than 4.5 are classified as Type 1 fuel.

(f) Natural or enriched uranium blankets are not considered in determining the fuel assembly average enrichment for comparison to the maximum allowed initial average enrichment.

(g) Rounding to one decimal place to determine initial enrichment is not permitted.

Table 3.1.2-2

NON FUEL HARDWARE[(a)] Post Irradiation Cooling Times and Allowable Average Burnup

Post-irradiation Cooling Time (years)	Maximum Burnup (MWD/MTU)			
	BPRAs and WABAs[(b, d)]	TPDs[(b)(c)]	RCCAs	Hafnium Flux Suppressors
≥ 6	≤ 20000	N/A	≤ 630000	≤ 20000
≥ 7	-	≤ 20000	-	-
≥ 8	≤ 30000	-	-	≤ 30000
≥ 9	≤ 40000	≤ 30000	-	-
≥ 10	≤ 50000	≤ 40000	-	-
≥ 11	≤ 60000	≤ 45000	-	-
≥ 12	-	≤ 50000	-	-
≥ 13	-	≤ 60000	-	-
≥ 14	-	-	-	-
≥ 15	-	≤ 90000	-	-
≥ 16	-	≤ 630000	-	-
≥ 20	-	-	-	-
Allowed Quantity and Location	Up to twelve (12) per transfer in any location	Up to twelve (12) per transfer in any location	Up to four (4) per transfer in Cells 1, 2, 3, and/or 4	Up to four (4) per transfer in Cells 1, 2, 3, and/or 4

(a) NON-FUEL HARDWARE burnup and cooling time limits are not applicable to Instrument Tube Tie Rods (ITTRs), since they are installed post-irradiation. NSAs are not authorized for loading in the STC.

(b) Linear interpolation between points is only permitted for BPRAs, WABAs, and TPDs, with the exception that interpolation is not permitted for TPDs with burnups greater than 90 GWd/MTU and cooling times greater than 15 years.

(c) N/A means not authorized for loading at this cooling time.

(d) Burnup and Cooling time limits in this column are only applicable to Loading Patterns 1-6 in Table 3.1.2-3. For Loading Patterns 7-12 in Table 3.1.2-3, the burnup and cooling time limits for a BPRA are the same as those for the fuel assembly they are located in.

Table 3.1.2-3 (Sheet 1 of 2)

Allowable STC Loading Configurations

Configuration[c]	Cells 1, 2, 3, 4[a][b]	Cells 5, 6, 7, 8, 9, 10, 11, 12[a][b]
1	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 25 years Initial Enrichment ≥ 2.3 wt% U-235
2	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235
3	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235
4	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.6 wt% U-235	Burnup ≤40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
5	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
6	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 40,000 MWD/MTU Cooling time ≥ 20 years Initial Enrichment ≥ 2.3 wt% U-235

Table 3.1.2-3 (Sheet 2 of 2)

Allowable STC Loading Configurations

Configuration[c]	Cells 1, 2, 3, 4[a][b]	Cells 5, 6, 7, 8, 9, 10, 11, 12[a][b]
7	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
8	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 15 years Initial Enrichment ≥ 3.4 wt% U-235
9	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 11 years Initial Enrichment ≥ 3.4 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 12 years Initial Enrichment ≥ 3.2 wt% U-235
10	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 10 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 55,000 MWD/MTU Cooling time ≥ 15 years Initial Enrichment ≥ 3.4 wt% U-235
11	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 6 years Initial Enrichment ≥ 3.2 wt% U-235	Burnup ≤ 45,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.2 wt% U-235
12	Burnup ≤ 60,000 MWD/MTU Cooling time ≥ 9 years Initial Enrichment ≥ 4.2 wt% U-235	Burnup ≤ 50,000 MWD/MTU Cooling time ≥ 14 years Initial Enrichment ≥ 3.6 wt% U-235

(a) Initial enrichment is the assembly average enrichment. Natural or enriched uranium blankets are not considered in determining the fuel assembly average enrichment for comparison to the minimum allowed initial average enrichment.

(b) Rounding to one decimal place to determine initial enrichment is permitted.

(c) Fuel with five middle Inconel spacers are limited to cells 1, 2, 3, and 4 for all loading configurations except loading configuration 6 which allows fuel with Inconel spacers in all cells.

3.1 INTER-UNIT FUEL TRANSFER

3.1.3 Shielded Transfer Canister (STC) Initial Water Level

LCO 3.1.3 The established water level in the STC shall be 9.0+0.5/-1.5 inches below the bottom of the STC lid.

APPLICABILITY: Prior to TRANSFER OPERATIONS when the STC is in the HI-TRAC and the STC lid has been installed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. STC water level not within limit.	----------------NOTE-------------- Water used for level restoration must meet the boron concentration requirement of LCO 3.1.1. -- A.1 Initiate action to restore STC water level.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify the initial STC water level is within limit by verifying the following during STC water level establishment: a. steam is emitted from the STC drain tube; and b. the volume of water removed is ≥ 35.4 gallons and ≤ 47.9 gallons.	Once prior to TRANSFER OPERATIONS.

3.1 INTER-UNIT FUEL TRANSFER

3.1.4 Shielded Transfer Canister (STC) Pressure Rise

LCO 3.1.4 The pressure rise in the STC cavity shall be ≤ 0.2 psi/hr averaged over a rolling 4 hour period.

APPLICABILITY: Over a 24 hour period after successful completion of LCO 3.1.3 and prior to TRANSFER OPERATIONS when the STC is in the HI-TRAC and the STC lid has been installed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Rate of STC cavity pressure rise not within limit.	A.1.1 Establish a vent path on the STC. AND ------------NOTE------------- Water used for recirculation must meet the boron concentration requirement of LCO 3.1.1. -------------------------------- A.1.2 Begin circulation of borated water in the STC to establish and maintain the STC water exit temperature < 180°F. AND A.1.3 Begin actions to determine the reason for exceeding the pressure rise limit.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action A.1.3 indicates a fuel misload.	B.1.1	Return the STC to the spent fuel pool and remove the STC lid.	12 hours
		AND	
	B.1.2	Return any misloaded fuel to the IP3 spent fuel pit in accordance with IP3 Appendix A Technical Specification LCO 3.7.16.	24 hours
C. Required Action A.1.3 does not indicate a fuel misload.	C.1	Develop and initiate corrective actions necessary to return the STC to compliance with LCO 3.1.3 and LCO 3.1.4.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	-------------------------NOTE------------------------ Pressure measurements shall be taken once upon establishing required water level AND hourly thereafter for 24 hours. Pressure may initially drop during pressure stabilization. --- Verify by direct measurement that the rate of STC cavity pressure rise is within limit.	Once prior to TRANSFER OPERATIONS.
SR 3.1.4.2	Verify that an ASME code compliant pressure relief valve or rupture disc and two channels of pressure instrumentation with a range of at least 0.1 psia to 15 psia and calibrated to within 1% accuracy within the past 12 months are installed on the STC.	During performance of SR 3.1.4.1.

3.1 INTER-UNIT FUEL TRANSFER

3.1.5 Shielded Transfer Canister (STC) Unloading

---NOTE---

1. Only IP3 spent fuel assemblies are permitted to be in the STC.
2. Once each IP3 spent fuel assembly removed from the STC has been placed in an IP2 spent fuel rack location and disconnected from the spent fuel pit bridge crane, it may not be returned to the STC.

LCO 3.1.5 IP3 spent fuel assemblies transferred to IP2 via the STC must be either in an approved IP2 spent fuel pit storage rack location per IP2 Appendix A Technical Specification LCO 3.7.13, in their authorized STC fuel basket cell, or be in transit between these two locations.

APPLICABILITY: Whenever the STC is in the Unit 2 spent fuel pit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more fuel assemblies not in the required location.	A.1 Initiate action to restore compliance with LCO 3.1.5	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify by administrative means that a fuel assembly returned to the STC has been re-loaded into the same STC cell from which it was removed.	Once, after each re-loaded fuel assembly is returned to the STC.

4.0 DESIGN FEATURES

4.1 Inter-Unit Fuel Transfer

4.1.1 Fuel Assemblies

Fuel assemblies selected for inter-unit transfer of fuel shall meet the fuel characteristics specified in Table 4.1.1-1.

4.1.2 Criticality

4.1.2.1 The Shielded Transfer Canister (STC) is designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water;

c. A nominal 9.218 inch center-to-center distance between fuel assemblies placed in the STC basket;

d. Basket cell ID: 8.79 in. (nominal);

e. Basket cell wall thickness: 0.28 in. (nominal);

f. B_4C in the Metamic neutron absorber: ≥ 31.5 wt.% and ≤ 33.0 wt.%;

g. The B_4C in the Metamic neutron absorber will contain boron with an isotopic B-10 content of at least 18.4%;

h. Metamic panel thickness: ≥ 0.102 in.;

i. The size and location of the neutron absorber panels shall be in accordance with drawing 6015, revision 6, which can be found in the Licensing Report (Holtec International Report HI-2094289).

4.1.2.2 Drainage

The STC is designed and shall be maintained to prevent inadvertent draining.

4.1.2.3 Capacity

The STC is designed and shall be maintained with a capacity of no more than 12 fuel assemblies.

(continued)

4.0 DESIGN FEATURES (continued)

4.1.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 2004 Edition, is the governing Code for the STC, as clarified below, except for Code Sections V and IX. The latest effective editions of ASME Code Sections V and IX, including addenda, may be used for activities governed by those sections, provided a written reconciliation of the later edition against the 2004 Edition, is performed. Table 4.1.3-1 lists approved alternatives to the ASME Code for the design of the STC.

4.1.4 Geometric Arrangements and Process Variables

The following are geometric arrangements and process variables that require a one time verification as part of each inter-unit fuel transfer operation:

1. LOADING OPERATIONS, TRANSFER OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures ≥ 0°F.

2. LOADING OPERATIONS shall only be conducted when the spent fuel pit water temperature and the fuel handling building ambient temperatures are both ≤ 100°F.

3. LOADING OPERATIONS shall only be conducted when the IP3 spent fuel pit contains no unirradiated fuel assemblies.

4. LOADING OPERATIONS shall only be conducted when the irradiated fuel assemblies in the IP3 spent fuel pit have been subcritical for at least 90 days.

5. TRANSFER OPERATIONS shall only be conducted when the outside air temperature is ≤ 100°F.

6. TRANSFER OPERATIONS shall only be conducted when the STC trunnions are offset from the HI-TRAC trunnions in the azimuthal direction by at least 30 degrees.

7. TRANSFER OPERATIONS shall only be conducted after STC seal leak tests have demonstrated no detected leakage when tested to a sensitivity of 1×10^{-3} ref-cm^3/s in accordance with the "pre-shipment" test requirements of ANSI N14.5.

8. Prior to installing the HI-TRAC lid the HI-TRAC water level shall be verified by two separate inspections to be within +0/-1 inch of the top of the STC lid.

(continued)

4.0 DESIGN FEATURES (continued)

9. TRANSFER OPERATIONS shall only be conducted after the combined leak rate through the HI-TRAC top lid and vent port cover seals are confirmed to be water tight using an acceptable leak test from ANSI N14.5 and the pool lid seal is verified to be water tight by visual inspection.

10. TRANSFER OPERATIONS shall not occur with a TRANSPORTER that contains > 50 gallons of diesel fuel.

Table 4.1.1-1

Fuel Assembly Characteristics

Fuel Assembly Class	15x15[a]
No. of Fuel Rod Locations	204
Cladding Type	ZR
Guide/Instrument Tube Type	ZR
Design Initial U (kg/assembly)	≤ 473
Fuel Rod Clad O.D. (in)	≥ 0. 422
Fuel Rod Clad I.D. (in)	≤ 0. 3734
Fuel Pellet Diameter (in)	≤ 0. 3659
Fuel Rod Pitch (in)	≤ 0.563
Active Fuel Length (in)	≤ 144
Fuel Assembly Length (in)	≤ 160
Fuel Assembly Width (in)	≤ 8.54
No. of Guide and/or Instrument Tubes	21
Guide/Instrument Tube Thickness (in)	≥ 0. 017
Axial Blanket Enrichment (wt % U-235) [b]	≤ 3.2
Axial Blanket Length (in) [b]	≥ 6

(a) All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within the 15x15 class.

(b) Applicable only if axial blankets are present.

4.0 DESIGN FEATURES (continued)

Table 4.1.3-1 (page 1 of 2)

List of ASME Code Alternatives for the STC

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
STC Confinement Boundary	ND-1000	Statement of requirements for Code stamping of components.	Cask confinement boundary is designed, and will be fabricated in accordance with ASME Code, Section III, Subsection ND to the maximum practical extent, but Code stamping is not required.
STC Confinement Boundary	ND-2000	Requires materials to be supplied by ASME-approved material supplier.	Holtec approved suppliers will supply materials with CMTRs per ND-2000.
STC and STC basket assembly	ND-3100 NG-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are not applicable. The Licensing Report, serving as the Design Specification, establishes the service conditions and load combinations for fuel transfer.
STC Confinement Boundary	ND-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of cask vessel is as a radionuclide confinement boundary under normal and hypothetical accident conditions. Cask is designed to withstand maximum internal pressure and maximum accident temperatures.
STC Confinement Boundary	ND-8000	States requirement for name, stamping and reports per NCA-8000	STC to be marked and identified in accordance with drawing 6013[(a)]. Code stamping is not required. QA data package prepared in accordance with Holtec's approved QA program.

4.0 DESIGN FEATURES (continued)

Table 4.1.3-1 (page 2 of 2)

List of ASME Code Alternatives for the STC

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
STC Basket Assembly	NG-4420	NG-4427(a) requires a fillet weld in any single continuous weld may be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.	Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal STC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the STC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis. From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).
STC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	STC basket to be marked and identified in accordance with drawing 6015[a]. No Code stamping is required. The STC basket data package is to be in conformance with Holtec's QA program.

(a) Holtec International Report HI-2094289

5.0 PROGRAMS

The following programs shall be established, implemented and maintained.

5.1 Transport Evaluation Program

a. For lifting of the loaded STC or loaded HI-TRAC using equipment which is integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply.

b. This program is not applicable when the loaded HI-TRAC is in the fuel building or is being handled by equipment providing support from underneath (e.g., on air pads).

c. The loaded HI-TRAC may be lifted to any height necessary during TRANSFER OPERATIONS provided the lifting equipment is designed in accordance with items 1, 2, and 3 below.

1. The metal body and any vertical columns of the lifting equipment shall be designed to comply with stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. All vertical compression loaded primary members shall satisfy the buckling criteria of ASME Section III, Subsection NF.

2. The horizontal cross beam and any lifting attachments used to connect the load to the lifting equipment shall be designed, fabricated, operated, tested, inspected, and maintained in accordance with applicable sections and guidance of NUREG-0612, Section 5.1. This includes applicable stress limits from ANSI N14.6.

3. The lifting equipment shall have redundant drop protection features which prevent uncontrolled lowering of the load.

d. The lift height of the loaded HI-TRAC above the transport route surface or other supporting surface shall be limited to 6 inches, except as provided in Specification 5.1.c.

5.2 Metamic Coupon Sampling Program

A coupon surveillance program shall be implemented to maintain surveillance of the Metamic neutron absorber material under the radiation, chemical, and thermal environment of the STC.

The surveillance program will be implemented to monitor the performance of Metamic by installing a minimum of four bare coupons near the maximum gamma flux elevation (mid height) at no less than four circumferential downcomer areas around the STC fuel basket. At any time during its use the STC must have a minimum of one coupon installed in each quadrant. Metamic coupons used for testing must have been installed during the entire fuel loading history of the STC.

The following specifications apply:

(i) Coupon size will be nominally 4" x 6". Each coupon will be marked with a unique identification number.

(continued)

5.0 PROGRAMS (continued)

(ii) Pre-characterization testing: Before installation, each coupon will be measured and weighed. The measurements shall be taken at locations pre-specified in the test program. Each coupon shall be tested by neutron attenuation before installation in the STC. The weight, length, width, thickness, and results of the neutron attenuation testing shall be documented and retained.

(iii) Four coupons shall be tested at the end of each inter-unit fuel transfer campaign. A campaign shall not last longer than two years. The coupons shall be measured and weighed and the results compared with the pre-characterization testing data. The results shall be documented and retained.

(iv) The coupons shall be examined for any indication of swelling, delamination, edge degradation, or general corrosion. The results of the examination shall be documented and retained.

(v) The coupons shall be tested by neutron attenuation and the results compared with the pre-characterization testing data. The results of the testing shall be documented and retained. Results are acceptable if the measured value is within +/-2.5% of the value measured for the same coupon at manufacturing.

(vi) The coupons shall be returned to their locations in the STC unless anomalous material behavior is found. If the results indicate anomalous material behavior, evaluation and corrective actions shall be pursued.

5.3 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:

1. a change in the TS incorporated in the license; or
2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that do not meet the criteria of Specification 5.3.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).

(continued)

5.0 PROGRAMS (continued)

5.4 Radiation Protection Program

5.4.1 The radiation protection program shall appropriately address STC loading and unloading conditions, including transfer of the loaded TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.

5.4.2 Total (neutron plus gamma) measured dose rates shall not exceed the following:

a. 1400 mrem/hr on the top of the STC (with lid in place).

b. 5 mrem/hr on the side of the TRANSFER CASK

5.4.3 The STC and TRANSFER CASK surface neutron and gamma dose rates shall be measured as described in Section 5.4.6 for comparison against the limits established in Section 5.4.2.

5.4.4 If the measured surface dose rates exceed the limits established in Section 5.4.2, then:

a. Administratively verify that the correct contents were loaded in the correct fuel basket cell locations.

b. Perform a written evaluation to determine whether TRANSFER OPERATIONS can proceed without exceeding the dose limits of 10 CFR 72.104 or 10 CFR 20.1301.

5.4.5 If the verification and evaluation performed pursuant to Section 5.4.4 show that the fuel is loaded correctly and the dose rates from the STC and TRANSFER CASK will not cause the dose limits of 10 CFR 72.104 or 10 CFR 20.1301 to be exceeded, TRANSFER OPERATIONS may occur. Otherwise, TRANSFER OPERATIONS shall not occur until appropriate corrective action is taken to ensure the dose limits are not exceeded.

5.4.6 STC and TRANSFER CASK surface dose rates shall be measured at approximately the following locations:

a. The dose rate measurement shall be taken at the approximate center of the STC top lid. Two (2) additional measurements shall be taken on the STC lid approximately 180 degrees apart and 12 to 18 inches from the center of the lid, avoiding the areas around the inlet and outlet ports. The measurements must be taken when the STC is in the HI-TRAC after the steam space is established and prior to HI-TRAC lid installation.

(continued)

5.0 PROGRAMS (continued)

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b. A minimum of four (4) dose rate measurements shall be taken on the side of the TRANSFER CASK approximately at the cask mid-height plane. The measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket.