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.

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 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 indentions of the logical connectors.

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

(continued)

1.2 Logical Connectors (continued)

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

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

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2. Reduce . . .	

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

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 storage of irradiated fuel. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.
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 discuss 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 as well as certain Notes in the Surveillance column that modify performance requirements.

EXAMPLES

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify parameter is within limits.	12 hours

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

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify parameter is within limits.	Within 24 hours prior to moving irradiated fuel 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. The use of "prior to" indicates that the surveillance must be performed once before the initiation of fuel handling activities. This type of Frequency does not qualify for 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).

2.0 SAFETY LIMITS

This section is not applicable to defueled facilities.

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

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during specified conditions in the Applicability for the individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance whether such failure is experienced during 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 or as measured from the time a specified condition of the Frequency is met.

SR 3.0.3

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

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

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

3.1 DEFUELED PLANT SYSTEMS

3.1.1 Spent Fuel Pool Water Level

LCO 3.1.1 The spent fuel storage water level shall be $\geq$ 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 pool.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify the spent fuel pool water level is $\geq$ 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	Within 24 hours prior to movement of irradiated fuel assemblies. AND and 24 hours thereafter.

3.1 DEFUELED PLANT SYSTEMS

3.1.2 Spent Fuel Pool Boron Concentration

LCO 3.1.2 The spent fuel pool boron concentration shall be $\geq$ 500 ppm.

APPLICABILITY: During movement of fuel assemblies in the spent fuel pool, or

When fuel assemblies are stored in Region 2 of the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u>	
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	<u>OR</u>	
	A.2.2 Verify by administrative means Region 2 spent fuel pool verification has been performed since the last movement of fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILANCE		FREQUENCY
SR 3.1.2.1	Verify the spent fuel pool boron concentration is within limit.	Within 31 days prior to movement of irradiated fuel assemblies. AND 31 days thereafter.

3.1 DEFUELED PLANT SYSTEMS

3.1.3 Spent Fuel Assembly Storage

LCO 3.1.3 The combination of initial enrichment and discharge fuel burnup of each spent fuel assembly stored in Region 2 shall be within the Acceptable Burnup Domain of Figure 3.1.3-1.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Initiate action to move the noncomplying fuel assembly from Region 2.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILANCE		FREQUENCY
SR 3.1.3.1	Verify by administrative means the initial enrichment and discharge fuel burnup of the fuel assembly is in accordance with Figure 3.1.3-1.	Prior to storing the fuel assembly in Region 2

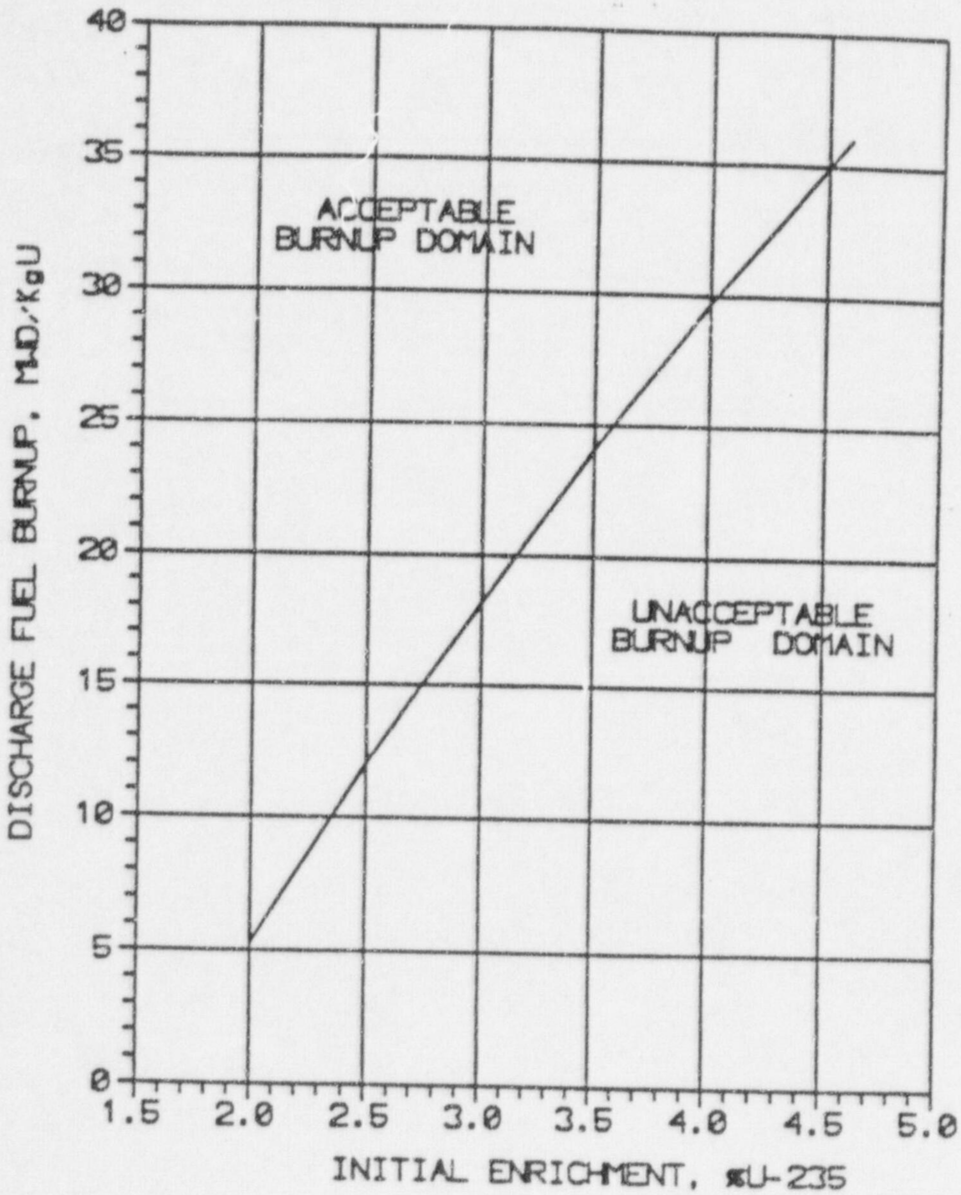

Figure 3.1.3-1 (Page 1 of 1)
Fuel Assembly Burnup Limits in Region 2

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Location

Zion Units 1 and 2 are located at the Zion Station which consists of a tract of land of approximately 250 acres located in the extreme eastern portion of the city of Zion, Lake County, Illinois, on the west shore of Lake Michigan approximately 6 miles NNE of the center of the city of Waukegan, Illinois, and 8 miles south of the center of the city of Kenosha, Wisconsin. It is located at longitude 87° 48.1' W and latitude 42° 26.8' N.

4.0 DESIGN FEATURES

4.2 Fuel Storage

4.2.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 4.65 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties;

c. A nominal 9.14 inch center to center distance between fuel assemblies placed in Region 2 of the spent fuel storage racks;

d. A nominal 10.54 inch north-south and 10.78 east-west center to center distance between fuel assemblies placed in Region 1 of the spent fuel storage racks;

e. One row of six storage cells with a nominal 18.75 inch center to center distance between cells for storing failed fuel canisters in Region 1 of the spent fuel storage racks;

f. Irradiated fuel assemblies with a discharge burnup in the "acceptable burnup domain" of Figure 3.1.3-1 allowed unrestricted storage in either Region 1 or Region 2 of the spent fuel storage rack(s); and

g. New or irradiated fuel assemblies with a discharge burnup in the "unacceptable burnup domain" of Figure 3.1.3-1 stored in Region 1 of the spent fuel storage racks.

4.2 Fuel Storage

4.2.1 Criticality (continued

4.2.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.65 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties; and

c. A nominal 21 inches center to center distance between fuel assemblies placed in the storage racks.

4.2.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent draining of the pool below elevation 598 ft.

4.2.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 3012 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Decommissioning Plant Manager shall be responsible for overall plant operations and shall delegate in writing the succession to this responsibility during his absence.

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

5.1.2 The Shift Supervisor shall be responsible for the shift command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 General Organizational Requirements

Onsite and offsite organizations shall be established for station and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safe storage and handling of nuclear fuel.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Manual or a site specific quality assurance program description incorporated directly or by reference in the DSAR.

b. The Decommissioning Plant Manager shall be responsible for overall plant safety and shall have control over those onsite activities necessary for safe storage and handling of nuclear fuel.

c. A Corporate Vice-President shall have corporate responsibility for the safe handling and storage of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure the safe handling and storage of nuclear fuel.

d. The individuals who train the Certified Fuel Handlers and those who carry out health physics and quality assurance functions may report to an appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

(continued)

5.2 Organization (continued)

5.2.2 Facility Staff

The facility staff organization shall include the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 5.2.2-1.

b. 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.

c. All fuel handling operations shall be directly supervised by a Certified Fuel Handler.

d. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform functions important to the safe storage and handling of nuclear fuel assemblies (e.g., Certified Fuel Handlers, non-certified operators, radiation protection personnel, and key maintenance personnel) such that the heavy use of overtime is not routinely required.

e. The Shift Supervisor shall be a Certified Fuel Handler.

Table 5.2.2-1
Minimum Shift Crew Composition[a]

Position	Minimum Crew Number
Shift Supervisor	1
Non-certified operator	1
Total	2

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

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff Qualifications

5.3.1 Staff Qualifications

Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971, with the following exceptions:

Either the Manager of the Health Physics Department or the Lead Health Physicist shall meet or exceed the qualifications of "Radiation Protection Manager" of Regulatory Guide 1.8, September 1975.

The Decommissioning Operations Manager shall meet the requirement of Operations Manager in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel," dated March 8, 1971, with the exception that this individual may be qualified as a Certified Fuel Handler at time of appointment in lieu of holding a Senior Reactor Operator license.

5.4 Training

5.4.1 Training

A training and retraining program for the Certified Fuel Handlers shall be maintained under the direction of the Decommissioning Plant Manager or designee.

5.0 ADMINISTRATIVE CONTROLS

5.5 Procedures

5.5.1 Procedures

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;

b. Fire Protection Program implementation; and

c. All programs specified in Specification 5.6.

5.0 Administrative Controls

5.6 Programs and Manuals

The following programs shall be established, implemented and maintained.

5.6.1 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 5.7.2 and Specification 5.7.3.

c. Licensee initiated changes to the ODCM:

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

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

    ii. A determination that the change(s) will 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 do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

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

(continued)

5.6 Programs and Manuals

5.6.1 Offsite Dose Calculation Manual (ODCM) (continued)

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 effective. 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.

d. The ODCM shall contain the limits to be used for releasing solid material to unrestricted areas. Compliance with these limits shall be verified by instruments set at lower limits of detection (LLDs) and maximum allowable gamma activity concentration sensitivities contained in NRC Regulatory Guide 1.86, June 1974, and Nuclear Energy Institute Topical Report 97-02, May 1997, respectively. Applicable radionuclide distributions, scaling factors, and sampling methods shall also be specified in the ODCM.

5.6.2 Radioactive Effluent Controls Program

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

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

(continued)

5.6 Programs and Manuals

5.6.2 Radioactive Effluent Controls Program (continued

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten 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 to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31 day period would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to Appendix I to 10 CFR 50.

(continued)

5.6 Programs and Manuals

5.6.2 Radioactive Effluent Controls Program (continued)

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the following:

1. 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

2. For tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of a 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 to areas beyond the site boundary, conforming to Appendix I to 10 CFR 50;

i. Limitations on the annual and quarterly doses to a member of the public from tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to Appendix I to 10 CFR 50; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 are applicable to Radioactive Effluent Controls Program surveillance frequencies.

(continued)

5.0 Administrative Controls

5.6 Programs and Manuals

5.6.3 Outdoor Storage Tank Radioactivity Monitoring Program

This program provides controls for the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. This program is required if radioactive liquid is contained in unprotected (as defined below) outdoor storage tanks. The liquid radwaste quantities shall be determined in accordance with the ODCM

The program shall include 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 Outdoor Storage Tank Radioactivity Monitoring Program surveillance frequencies.

(continued)

5.0 Administrative Controls

5.6 Programs and Manuals

5.6.4 Technical Specification (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 involves an unreviewed safety question as defined in 10 CFR 50.59.

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

d. Proposed changes that meet the criteria of b(1) or b(2) 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) as modified by approved exemptions.

5.0 Administrative Controls

5.7 Reporting Requirements

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

5.7.1 Occupational Radiation Exposure Report

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

A tabulation covering the previous calendar year shall be submitted prior to April 30 of each year on the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man rem exposure according to work and job functions (e.g., fuel handling, surveillance, routine maintenance, special maintenance (describe maintenance), and waste processing). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

5.7.2 Annual Radiological Environmental Operating Report

---NOTE---
A single submittal may be made for a multiple unit station.

The Annual Radiological Environmental Operating Report covering unit activities during the previous calendar year shall be submitted before May 15 of each year. The report shall include summaries, interpretations, and analysis 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 (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I of 10 CFR Part 50.

(continued)

5.7 Reporting Requirements

5.7.2 Annual Radiological Environmental Operating Report (continued)

In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.7.3 Radioactive Effluent Release Report

------------------------------NOTE------------------------------

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

The Radioactive Effluent Release Report covering unit activities shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and Process Control Program and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to10 CFR Part 50.

5.0 Administrative Controls

5.8 High Radiation Area

5.8.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the "control device" or "alarm signal" required by 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is equal to or less than 1000 mrem/hr at 30 cm (12 in) from the radiation source or from any surface which the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

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

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

(continued)

5.8 High Radiation Area

5.8.2 In addition to the requirements of Specification 5.8.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in) from the radiation source or from any surface which the radiation penetrates shall require the following:

a. Locked doors to prevent unauthorized entry. The keys shall be maintained under the administrative control of the operating shift supervision on duty and/or health physics supervision.

b. Personnel access and exposure control over activities being performed within these areas shall be specified by an approved RWP. During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, continuous surveillance and radiation monitoring of the work area by an individual qualified in radiation protection procedures may be substituted for the routine RWP procedure.

c. Each person entering the area shall be provided with an alarming radiation monitoring device which continuously integrates the radiation dose rate (such as an electronic dosimeter). Continuous coverage by a radiation technician may be substituted for alarming dosimetry.

5.8.3 For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/hr at 30cm (12 in.), that are located within large areas (with the exception of 5.8.4), including the containment outside the missile barrier, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded (by a more substantial obstacle than a rope), conspicuously posted, and a flashing light shall be activated as a warning device.

5.8 High Radiation Area

5.8.4 For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mrem/hr at 30 cm (12 in.), that are located within the containment inside the missile barrier, where no enclosure exists for purposes of locking the individual area, the access control shall be per the following:

a. The missile barrier ingress/egress points shall be barricaded, locked and conspicuously posted to prevent access; or

b. 1. The missile barrier ingress/egress points shall be conspicuously posted and have direct or electronic surveillance that is capable of preventing unauthorized entry; and

2. Additional localized postings shall be provided in areas with normal personnel access inside the missile barrier to inform personnel of dose rates greater than 1000 mrem/hr at 30 cm (12 in.).

5.0 ADMINISTRATIVE CONTROLS

5.9 Reviews

5.9.1 Qualified Technical Review

Thorough reviews of the documents specified below shall be conducted by a Qualified Technical Reviewer. Persons performing these reviews shall be knowledgeable in the subject area being reviewed. Qualified Technical Reviews must be completed prior to implementation of proposed activities.

a. Qualified Technical Reviewers shall be individuals without direct responsibility for the document under review; these reviewers may be from the same functionally cognizant organization as the individual or group performing the original work.

b. Qualified Technical Reviewers shall have at least 5 years of professional experience and either a Bachelor's degree in Engineering or the Physical Sciences or shall have equivalent qualifications evaluated on a case by case basis and approved by the Decommissioning Plant Manager. The Decommissioning Plant Manager shall document the appointment of Qualified Technical Reviewers.

c. The following subjects shall be independently reviewed by a Qualified Technical Reviewer:

   1. Safety evaluations for changes in the facility as described in the DSAR, changes in procedures as described in the DSAR, and tests or experiments not described in the DSAR to verify that such actions do not involve a change to the Technical Specifications or will not involve an unreviewed safety question as defined in 10 CFR 50.59;

   2. Proposed changes to the programs required by Specification 5.6, to verify that such changes do not involve a change to the Technical Specifications and will not involve an unreviewed safety question as defined in 10 CFR 50.59; and

   3. Proposed changes to the license, Technical Specifications, or Bases.

(continued)

5.9 Reviews (continued)

5.9.2 Station Review Committee (SRC)

The SRC is responsible for reviewing and advising the Decommissioning Plant Manager on matters related to the safe storage of nuclear fuel. This review function is independent of line organization responsibilities.

a. The SRC shall include a minimum of five members. Alternates may be substituted for regular members. The licensee shall designate in writing the chairman, the members, and alternates for the SRC.

b. The SRC shall collectively have experience and knowledge in the following functional areas:

   1. Fuel handling and storage (including the potential for criticality),
   2. Chemistry and radiochemistry,
   3. Engineering,
   4. Radiation protection, and
   5. Regulatory assurance.

c. The SRC shall hold at least one meeting per quarter.

d. A quorum shall consist of three regular members or their duly appointed alternates. Those members representing the line organizations responsible for the operation and maintenance of the facility shall not constitute a majority of the quorum. At least one member of the quorum shall be the chairman or the chairman's designated alternate.

e. As a minimum, the SRC shall perform the following functions:

   1. Advise the Decommissioning Plant Manager on all matters related to safe storage of nuclear fuel; and

   2. Notify the responsible Corporate Vice-President of any safety significant disagreement between the SRC and the Decommissioning Plant Manager within 24 hours.

(continued)

5.9 Reviews (continued)

5.9.2 Station Review Committee (SRC) (Continued)

f. The SRC shall be responsible for reviewing:

1. The safety evaluations for new documents or changes to documents completed under the provisions of 10 CFR 50.59 to verify that such actions do not involve an unreviewed safety question as defined in 10 CFR 50.59. This review may be completed after implementation of the affected procedure;

2. Changes to structures, systems, or components important to the safe storage of nuclear fuel to verify that such changes do not involve an unreviewed safety question as defined in 10 CFR 50.59. This review may be completed after implementation of the change;

3. Tests or experiments involving the safe storage of nuclear fuel to verify that such tests or experiments do not involve an unreviewed safety question as defined in 10 CFR 50.59. This review may be completed after performance of the test or experiment;

4. Proposed changes to the license, Technical Specifications, or Bases.

5. Violations of codes, regulations, orders, license requirements, or internal procedures/instructions having nuclear safety significance;

6. Indications of unanticipated deficiencies in any aspect of design or operation of structures, systems, or components that could affect safe storage of nuclear fuel;

7. Significant accidental, unplanned, or uncontrolled radioactive releases, including corrective action(s) to prevent recurrence;

(continued)

5.9 Reviews (continued)

5.9.2 Station Review Committee (SRC) (Continued)

8. Significant operating abnormalities or deviations from normal and expected performance of equipment that affect safe storage of nuclear fuel;

9. Internal and external experience information related to the safe storage of nuclear fuel that may indicate areas for improving facility safety; and

10. Reportable Events.

Reports or records of these reviews shall be forwarded to the Decommissioning Plant Manager within 30 days after completion of the review.

5.9.3 Records

Written records of reviews shall be maintained. As a minimum, these records shall include:

a. Results of the activities conducted under the provisions of Specifications 5.9.1 and 5.9.2; and

b. Determination of whether each item considered under Specifications 5.9.2.f.1 through 5.9.2.f.3 involves an unreviewed safety question as defined in 10 CFR 50.59.

ZION

NUCLEAR STATION

PERMANENTLY DEFUELED

TECHNICAL SPECIFICATIONS

BASES

TABLE OF CONTENTS

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 and LCO 3.0.2 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification.)
LCO 3.0.2	LCO 3.0.2 establishes that, upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that: a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time or is no longer applicable, unless otherwise specified.

(continued)

BASES

LCO 3.0.2 (continued)

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

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

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

SR 3.0.1

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

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

SR 3.0.2

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

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

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

(continued)

BASES

SR 3.0.3

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

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

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

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

(continued)

BASES

SR 3.0.3 (continued)

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

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

B 3.1 DEFUELED PLANT SYSTEMS

B 3.1.1 Spent Fuel Pool Water Level

BASES

BACKGROUND	When the plant was operational, this specification provided assurance that the assumptions regarding the iodine decontamination factor would be met following a fuel handling accident. Following the permanent defueling of the reactors, the fuel handling accident was re-analyzed based on the extended time since shutdown and corresponding reductions in iodine activity that are consistent with the plant's permanently defueled condition. As described in Ref. 1 and 2, these new analyses determined that 10 CFR 100 and 10 CFR 50 App. A, Criterion 19 limits would still be met even with no decontamination by the spent fuel pool water. Although the specification for spent fuel pool water level during fuel handling operations is no longer needed to ensure an adequate iodine decontamination factor, the specification continues to provide assurance of adequate cooling for the irradiated fuel being handled by ensuring that it remains covered by water and provides significant shielding for personnel safety. Therefore the specification was retained essentially unchanged from the operational Technical Specifications. The assumptions in the fuel handling accident analyses are given in Ref. 1 and 2.
APPLICABLE SAFETY ANALYSES	In the operational Technical Specifications, the specification for minimum water level in the spent fuel pool during fuel handling activities provided assurance of substantial iodine removal if a fuel handling accident were to occur. However, as indicated in Ref. 1 and 2, the limits of 10 CFR 100 and 10 CFR 50 App. A, Criterion 19 would not be exceeded if a

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	fuel handling accident occurred even with no removal of iodine activity by spent fuel pool water. This is the result of the decay of radioactive iodine during the lengthy period since the last reactor operation at the station. However the specification for water level in the in the SFP also ensures that irradiated fuel which is not in the storage racks will be adequately cooled by ensuring that it remains covered with water during normal fuel handling operations, and provides significant shielding for personnel safety.
LCO	The spent fuel pool water level is required to be 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The 23 ft level was formerly based on preserving the assumptions of the previous fuel handling accident analysis. Although this level is no longer needed for iodine decontamination following a fuel handling accident, past practice indicates that this level provides assurance that the irradiated fuel being handled will be covered by water and consequently will be adequately cooled and provides significant shielding for personnel safety. The 23 ft level has therefore been retained as the minimum required for movement of irradiated fuel assemblies within the spent fuel pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool since it is during such activities that irradiated fuel may be higher than the top of the fuel in the storage racks. An irradiated fuel assembly that is higher than that stored in the racks may not be protected against uncovery by design features which ensure that fuel in the racks remains covered with water. This design feature consists of the lowest pipe opening in the spent fuel pool being at approximately 598' which is above the top of the fuel stored in the racks (approximately 590').

(continued)

BASES

ACTIONS

A.1

When the initial conditions for an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This action effectively precludes the possibility of withdrawing an irradiated fuel assembly above the water level which would result in a loss of cooling and shielding. This does not preclude movement of a fuel assembly to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

This SR verifies that the spent fuel pool water level is sufficiently high to ensure adequate cooling and shielding for the fuel being handled. The water level in the spent fuel pool must be checked periodically. The 24 hour Frequency is appropriate because the volume in the pool is normally stable and is acceptable based on operating experience. In addition, water level changes are controlled by plant procedures.

REFERENCES

1. Zion Station Calculation 22S-0-110X-0057, Fuel Handling Accident Offsite Dose Calculation with Extended Radioactive Decay and no AB Filtration

2. Zion Station Calculation 22S-0-110X-0059, Fuel Handling Accident Control Room Dose Calculation with Extended Radioactive Decay

B 3.1 DEFUELED PLANT SYSTEMS

B 3.1.2 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The spent fuel pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate areas. Region 1, with 336 storage positions, is designed to accommodate fuel with a maximum initial enrichment of 4.65 wt% U-235, regardless of burnup. Region 2, with 2670 storage positions, is designed to accommodate fuel with various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.1.3-1. Region 1 also contains six (6) defective fuel assembly storage containers.

The water in the spent fuel pool normally contains dissolved boron which results in large subcriticality margins. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water. The design maintains each region in a subcritical condition with the regions fully loaded.

The double contingency principle discussed in ANSI N16.1-1975 and an NRC letter dated April 14, 1978 (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions since only a single accident or event need be considered at one time. For example, the most severe scenario is associated with the movement of fuel from Region 1 to Region 2 and accidental misloading of a fuel assembly in Region 2. This could potentially increase the reactivity of Region 2. To prevent criticality if an accidental misloading event were to occur, boron is dissolved in the spent fuel pool water.

(continued)

BASES

BACKGROUND (continued)	Safe storage in the high density storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.1.3, "Spent Fuel Assembly Storage." Prior to movement of an assembly to Region 2, it is necessary to perform SR 3.1.3.1.
APPLICABLE SAFETY ANALYSES	Events can be postulated that could increase the k_{eff} of the spent fuel pool. However, the presence of dissolved boron in the spent fuel pool water prevents criticality in both regions of the pool. These postulated events are of two types. A fuel assembly could be inadvertently misloaded in Region 2 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated event is associated with a fuel assembly which is dropped adjacent to a fully loaded Region 2 storage rack. This could have a small positive reactivity effect on Region 2. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated event scenarios. Analyses of these two types of events are described in Ref. 2.
LCO	The specified minimum spent fuel pool boron concentration is 500 ppm. The specified concentration of dissolved boron in the spent fuel pool preserves the assumptions used in the analyses of the postulated event scenarios as described in Ref. 2.
APPLICABILITY	This LCO applies during movement of fuel assemblies in the spent fuel pool or whenever fuel assemblies are stored in Region 2 of the spent fuel pool until a spent fuel pool verification has been performed following the last movement of fuel assemblies in Region 2 of the spent fuel pool.

(continued)

BASES

APPLICABILITY (continued)

The LCO applies during movement of fuel assemblies in the pool because the potential for a dropped fuel assembly exists during such movements. The LCO also applies when fuel assemblies are stored in Region 2 of the spent fuel pool, until a verification has been performed following the last movement because during movement there is the potential for an inadvertent misloading of an assembly that should be in Region 1 into Region 2. However the independent verification provides adequate assurance that no misloading has occurred in Region 2. There is no restriction regarding storage of fuel assemblies in Region 1 since any fuel assembly meeting the limitations described under Design Features may be stored in Region 1.

ACTIONS

A.1, A.2.1, and A.2.2

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude the occurrence of a reactivity event or to mitigate the consequences of a reactivity event in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies in the spent fuel pool. This does not preclude movement of a fuel assembly to a safe position.

Action must also be immediately initiated to restore the boron concentration simultaneously with suspending movement of fuel assemblies.

An acceptable alternative to restoring the boron concentration is to verify by administrative means that the spent fuel pool verification has been performed since the last movement of fuel assemblies in Region 2 of the spent fuel pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed events are fully addressed. The 31 day Frequency is appropriate considering the volume of the spent fuel pool, the normally maintained boron concentration, and because no major dilution of pool water is expected to take place over this period of time.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. Letter from C. Y. Shiraki, NRC to T. J. Kovach, ComEd, dated February 23, 1993, Issuance of License Amendment 142/131, allowing increase of the Spent Fuel Pool storage Capacity to 3012 assemblies

B 3.1 DEFUELED PLANT SYSTEMS

B 3.1.3 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate areas. Region 1, with 336 storage positions, is designed to accommodate fuel with a maximum initial enrichment of 4.65 wt% U-235, regardless of burnup. Region 2, with 2670 storage positions, is designed to accommodate fuel with various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure 3.1.3-1. Region 1 also contains six (6) defective fuel assembly storage containers.

The water in the spent fuel pool normally contains dissolved boron which results in large subcriticality margins. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water. The design maintains each region in a subcritical condition with the regions fully loaded.

The double contingency principle discussed in ANSI N16.1-1975 and an NRC letter dated April 14, 1978 (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions since only a single accident or event need be considered at one time. For example, the most severe scenario is associated with the movement of fuel from Region 1 to Region 2 and accidental misloading of a fuel assembly in Region 2. This could potentially increase the reactivity of Region 2. To prevent criticality if an accidental misloading event were to occur, boron is dissolved in the spent fuel pool water.

(continued)

BASES

BACKGROUND (continued)	Safe storage in the high density storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.1.3, "Spent Fuel Assembly Storage." Prior to movement of an assembly to Region 2, it is necessary to perform SR 3.1.3.1.
APPLICABLE SAFETY ANALYSES	The hypothetical accidents can only take place during or as a result of the movement of an assembly. For these accident occurrences, the presence of soluble boron in the spent fuel pool (controlled by LCO 3.1.2, "Spent Fuel Pool Boron Concentration") prevents criticality in both regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the provisions of LCO 3.1.3.
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with Figure 3.1.3.-1, ensures that the k_{eff} of the spent fuel pool will always remain < 0.95, assuming the pool to be flooded with unborated water. This is supported by the analyses described in Ref. 2.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in Region 2 of the spent fuel pool. The provisions of Design feature 4.2.1 provide protection against criticality for fuel stored in Region 1 of the spent fuel pool.

(continued)

BASES

ACTIONS

A.1

When the configuration of fuel assemblies stored in Region 2 the spent fuel pool is not in accordance with Figure 3.1.3-1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.1.3-1.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.1.3-1. For fuel assemblies in the unacceptable range of Figure 3.1.3-1, storage is only allowed in Region 1.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. Letter from C. Y. Shiraki, NRC to T. J. Kovach, ComEd, dated February 23, 1993, Issuance of License Amendment 142/131, allowing increase of the Spent Fuel Pool storage Capacity to 3012 assemblies

ZION STATION
LICENSE AMENDMENT REQUEST NUMBER 98-06;
PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARD CONSIDERATIONS
FOR PROPOSED CHANGES

ATTACHMENT C

SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

ComEd has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
>
> Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> Involve a significant reduction in a margin of safety.

The proposed amendment would eliminate License Conditions that are no longer applicable with Units 1 and 2 permanently shutdown and defueled, and replace the existing operational Custom Technical Specifications (CTS) with Permanently Defueled Technical Specifications (PDTS). The specific changes in the License Conditions and the CTS have been categorized as:

Administrative Changes
Editorial Changes
More Restrictive Changes
Redundancy or Relocation Changes
Less Restrictive Changes

The determination that the criteria set forth in 10 CFR 50.92 are met for these changes is indicated in the following table. In this table the changes in the first four categories have been evaluated against the criteria of 10 CFR 50.92 on a categorical basis. Changes in the last category, Less Restrictive Changes, have been addressed individually since the reasons that 10 CFR 50.92 criteria are satisfied differ between changes. Based upon the evaluations presented in this table, ComEd has concluded that all changes involved in this proposed amendment involve no significant hazards consideration.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
Administrative Changes These are changes in which a License Condition or CTS requirement has been eliminated because the mode of applicability or conditions which invoke the requirement will no longer occur. Consequently the License Condition or specification would never require or prohibit any action. For CTS Definitions, these are changes in which the definition has been eliminated because the defined term is not used.	No. This type of change only removes requirements that are no longer used. Since the removal of these requirements does not affect any structures, systems, or components (SSCs) or the conduct of activities with the units permanently defueled, there is no change in the probability or consequences of any accident.	No. Since the removal of these requirements does not affect any SSCs or the conduct of activities, no new types of accidents are created.	No. Since the removal of these requirements does not affect any SSCs or the conduct of activities, there is no reduction in any safety margin.
Editorial Changes These are changes in format, word choice, grammar, or terminology that do not alter any requirement.	No. This type of change does not alter the meaning of the specification. Since there is no change in requirements, there is no change in the probability or consequences of any accident.	No. Since there is no change in requirements, no new types of accidents are created.	No. Since there is no change in requirements, there is no reduction in any safety margin.
More Restrictive Changes These are changes in which the resulting requirement is more restrictive than the original License Condition or CTS requirement.	No. This type of change adds new requirements, removes existing exceptions, or renders existing limits more conservative. Such changes do not change the probability or consequences of any accident.	No. Since the change renders the specifications more restrictive, no new types of accidents are created.	No. Since the change renders the specifications more restrictive, there is no reduction in any safety margin.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
Redundancy or Relocation Changes These are changes in which a License Condition or CTS requirement is eliminated either because it is redundant to requirements in regulations or other specifications, or because it does not meet the criteria of 10 CFR 50.36(c) and has been/will be relocated to ComEd controlled documents.	No. In this type of change the requirement continues to exist but is no longer in the license or technical specifications. Since there is no change in requirements, there is no change in the probability or consequences of any accident.	No. Since there is no change in requirements, no new types of accidents are created.	No. Since there is no change in requirements, there is no reduction in any safety margin.
Less Restrictive Changes These are changes in which the resulting requirement is less restrictive than the original License Condition or CTS requirement. The individual changes in this category are identified below along with the basis for the change and a No Significant Hazards Consideration evaluation			
<u>License condition 2.C.(5)</u> (Safe Shutdown Fire Protection Program) This License Condition is no longer needed since, in accordance with 10 CFR 50.48(b) - (d), the operational fire protection program was based on maintaining the ability to achieve and maintain safe shutdown in the event of a fire. ComEd has submitted the certifications of permanent shutdown and defueling required by 10 CFR 50.82(a)(1) and accordingly, the fire protection program is governed by 10 CFR 50.48(f). This regulation requires a program that addresses the potential for fires which could cause the release or	No. The effect of the change is to recognize that the objective of the fire protection program has been made relevant to the units defueled condition. This will not change the probability of a fire. The consequence of concern in the operational program was the inability to shutdown the units. That consequence is no longer a concern. Therefore the possible change in consequences is not significant.	No. The only events of concern are fires. The change program objectives will not create any new types of fires.	No. The only margin of safety that could be attributed to the operational fire protection program would be a measure of the ability to shutdown the units. This margin of safety is no longer relevant.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
spread of radioactive materials (i.e., which could result in a radiological hazard). Since the regulation is self-invoking, no new license condition is needed.			
License Condition 2.C.(8) (Program for Secondary Chemistry) This License Condition is no longer needed to ensure safety because both the primary and secondary sides of the steam generators are depressurized for SAFSTOR conditions. Consequently, the conditions promoting steam generator tube degradation have been essentially eliminated and the consequences of such degradation are no longer significant.	No. Neither the probability nor the consequences of an accident are affected because the type of accident/event which was the concern of this program, (primary to secondary leak) is no longer credible with the primary and secondary sides of the steam generators depressurized and vented.	No. The only accident/event of concern involving secondary chemistry was the primary to secondary leak.	No. There are no longer any margins of safety concerning primary to secondary leaks.
License Condition 2.C.(9) (Program for Leakage From Systems Outside Containment Following An Accident) This License Condition is no longer needed to ensure safety because design basis accidents inside containment are no longer credible. Additionally, all primary systems penetrating containment will be depressurized for long term SAFSTOR conditions.	No. Neither the probability nor the consequences of an accident are affected because the type of accident/event which was the concern of this program, (leakage from systems penetrating containment containing highly radioactive fluids during an accident) is not credible with the RCS depressurized and vented and the reactors defueled.	No. This program was only concerned with accidents resulting in post accident highly radioactive leakage from systems penetrating containment.	No. There are no longer any margins of safety concerning post accident highly radioactive systems penetrating containment.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
License Condition 2.C.(10) (Program for Airborne Iodine Determination Under Accident Conditions) This License Condition is no longer needed to ensure safety because sufficient time has elapsed since the units were shut down for the radioactive iodine in the fuel to decay to levels that would not result in exceeding the exposure limits for personnel in the control room stated in 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 following a fuel handling accident. The only remaining credible accident or event that could cause a breach of the fuel cladding is a fuel handling accident. Calculations have shown that the GDC 19 limits would be met for a fuel handling accident even without credit for the charcoal filters in the control room ventilation system.	No. Neither the probability nor the consequences of an accident are affected because the condition which was the concern of this program (high iodine levels prohibiting access to vital areas under accident conditions) is not credible since the iodine has undergone an extended decay period.	No. The only accident/events of concern involving this program were those resulting in high post accident airborne iodine levels.	No. Compliance with the margins of safety involving post accident iodine stated in the regulations is achieved without this program
License Condition 2.C.(11) (March 14, 1983, Order concerning certain NUREG-0737 Issues) This license condition no longer needed since none of the issues subject to the order are relevant with the units permanently defueled. These issues are: simulator examinations, plant shielding, post accident sampling of reactor coolant and	No. All but one of these issues are limited to accidents (such as a LOCA) that are only credible for operational units with fueled reactors. Since these accidents are no longer credible, their probability and consequences cannot increase. The one issue	No. The subject issues pertained only to the prevention and mitigation of identified accidents. Even if the plant were still operational the elimination of these requirements would not create any new	No. The accident analysis demonstrates that the required margin of safety (dose limit at the EAB) for the only remaining credible accident that could involve any of the subject issues will remain well within

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
containment atmosphere, training for mitigating core damage, auxiliary feedwater flow indication, containment isolation dependability, post accident containment radiation monitor, containment pressure indication, containment water level indication, containment hydrogen indication, and post accident effluent monitors for iodine and noble gas. Regarding the last issue, the above described accident analysis demonstrates that 10 CFR 100 limits would not be exceeded if a fuel handling accident were to occur. Therefore, inclusion of a license condition concerning these monitors is no longer appropriate.	that could potentially involve a fuel handling accident is that concerning post accident effluent monitors for iodine and noble gas. Such monitors have no affect on the probability of the accident. The accident analysis shows that the consequences remain acceptable even with no credit for the monitors.	type of accident.	established limits.
CTS 3.12 / 3.12 contain the requirements for limiting the quantity of radioactivity in the gas decay tanks and for limiting the hydrogen concentration in the waste gas system. The specifications in this section have not been included in the PDTS since the gas decay tanks have been vented and are no longer in use, and since there is no longer any source of hydrogen in the waste gas system.	No. The rupture of a gas decay tank is no longer credible, nor is a hydrogen explosion in the waste gas system. Since the accidents are no longer credible, elimination of associated limits will not increase their probability of occurrence or consequences.	No. Elimination of the curie content and hydrogen concentration limits does not involve new failure mechanisms or modes since there is no longer a hazard.	No. Since the gas decay tanks are no longer in use and since there is no longer any source of hydrogen in the waste gas system, there is no longer any attribute to which a safety margin can be applied.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 3.13.2 / 4.13.2 contain the requirements for operability, compensatory action, and surveillance of the fuel building ventilation exhaust system including filters and charcoal adsorbers during movement of irradiated fuel or loads over irradiated fuel in the fuel building. The specifications in this section have not been included in the PDTS since the radioactive iodine in the irradiated fuel has decayed such that the dose limits for personnel in the control room given in 10 CFR 50, App. A, Criterion 19 and the limits for dose at the site boundary given in 10 CFR 100 would not be exceeded in the event of a design basis fuel handling accident even if no credit is assumed for any charcoal adsorbtion.	No. The probability of a fuel handling accident is unaffected by the operability of the ventilation system. The accident consequences (doses due to radioactive iodine) which the fuel building ventilation exhaust system was designed to mitigate are no longer significant. Therefore elimination of the ventilation system operability requirements will not significantly increase these consequences if the accident should occur.	No. The inoperability of the fuel building ventilation exhaust system would not initiate any new accident, nor would it introduce any new failure mechanisms or modes involving the design and operation of other systems structures, or components.	No. Due to the decay of iodine since the units last operated, the margins of safety for the consequences of a fuel handling accident established by the NRC in 10 CFR 50, App. A, Criterion 19 and 10 CFR 100 have already been satisfied.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 3.13.11 / 4.13.11 contain the limits, compensatory action, and surveillance requirements for the spent fuel pool water level.			
The CTS applicability statement includes movement of control rods in the SFP. This has not been included in the PDTS applicability statement since the specification is no longer based on scrubbing of iodine in the event of a fuel handling accident. There has been sufficient decay of iodine such that applicable post accident release limits can be met without crediting iodine removal by the SFP water. The specification is now based on ensuring that fuel being moved remains under water. Therefore, the applicability statement need no longer include movement of a control rod.	No. The accident of concern is one in which fuel is damaged by a falling object. The probability of such an accident is not increased by eliminating water level requirements for handling of control rods since the level has no effect on the likelihood that the rod will be dropped. The consequences of such an accident are not significantly affected since there is no longer significant amount of iodine for the water to remove. Also no credit was taken in the accident analyses for the slowing effect of the water.	No. The elimination of SFP water level requirements for handling of control rods does not introduce any new failure mechanisms or modes.	No. Due to the decay of iodine since the units last operated, the margins of safety for the consequences of an accident involving a dropped control rod, which are established by the NRC in 10 CFR 50, App. A, Criterion 19 and 10 CFR 100 have already been satisfied.
The CTS requirement that the surveillance be performed within 2 hours prior to the start of fuel movement has been changed to within 24 hours prior to the start of fuel movement. Since the spent fuel pool water level is not subject to sudden or frequent changes, the 24 hour limit provides	No. Changing the timing of the initial SFP level verification in no way affects the likelihood of whether a fuel assembly or other object will be dropped. The consequences are not affected	No. The change only affects the timing of performance of the initial level surveillance. Performance of the surveillance by observing	No. The margin of safety involved in this surveillance is defined by the minimum SFP water level. The level requirements are

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
adequate assurance of safety.	since operating experience has shown there is a high degree of certainty the water level in the SFP will not undergo large fluctuations in a 24 hour period. This substantiated by the existing frequency requirement for subsequent surveillances, which is 24 hours.	the SFP water level does not involve any physical change in the plant or the manner in which any structure, system, or component is operated.	unchanged. Only the timing of the initial level verification is affected.
CTS 3.13.14 / 4.13.14 contain the limits, compensatory action, and surveillance requirements for SFP boron concentration. The following changes were made to the CTS requirements:			
The CTS applicability statement of "Whenever fuel assemblies are in the spent fuel storage pool" was modified to be consistent with Zion ITS 3.7.15 applicability statement of "When fuel assemblies are stored in Region 2 of the spent fuel pool and a spent fuel pool verification has not been performed	No. The accident of concern is a criticality in the SFP. The probability of this accident has not been significantly increased since sub-criticality continues to be ensured by the independent verification of Region 2, regardless of boron concentration. The verification provides an independent mechanism to ensure that the assumptions of the	No. The only accident involving boron concentration and fuel assembly location is a criticality accident.	No. The margin of safety accepted by the NRC is that the K_{eff} of the SFP will remain below 0.95. The criticality analyses show that this margin will still be maintained with the proposed changes to the applicability statement.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
since the last movement of fuel assemblies in the spent fuel pool." This is based on the Zion criticality analyses which demonstrate that the K_{eff} of the SFP will remain below 0.95 with any fuel assembly authorized for Zion stored in Region 1 of the SFP and with unborated water in the pool.	criticality analyses are met. This verification is only needed for Region 2 since Region 1 can safely accommodate any fuel authorized in the Design Features specifications. The consequences of a criticality accident are unaffected since the function of the specification is to prevent the accident rather than to mitigate it.		
A new Action (A.2.2) was included in the PDTS to allow an alternative to suspending fuel movements and restoring boron concentration if the concentration is below the limit. The alternative action is to verify that only the proper fuel assemblies are stored in Region 2. This provides assurance that the K_{eff} of the SFP will remain below 0.95. even though the boron concentration is below limits and provides a compensatory measure which is consistent with the Applicability statement.	No. This change provides an alternative action that is as effective as the existing action in providing assurance that a criticality accident will not occur. This change only increases the options for ensuring that the accident will not happen and therefore does not increase the probability of the accident. The consequences of a criticality accident are unchanged since they are unaffected by the choice of actions taken to ensure that the accident will not happen.	No. The only accident involving boron concentration and fuel assembly location is a criticality accident.	No. The margin of safety accepted by the NRC is that the K_{eff} of the SFP will remain below 0.95. The criticality analyses show that this margin will still be maintained with the proposed changes to the action statement.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 3.14 / 4.14 contain the requirements for operability, compensatory action, and surveillance of various area and process radiation monitors. The specifications for the monitors that are required by the CTS to be operable with both units defueled monitors have not been included in the PDTS since none of these monitors is credited in the analyses of the remaining credible accidents. These monitors are the SFP area, control room area, Technical Support Center area, auxiliary building area, component cooling system, and the service water system monitors.	No. Elimination of operability requirements for these monitors from the technical specifications does not affect the probability of any accident since there are no initiating events associated with the monitors. The analyses which determined the consequences of the remaining credible accidents demonstrated acceptable results without taking any credit for the monitors.	No. The monitors are reactive components that sense radiation levels. The failure or inoperability of these monitors does not create any conditions that can result in new accidents or cause other SSCs to fail in such a manner that result in new accidents.	No. These monitors are not needed to maintain post accident doses within the margins of safety established in 10 CFR 100 and 10 CFR 50, App. A GDC 19.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 3.17 / 4.17 contain the requirements for operability, compensatory action, and surveillance of the particulate filters and charcoal adsorbers in various ventilation systems including the control room and fuel handling ventilation systems. The specifications in this section have not been included in the PDTS since no credit is taken for any ventilation system function in the analyses of the accidents that are credible with both units permanently defueled	No. Even when the units were operational, these filters and adsorbers did not serve to decrease the probability of an accident. Since the iodine has decayed such that the dose limits in 10 CFR 50, App. A, Criterion 19 and in 10 CFR 100 would not be exceeded in the event of a design basis fuel handling accident, they have not been credited in any of the remaining accident analyses and there are no significant consequences from the elimination of the operability requirements for these filters and adsorbers.	No. Elimination of operability requirements for these filters and adsorbers will not produce any new failure modes in the associated systems that would result in any type of design basis accidents.	No. Due to the decay of iodine since the units last operated, these filters and adsorbers are not needed to maintain doses within the margins of safety established in 10 CFR 100 and 10 CFR 50, App. A GDC 19.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 5.2 provides a general description of the design and function of the RCS. This specification has not been included in the PDTS since the RCS no longer performs the stated functions of removing heat from the core or serving as a post accident boundary for fission products. The RCSs of both units have been depressurized and vented for the SAFSTOR period.	No. The RCS is not involved in the design basis accidents that remain credible with both units permanently defueled. The probability of an accident involving the integrity or functionality of the RCS is essentially zero. With no core in the reactor, the off site consequences of a accident involving the RCS are no longer significant.	No. Eliminating design restrictions on the RCS will not produce any new type of accident since it no longer serves any function related to design basis accidents.	No. There is no longer any margin of safety associated with the RCS.
CTS 5.3 provides a general description of the design and size of the core. This specification has not been included in the PDTS since all fuel has been permanently removed from both reactors. Consequently there is no longer any reactor core.	No. The probability and consequences of an accident involving the reactor core have not been increased since there no longer is any core.	No. There can be no new accident involving the reactor core since there no longer is any core.	No. There is no longer any margin of safety involving the reactor core since there no longer is any core.
CTS 5.4 provides a general description of the design and function of the containment. This specification has not been included in the PDTS since the containments no longer perform the stated functions of serving as a post accident	No. There is no longer any fuel in the containment so it is no longer needed to reduce the probability of accidents caused by external sources. With the reactors defueled, the containment is no	No. Even when the units were operational there was no accident that would be created by a violation of any of the these design features. Therefore	No. The containment no longer functions to limit parameters following a design basis accident. Therefore there is no reduction in safety margins

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
boundary for fission products or biological shield. All fuel has been permanently removed from both containments.	longer needed to reduce the consequences of an accident by serving as a post accident fission product barrier or radiation shield.	deleting these features can't produce a new kind of accident.	from deleting the description of its design features.
CTS 5.6 provides a general description of the Seismic Class 1 equipment vital to safe shutdown or containment isolation, or whose failure might cause or increase the severity of a Loss of Coolant Accident (LOCA). The specification also provides a description of requirements for meeting a Design Basis Earthquake and special requirements for safe shutdown equipment. The specification also notes that other SSCs are designed to withstand an Operational Basis Earthquake or per applicable codes, and are defined as Seismic Class 2 or 3. These descriptions have not been included in the PDTS since safe shutdown, post accident containment isolation, LOCAs, and the ability to withstand an earthquake and keep operating are no longer of concern. Those seismic design features that are relevant with the units permanently shutdown and defueled will be described in the DSAR.	No. The SSCs identified in the specification do not affect the probability or consequences of the accidents that remain credible with both units permanently defueled. Consequently the seismic qualification of these systems also has no affect the probability or consequences of these accidents. Elimination of requirements pertaining to an Operational Basis Earthquake also has no affect the probability or consequences of these accidents since the units will no longer be operational. The DSAR requirements will provide assurance that the seismic qualifications of SSC's are adequate to preclude increasing the probability or consequences of the accidents that do remain credible.	No. The SSCs identified in the specification are only involved with accidents that could affect operational units. Elimination of seismic qualification requirements for these SSCs can only affect their ability to respond the analyzed operational accidents.	No. There is no longer any safety margin associated with the SSCs identified in the specification. The DSAR requirements will ensure that an adequate safety margin is provided for those SSCs involved with the accidents that remain credible.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 6.1.1.a requires that lines of authority, responsibility, and communication be documented in the Quality Assurance Manual, which is also used for ComEd's operating sites. A provision has been added to the PDTS allowing this documentation to be contained in a site-specific quality assurance program description incorporated directly or by reference in the DSAR. This provision would allow implementation of a site-specific program appropriate to the permanently defueled status of the Zion units.	No. This is an administrative change involving the location of quality assurance requirements. There is no mechanism for it to directly affect the probability or consequences of an accident.	No. The change does not directly affect any structure, system, or component or the manner in which they are operated or maintained. Therefore the change cannot introduce any new failure mechanism or create a new accident initiator.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
CTS 6.1.3 contains the requirements for minimum shift manning. The CTS Figure 6.1-1 requirement that at least two Non-certified operators be on shift has been changed in PDTS Table 5.2.2-1 such that one Non-certified operator is required, for a total of 2 individuals required to be on shift. The proposed manning has been found to be acceptable for other permanently shutdown sites with a single spent fuel pool.	No. Single unit shutdown sites have been successfully functioning with a minimum shift crew of two individuals. Since Zion has a shared SFP and support systems, the demands on the crew are no greater than for a single unit site. Therefore would be no significant reduction in the ability of the crew to prevent accidents or operational events, nor would there be any significant reduction in the ability of the crew to mitigate an accident or event.	No. The number of individuals on shift does not affect the failure mode of any equipment or create any new accident initiators. The credible accidents and operational events remain limited to fuel handling accidents, low level radioactive waste handling accidents, and loss of cooling to the spent fuel pool at normal and reduced levels.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 6.1.5 requires that training/retraining of plant personnel be in accordance with ANSI N18.1. This specification has not been included in the PDTS since, with the units permanently shutdown and defueled only the training/retraining program for the Certified Fuel Handlers need be specified in the PDTS. Some of the ANSI N18.1 requirements, such as those for training in startup and shutdown procedures and emergency shutdown systems, are no longer appropriate. Moreover, the spectrum of credible accidents and the quantity and complexity of activities required for safety has been greatly reduced from that at an operating plant. Consequently it is only necessary that the PDTS specify the training/retraining requirements for the personnel who are most directly responsible for maintaining facility safety. These personnel are the Shift Supervisors, who are required by PDTS 5.2 to be Certified Fuel Handlers. The training and retraining program for Certified Fuel Handlers was reviewed and approved by the NRC and must be maintained as specified by PDTS Section 5.4. The training/retraining of other plant personnel will be governed by ComEd controlled documents.	No. The on shift individuals having overall responsibility for preventing the few design basis accidents that remain credible or mitigating their consequences are the Certified Fuel Handlers. The training and retraining requirements for these individuals will be maintained in accordance with a program required by the PDTS. This program and the ComEd controlled programs training/retraining of other plant personnel will provide adequate assurance that there will be no increase in the probability or consequences of an accident.	No. There are no credible mechanisms for changes in training requirements to directly result in new or different types of accidents.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 6.1.6 requires that retraining of personnel be conducted at intervals not to exceed two years. This specification has not been included in the PDTS for reasons similar to that given for CTS 6.1.5 above, i.e., inclusion in the PDTS of specific retraining requirements for personnel who are not required to be Certified Fuel Handlers is not necessary. Only the retraining requirements for the Certified Fuel Handlers need be specified in the PDTS. The retraining program for the Certified Fuel Handlers (which includes biennial retraining) was reviewed and approved by the NRC and must be maintained as specified by PDTS Section 5.4. The training/retraining of other plant personnel will be governed by ComEd controlled documents.	No. As discussed above it is only necessary that training/retraining requirements for the Certified Fuel Handlers will be maintained in the PDTS by reference. This includes the periodicity of the retraining. The frequency of retraining for Certified Fuel Handlers is specified in the NRC approved program. This provides adequate assurance that there will be no increase in the probability or consequences of the few design basis accidents that remain credible.	No. There are no credible mechanisms for changes in training requirements to directly result in new or different types of accidents.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
CTS 6.2.1.b requires that Emergency Operating Procedures (EOPs) be prepared, implemented, and maintained in accordance with NUREG-0737 and Generic Letter 82-33 (Supplement 1 to NUREG-0737). This specification has not been included in the PDTS. In response to the requirements in the above identified documents, ComEd prepared and implemented a procedure generation package for	No. The subject EOPs were only used in response to accidents that had already begun and therefore had no effect on the probability of an accident. The EOPs only addressed operational accidents and therefore had no effect on the consequences of the accidents that are credible with both units defueled.	No. The deletion of EOP requirements cannot directly result in new or different types of accidents.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
upgrading the EOPs when the plant was operational. However, none of the EOPs are applicable with the units permanently defueled.			
CTS 6.2.1.c requires that Station Security Plan procedures be prepared, implemented, and maintained. Since Station Security Plan implementing procedures are listed in this specification, their review requirements are specified in CTS 6.2.3 and 6.2.4. These review requirements may be excluded from the PDTS since the Station Security Plan contains review requirements for these procedures.	No. This change is an administrative change involving the review of security procedures, and cannot directly affect the probability or consequences of an accident.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is operated and maintained. Therefore it cannot produce a new or different kind o accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
CTS 6.2.1.d requires that Generating Station Emergency Response Plan procedures be prepared, implemented, and maintained. Since Generating Station Emergency Response Plan procedures are listed in this specification, their review requirements are specified in CTS 6.2.3 and 6.2.4. These review requirements may be excluded from the PDTS since the Station Security Plan contains review requirements for these procedures.	No. This change is an administrative change involving the review of Generating Station Emergency Response Plan procedures, and cannot directly affect the probability or consequences of an accident.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is normally operated and maintained. Therefore it cannot produce a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 6.2.1.e requires that Process Control Program procedures be prepared, implemented, and maintained. Since Process Control Program procedures are listed in CTS 6.2.1, their review requirements are specified in CTS 6.2.3 and 6.2.4. These review requirements have not been included in the PDTS since, review requirements for these procedures are already contained in 10 CFR 71.113.	No. This change is an administrative change involving the review of Process Control Program procedures, and cannot directly affect the probability or consequences of an accident.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is normally operated and maintained. Therefore it cannot produce a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
CTS 6.2.1.h requires that procedures be prepared, implemented, and maintained for a Post Accident Sampling Program which would ensure the capability to obtain and analyze reactor coolant and containment atmosphere samples, and collect and analyze or measure radioactive iodine and particulates in plant gaseous effluents under accident conditions. This specification has not been included in the PDTS since, with both units defueled, there are no credible accident scenarios that release significant radioactivity to the reactor coolant, containment atmosphere, or plant gaseous effluents, or that will result in severe accident conditions that would preclude obtaining samples.	No. Requirements concerning post accident sampling cannot affect the probability that any accident will occur. The consequences of the accidents that remain credible have been shown to be within the criteria given in 10 CFR 100 and 10 CFR 50, App. A Criterion 19 with no credit for sampling or active response by station personnel. Therefore the consequences will not be significantly affected by removal of these requirements for sampling procedures from the technical specifications.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is normally operated and maintained. Therefore it cannot produce a new or different kind of accident	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
Site procedures will be maintained, as part of the radiation protection program required by 10 CFR 20.1101, which are adequate to for sampling gaseous effluents under the conditions which would result from the accidents that remain credible with both units permanently defueled.			
CTS 6.2.6 requires that the listed programs be established, implemented and maintained. The Zion PDTS contain a new specification, 5.6.1.d, that is not in the CTS. This specification requires that the ODCM contain the limits for releasing solid material to unrestricted areas, and that the limits be based on the lower limits of detection (LLDs) established in accordance with certain NRC and industry standards. The specification also requires that applicable radionuclide distributions, scaling factors, and sampling methods be specified in the ODCM. Inclusion of these requirements will establish approved, consistent, and explicit material release requirements that are independent of technological changes which can alter LLDs.	No. This change does not affect any of the analyzed accidents. The change only affects the LLD for unrestricted release of material and there is no analyzed accident involving such low level material.	No. The proposed LLDs are sufficiently low such that there is no threat to public health and safety involved. Consequently, there is no new accident involved with use of the LLDs established by this proposed change.	No. The safety margin involved in this change is established by 10 CFR 20.1402 at 25 mrem/yr. The industry standards specified in the proposed change would limit exposures to 5 mrem/yr. Therefore the margin of safety would be maintained.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
Zion PDTS Section 5.6 includes a Technical Specification Bases Control Program that is not in the CTS. This program, which is contained in specification 5.6.4, provides a means for processing changes to the Bases of the PDTS without prior NRC approval provided the change meets the criteria of 10 CFR 50.59.	No. This change is an administrative change involving the review of Bases changes , and cannot directly affect the probability or consequences of an accident.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is operated and maintained. Therefore it cannot produce a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
<u>6.2.6.A</u> requires that a Radioactive Effluent Controls Program established, implemented and maintained in accordance with 10 CFR 50.36a, and specifies the elements that the program is to contain. A statement has been added to the end of Zion PDTS 5.6.2 to clearly indicate that the 25% surveillance frequency allowance provided by SR 3.0.2 is also applicable to the Radioactive Effluent Controls Program surveillances. This is consistent with the Zion ITS.	No. The change only involves the grace period allowed for performing surveillances required by the Radioactive Effluent Controls Program. There is no change in the nominal periodicity of the surveillances. The credible accidents in no way involve the frequency of these surveillances.	No. There is no mechanism for requirements concerning a surveillance grace period to create any accident.	No. There is no safety margin involved with the grace period for performing surveillances required by the Radioactive Effluent Controls Program

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 6.6.1.B contains the requirements pertaining to the Occupational Exposure Report. The due date for the report was changed from March 1 of each year, as specified in the CTS, to April 30 of each year consistency with the standard Improved Technical Specifications.	No. This change is an administrative change involving the due date for an annual report, and cannot directly affect the probability or consequences of an accident.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is operated and maintained. Therefore it cannot produce a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.
CTS 6.6.1.C contains the requirements pertaining to the Annual Radiological Environmental Operating Report. The due date for the report was changed from May 1 of each year to May 15 of each year for consistency with the standard Improved Technical Specifications.	No. This change is an administrative change involving the due date for an annual report, and cannot directly affect the probability or consequences of an accident.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is operated and maintained. Therefore it cannot produce a new or different kind of accident.	No. This change does not directly involve any limits or parameters and therefore cannot affect any margin of safety.

<table>
<tr><th>Description of Change</th><th>Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?</th><th>Does the change create the possibility of a new or different kind of accident from any previously evaluated?</th><th>Does the change involve a significant reduction in a margin of safety?</th></tr>
<tr><td>CTS 6.6.1.E requires submittal of a monthly report containing operating data such as hours critical, hours on-line, net electrical energy produced, shutdowns, daily power levels, etc.

This specification has not been included in the PDTS since this information is no longer relevant with the units permanently shut down. Elimination of this specification is consistent with Generic Letter 97-07 which prescribes the contents of the Monthly Operating Report and which is addressed to:

"All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel."

ComEd has submitted this certification.</td><td>No. This change is an administrative change involving the need for a monthly report of operating data, and cannot directly affect the probability or consequences of an accident.</td><td>No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is operated and maintained. Therefore it cannot produce a new or different kind of accident.</td><td>No. This change does not directly involve any changes to limits or parameters and therefore cannot affect any margin of safety.</td></tr>
</table>

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 6.6.3.B.e requires an annual report regarding expansion plans for Waukegan Regional Airport including FAA form # 5010 (Airport Master Record). This specification has not been included in the PDTS since it is not necessary to ensure safety. A 1989 study supporting License Amendment 119/108 evaluated the probability of aircraft crashing in the vicinity of certain important plant structures and causing fires that threatened safety related components. This study determined that the probability of such an event for the Crib House air intakes, including the entire roof area and a 40 foot zone around the air intakes was 7.5×10^{-9} per year. As documented in the NRC SER for the amendment, this probability would remain below 1.0×10^{-7} per year even allowing for estimated growth of the airport through 2008. With both units permanently defueled, the Crib House and its components are no longer safety related. However the target area is comparable to that of the fuel building. Based on this low apparent probability combined with the fact that the FAA form # 5010 has not changed since 1990, ComEd considers that this specification is no longer needed to ensure safety and can be excluded from the PDTS.	No. The elimination of the reporting requirement will not affect the probability of an aircraft crash at the site. The apparent probability of an aircraft crash affecting the fuel building remains acceptably low. The elimination of the reporting requirement does not affect the extent of damage that could be caused by such a crash. Therefore the consequences of the accident are unchanged.	No. The change in reporting requirements does not affect the manner in which any SSC functions or fails to function, or the manner in which it is operated. Consequently, the change produces no new accident initiators.	No. The change in reporting requirements does not change limits or parameters and therefore cannot affect any margin of safety.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION FOR PROPOSED CHANGES

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
CTS 6.8 requires that all doors listed in a CTS Table be closed if there is the possibility of flooding. This specification has not been included in the PDTS since there is no significant safety concern from a flooding event with the units permanently defueled. The most significant flooding threat to the site would be that caused by a seiche on Lake Michigan. This could potentially result in a water level 2 feet above grade (592.0') for about 20 minutes. This is well below the top of the spent fuel pool (approx. 617'). If any components involved in spent fuel pool cooling were affected there would be adequate time to restore the components or to take other actions to compensate for their unavailability.	No. Elimination of this requirement does not affect the probability of a flooding event since that is an act of nature. The consequences of a credible flood event are not increased since, with the both units permanently defueled, there are no significant safety consequences.	No. This change is concerned solely with flooding events and does not involve any other type of event.	No. This change does not directly involve any changes to limits or parameters and therefore cannot affect any margin of safety.
CTS 6.9 requires that: Documentation for changes to the PCP contain sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s); Documentation for changes to the PCP contain a determination that the change will maintain the overall conformance of the solidified waste	No. This change is an administrative change involving the review of changes to the PCP, and cannot directly affect the probability or consequences of an accident.	No. This change does not directly affect any plant equipment involved with the safe storage and handling of nuclear fuel or how such equipment is operated and maintained. Therefore it cannot produce a new or different kind of accident.	No. This change does not directly involve any changes to limits or parameters and therefore cannot affect any margin of safety.

Description of Change	Does the change involve a significant increase in the probability or consequences of a previously evaluated accident?	Does the change create the possibility of a new or different kind of accident from any previously evaluated?	Does the change involve a significant reduction in a margin of safety?
product to existing requirements of Federal, State, or other applicable regulations; and Changes to the PCP become effective after review and acceptance by the Onsite Review and Investigative Function and the approval of the Decommissioning Plant Manager. These requirements have not been included in the PDTS since adequate review requirements will be incorporated into the ODCM.			

ZION STATION
LICENSE AMENDMENT REQUEST NUMBER 98-06;
PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT FOR PROPOSED CHANGES

ComEd has evaluated this proposed operating license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards consideration.

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards consideration.

(ii) there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

As documented in Attachment A, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

As documented in Attachment A, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

ZION STATION
LICENSE AMENDMENT REQUEST NUMBER 98-06;
PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

ATTACHMENT E

TABLE SUMMARIZING THE DISPOSITION OF THE ZION CTS IN THE ZION PDTS AND COMPARING THE DISPOSITION WITH THE MAIN YANKEE PDTS

ATTACHMENT E

TABLE SUMMARIZING THE DISPOSITION OF THE CTS IN THE ZION PDTS AND COMPARING THE DISPOSITION WITH THE MAIN YANKEE PDTS

Zion CTS Section/Specification	Equivalent Maine Yankee PDTS	Proposed Disposition in Zion PDTS*
Definitions		
1.1 ACTION	1.1 Definitions	1.1, Definitions
1.2 ACTUATION DEVICE	None	Not Included
1.3 ACTUATED EQUIPMENT	None	Not Included
1.4 ACTUATION LOGIC TEST	None	Not Included
1.5 AXIAL FLUX DIFFERENCE	None	Not Included
1.6 Previously deleted	N/A	N/A
1.7 CHANNEL CALIBRATION, INSTRUMENT	None	Not Included
1.8 CHANNEL CHECK	None	Not Included
1.9 CHANNEL FUNCTIONAL TEST	None	Not Included
1.10 Previously deleted	N/A	N/A
1.11 CONTAINMENT INTEGRITY	None	Not Included
1.12 Previously deleted	N/A	N/A
1.13 CONTROLLED LEAKAGE	None	Not Included
1.14 CORE ALTERATION	None	Not Included
1.14A CORE OPERATING LIMITS REPORT	None	Not Included
1.15 DEFINED TERMS	1.1 Definitions Note	1.1, Definitions Note
1.16 DEGREE OF REDUNDANCY	None	Not Included
1.17 DOSE EQUIVALENT I-131	None	Not Included
1.18 E-AVERAGE DISINTEGRATION ENERGY	None	Not Included
1.19 Previously deleted	N/A	N/A
1.20 IDENTIFIED LEAKAGE	None	Not Included
1.21 INSTRUMENT CHANNEL	None	Not Included
1.22 LEAKAGE	None	Not Included
1.23 MASTER RELAY TEST	None	Not Included
1.24 MEMBER(S) OF THE PUBLIC	None	Not Included
1.25 OFF-SITE AC POWER SOURCES	None	Not Included
1.26 OFFSITE DOSE CALCULATION MANUAL (ODCM)	5.6.2 Offsite Dose Calculation Manual (ODCM)	5.6.1, Offsite Dose Calculation Manual (ODCM)
1.27 OPERABLE- OPERABILITY	None	Not Included
1.28 OPERATING	None	Not Included
1.29 OPERATING CYCLE	None	Not Included
1.30 OPERATIONAL MODE-MODE	None	Not Included
1.31 PHYSICS TESTS	None	Not Included
1.32 PRESSURE BOUNDARY LEAKAGE	None	Not Included
1.32A PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	None	Not Included
1.33 PROCESS CONTROL PROGRAM (PCP)	None	Not Included
1.34 PROTECTION LOGIC CHANNEL	None	Not Included
1.35 PROTECTION SYSTEM	None	Not Included
1.36 PURGE-PURGING	None	Not Included
1.37 QUADRANT POWER TILT RATIO	None	Not Included
1.38 RATED THERMAL POWER	None	Not Included
1.39 REACTOR PRESSURE	None	Not Included
1.39 REFUELING CYCLE OR OUTAGE	None	Not Included
1.41 REPORTABLE EVENT	None	Not Included

TABLE SUMMARIZING THE DISPOSITION OF THE CTS IN THE ZION PDTS AND COMPARING THE DISPOSITION WITH THE MAIN YANKEE PDTS

Zion CTS Section/Specification	Equivalent Maine Yankee PDTS	Proposed Disposition in Zion PDTS*
1.42 SHUTDOWN MARGIN	None	Not Included
1.43 SITE BOUNDARY	None	Not Included
1.44 Previously deleted	N/A	N/A
1.45 SOURCE CHECK	None	Not Included
1.46 SURVEILLANCE FREQUENCY NOTATION	None	Not Included
1.47 THERMAL POWER	None	Not Included
1.48 UNIDENTIFIED LEAKAGE	None	Not Included
1.49 UNRESTRICTED AREA	None	Not Included
1.50 Previously deleted	N/A	N/A
1.51 VENTING	None	Not Included
Safety Limits / Limiting Safety System Settings		
1.1 / 2.1 Reactor Core	None	Not Included
1.2 / 2.2 Reactor Coolant System Pressure	None	Not Included
Limiting Conditions for Operation / Surveillance Requirements		
3.0.1 (LCO/Action Applicability)	LCO 3.0.1 (Applicability)	LCO 3.0.1 (Applicability)
3.0.2 (LCO/Action Compliance)	LCO 3.0.2 (Actions)	LCO 3.0.2 (Actions)
3.0.3 (Failure to comply -LCO/Action)	None	Not Included
3.0.4 (Mode change re. LCO/Action)	None	Not Included
3.0.5 (AC power availability)	None	Not Included
4.0.1 (Surveillance applicability)	SR 3.0.1 (Applicability)	SR 3.0.1 (Applicability)
4.0.2 (SR interval/extension)	SR 3.0.2 (Frequency)	SR 3.0.2 (Frequency)
4.0.3 (Failure to comply - SR)	SR 3.0.3 (Non-performance)	SR 3.0.3 (Non-performance)
4.0.4 (Mode change re. SR)	None	Not Included
4.0.5 (ISI &IST Surveillance Rules)	5.6.5 Inservice Testing Program	Not Included
3.1 / 4.1 Reactor Protection Instrumentation and Logic	None	Not Included
3.2 / 4.2 Reactivity Control and Power Distribution	None	Not Included
3.3 / 4.3 Reactor Coolant System (per unit)	None	Not Included
3.4 / 4.4 Safeguards Instrumentation and Control	None	Not Included
3.5 / 4.5 Reactor Containment Fan Coolers	None	Not Included
3.6 / 4.6 Containment Spray	None	Not Included
3.7 / 4.7 Steam Generator Emergency Heat Removal	None	Not Included
3.8 / 4.8 Emergency Core Cooling and Core Cooling Support	None	Not Included
3.9 / 4.9 Containment Isolation Systems	None	Not Included
3.10 / 4.10 Containment Structural Integrity	None	Not Included
3.11 / 4.11 Radioactive Liquids	None	5.6.3, Outdoor Storage Tank Radioactivity Monitoring Program

ATTACHMENT E

TABLE SUMMARIZING THE DISPOSITION OF THE CTS IN THE ZION PDTS AND COMPARING THE DISPOSITION WITH THE MAIN YANKEE PDTS

Zion CTS Section/Specification	Equivalent Maine Yankee PDTS	Proposed Disposition in Zion PDTS*
3.12 / 3.12 Gaseous Effluents	None	Not Included
3.13 / 4.13 Refueling Operations		
3.13.1 / 4.13.1 Core Reactivity	None	Not Included
3.13.2 / 4.13.2 Protection from Damaged Spent Fuel	None	Not Included
3.13.3 / 4.13.3 Containment Status	None	Not Included
3.13.4 / 4.13.4 Radiation Monitoring	None	Not Included
3.13.5 / 4.13.5 Refueling Equipment Operability	None	Not Included
3.13.6 / 4.13.6 (Refueling Actions)	None	Not Included
3.13.7 / 4.13.7 (Spent Fuel Pit Cooling System)	None	Not Included
3.13.8 / 4.13.8 (Fuel Inspection Program)	None	Not Included
3.13.9 / 4.13.9 Residual Heat Removal System Operation	None	Not Included
3.13.10 / 4.13.10 Water Level Reactor Vessel	None	Not Included
3.13.11 / 4.13.11 Water Level-Storage Pool	3.1.1 Fuel Storage Pool Water Level	3.1.1, Spent Fuel Pool Water Level
3.13.12 / 4.13.12 Previously deleted	N/A	N/A
3.13.13 / 4.13.13 Spent Fuel Pool Storage	4.2.3 Capacity	3.1.3, Spent Fuel Assembly Storage
3.13.14 / 4.13.12 Spent Fuel Storage Pool Boron Conc.	3.1.2 Fuel Storage Pool Boron Concentration	3.1.2, Fuel Storage Pool Boron Concentration
3.13.15 / 4.13.15 (Spec. 3.0.3 Non-applicability)	None	Not Included
3.13.16 / 4.13.16 (Spec. 3.0.4 Non-applicability)	None	Not Included
3.14 / 4.14 Plant Radiation Monitoring	None	Not Included
3.15 / 4.15 Auxiliary Electrical Power System	None	Not Included
3.16 Previously deleted	N/A	N/A
3.17 / 4.17 Ventilation	None	Not Included
3.18 / 4.18 Steam Generator Activity	None	Not Included
3.19 / 4.19 Failed Fuel Monitoring	None	Not Included
3.20 Previously deleted	N/A	N/A
3.21 Previously deleted	N/A	N/A
3.22 / 4.22 Shock Suppressors (Snubbers)	None	Not Included
3.23 Previously deleted	N/A	N/A
3.24 / 4.24 Sealed Source Contamination	None	Not Included
Design Features		
5.1 Site	4.1.1 Site Description	4.1.1, Site Description
5.2 Reactor Coolant System	None	Not Included
5.3 Reactor Core	None	Not Included
5.4 Containment System	None	Not Included
5.5 Fuel Storage		
5.5.1 New Fuel Storage	None	4.2, Fuel Storage
5.5.2 Spent fuel storage	4.2 Fuel Storage	4.2, Fuel Storage

ATTACHMENT E

TABLE SUMMARIZING THE DISPOSITION OF THE CTS IN THE ZION PDTS AND COMPARING THE DISPOSITION WITH THE MAIN YANKEE PDTS

Zion CTS Section/Specification	Equivalent Maine Yankee PDTS	Proposed Disposition in Zion PDTS*
5.6 Seismic Design	None	Not Included
Administrative Controls		
6.1 Organization		
6.1.1 (Onsite and Offsite organizations)	5.2.1 General Organizational Requirements	5.2.1, General Organizational Requirements
6.1.2 Previously Deleted	N/A	N/A
6.1.3 (Shift manning)	5.2.2 Unit Staff	5.2, Shift Staff
6.1.4 (Management and operating staff qualifications)	5.3 Unit Staff Qualifications	5.3, Staff Qualifications
6.1.5 (Retraining and replacement of station personnel)	None	Not Included
6.1.6 (Retraining interval)	None	Not Included
6.1.7 (Certified Fuel Handler training and retraining program)	5.4 Training	5.4, Training
6.2 Procedures and Programs		
6.2.1 (Procedures)	5.5.1 Procedures	5.5.1, Procedures
6.2.1.a (Procedures per R.G. 1.33, App. A)	5.5.1.a (R.G. 1.33 procedures)	5.5.1.a (R.G. 1.33 procedures)
6.2.1.b (Emergency Operating procedures per NUREG-0737, Sup. 1, and G.L. 83-33)	None	Not Included
6.2.1.c (Security Plan procedures)	None	Not Included
6.2.1.d (GSEP procedures)	5.5.1.b (Emergency Plan implementation)	Not Included.
6.2.1.e (PCP procedures)	None	Not included
6.2.1.f (ODCM procedures)	5.5.1.f (Programs specified in Specification 5.6)	5.5.1.c (Programs specified in Specification 5.6)
6.2.1.g (Fire Protection procedures)	5.5.1.e (Fire Protection procedures)	5.5.1.b (Fire Protection procedures)
6.2.1.h (Post Accident Sampling procedures)	None	Not Included
6.2.1.i (Overtime procedures)	5.2.2.e (Overtime procedures)	5.2.2.e (Overtime procedures)
6.2.2.A (Radiation Control procedures)	5.6.1 Radiation Protection Program	Not Included
6.2.2.B. High Radiation Area	5.8 High Radiation Area	5.8, High Radiation Area
6.2.3 (Technical review and control of procedures)	None	Not Included
6.2.4 (Temporary changes to procedures)	None	Not Included
6.2.5 (GSEP Drills)	None	Not Included
6.2.6 Programs	5.6 Programs and Manuals	5.6, Programs and Manuals
6.2.6.A Radioactive Effluent Controls Program	5.6.3 Radioactive Effluent Controls Program	5.6.2, Radioactive Effluent Controls Program
6.2.6.B Radiological Environmental Monitoring Program	None	Not Included
6.3 Actions to be Taken in the Event of a Reportable Event in Plant Operation	None	Not Included
6.4 Previously deleted	N/A	N/A
6.5 Plant Operating Records	None	Not Included

ATTACHMENT E

TABLE SUMMARIZING THE DISPOSITION OF THE CTS IN THE ZION PDTS AND COMPARING THE DISPOSITION WITH THE MAIN YANKEE PDTS

Zion CTS Section/Specification	Equivalent Maine Yankee PDTS	Proposed Disposition in Zion PDTS*
6.6 Reporting Requirements	5.7 Reporting Requirements	5.7, Reporting Requirements
6.6.1 Routine Reports		
6.6.1.A Startup Report	None	Not Included
6.6.1.B Annual Occupational Exposure Report	5.7.1 Occupational Exposure Report	5.7.1, Occupational Exposure Report
6.6.1.C Annual Radiological Environmental Operating Report	5.7.2 Annual Radiological Environmental Operating Report	5.7.2, Annual Radiological Environmental Operating Report
6.6.1.D Radioactive Effluent Release Report	5.7.3 Radioactive Effluent Release Report	5.7.3, Radioactive Effluent Release Report
6.6.1.E Monthly Operating Report	None	Not Included
6.6.1.F Core Operating Limits Report (COLR)	None	Not Included
6.6.1.G Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)	None	Not Included
6.6.2 Reportable Events	None	Not Included
6.6.3 Unique Reporting Requirements		
6.6.3.A Previously deleted	N/A	N/A
6.6.3.B Special Reports		
6.6.3.B.a In-Service Inspection Evaluation	None	Not Included
6.6.3.B.b Previously deleted	N/A	N/A
6.6.3.B.c Containment Building Structural Testing Report (ILRT, Tendon)	None	Not Included
6.6.3.B.d Changes to the Offsite Dose Calculation Manual (ODCM)	5.6.2 Offsite Dose Calculation Manual	5.6.1, Offsite Dose Calculation Manual
6.6.3.B.e Waukegan Regional Airport Expansion Plans	None	Not Included
6.6.3.B.f Low Temperature Overpressure Protection System Operation	None	Not Included
6.6.3.B.g Primary Coolant Specific Activity	None	Not Included
6.6.3.B.h Pressurizer PORV or Safety Valve Failure to Close	None	Not Included
6.6.3.B.i Pressurizer PORV or Safety Valve challenges	None	Not Included
6.6.3.B.j (not used)	N/A	N/A
6.6.3.B.k Steam generator tube inspection and/or plugging	None	Not Included
6.6.3.B.l Emergency Core Cooling System (ECCS) actuation and injection when RCS temp > 350 F	None	Not Included
6.6.3.B.m Diesel generator failures	None	Not Included
6.6.3.B.n Post Accident Radiation monitor inoperable greater than 7 days	None	Not Included
6.7 Offsite Dose Calculation Manual (ODCM)	5.6.2 Offsite Dose Calculation Manual	5.6.1, Offsite Dose Calculation Manual
6.8 Flooding Protection	None	Not included
6.9 Process Control Program (PCP)	None	Not included

ATTACHMENT E

TABLE SUMMARIZING THE DISPOSITION OF THE CTS IN THE ZION PDTS AND COMPARING THE DISPOSITION WITH THE MAIN YANKEE PDTS

Zion CTS Section/Specification	Equivalent Maine Yankee PDTS	Proposed Disposition in Zion PDTS*
6.10 Containment Leakage Rate Testing Program	None	Not Included

* In some instances, CTS specifications that have been included in the PDTS have been modified to be applicable to the units' permanently defueled status.

ZION STATION
LICENSE AMENDMENT REQUEST NUMBER 98-06;
PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

ATTACHMENT F

REFERENCES FOR PROPOSED AMMENDMENT

1) Letter from M. K. Webb, NRC, to M. J. Meisner, Maine Yankee Atomic Power Company, dated March 30, 1998, Issuance of License Amendment No. 161, Permanently Defueled Technical Specifications

2) NUREG 1431, Standard Technical Specifications for Westinghouse Plants, Revision 1, dated April 7, 1995.

3) Letter from O. D. Kingsley, ComEd, to U.S. NRC, dated February 13, 1998, Certification of Permanent Cessation of Operations

4) Letter from O. D. Kingsley, ComEd, to U. S. NRC, dated March 9, 1998, Certification of Permanent Fuel Removal

5) Letter from C. Y. Shiraki, U. S. NRC, to O. D. Kingsley, ComEd, dated December 19, 1997, Issuance of Amendments 178/165 to Facility Operating Licenses DPR-39 and DPR-48, Improved Technical Specifications

6) Letter from R. R. Assa, U. S. NRC, to O. D. Kingsley, ComEd, dated July 24, 1998, Issuance of Amendments 179/166 to Facility Operating Licenses DPR-39 and DPR-48, Restoration of Custom Technical Specifications, Reinstatement of Previous License Conditions, Changes in Management Titles and Responsibilities, Use of Certified Fuel Handlers, Changes to Shift Staffing and Crew Composition, and Elimination of Verbiage Implying the Units are Operational

7) Letter from M. J. Meisner, Maine Yankee, to U. S. NRC, dated October 20, 1997, under Docket 50-309, concerning Proposed Technical Specification Change No. 207, Permanently Defueled Technical Specifications

8) Letter from J. C. Brons, ComEd, to U. S. NRC, dated March 16, 1998, Request for Approval of Certified Fuel Handler Training and Retraining Program

9) Letter from R. R. Assa, U. S. NRC, to O. D. Kingsley, ComEd, dated July 20, 1998, Accepting ComEd's Proposed Certified Fuel Handler Program

ZION STATION
LICENSE AMENDMENT REQUEST NUMBER 98-06;
PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

ATTACHMENT F

REFERENCES FOR PROPOSED AMMENDMENT

10) Letter from John C. Brons, ComEd, to U. S. NRC, dated March 30, 1998, Application for Amendment to Restore Custom Technical Specifications, Reinstate Previous License Conditions, Change Management Titles and Responsibilities, Use Certified Fuel Handlers, Change Shift Staffing and Crew Composition, and Eliminate Verbiage Implying the Units are Operational

11) Letter from J. C. Brons, ComEd, to U. S. NRC, dated March 12, 1998, Application for Exemption from the 24 Month Update Requirement of 10 CFR 50.71

ZION STATION
LICENSE AMENDMENT REQUEST NUMBER 98-06;
PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

ATTACHMENT G

BASIS FOR REDUCTION IN MINIMUM NUMBER OF NON-CERTIFIED OPERATORS REQUIRED ON SHIFT

Com Ed considers that, consistent with the Maine Yankee PDTS, two operators (Certified fuel handler and non-certified operator) constitute adequate shift manning to ensure the safe storage and handling of nuclear fuel at Zion during routine, non-routine, and abnormal conditions.

Routine Conditions

Although Zion is a two unit site, the units share a single spent fuel pool with a common cooling system. Major support equipment such as service water pumps, component cooling pumps and heat exchangers, and makeup sources are located together in common areas regardless of unit assignment. Consequently, the routine functions of periodically monitoring pool level and the status of support equipment, and adding makeup for evaporative losses can readily be accomplished by one operator while the other operator staffs the control room. The number of individuals needed to perform these functions is no greater for a two unit site than for a single unit site.

Non-Routine Conditions

With five other nuclear plants, ComEd can augment the minimum shift crew if necessary to support non-routine operations (i.e. planned operations other than the routine monitoring of stored fuel). If movement of fuel or other components in the spent fuel pool were to be undertaken, additional personnel would be assigned on shift to assist with the operation. For example, during recent operations in and around the spent fuel pool in support of shipping unirradiated fuel for reprocessing, a dedicated fuel movement crew was used in addition to the normal minimum shift crew. Additional personnel would also be assigned if significant operations not involving the spent fuel pool are undertaken, as is currently being done to support synchronous condenser operation.

Abnormal Conditions

As described in Attachment A to this letter, the remaining credible accidents and operational events include only two that could result in a release of radioactivity; a fuel handling accident and a low level radioactive waste handling accident. The analyses of both these accidents have shown that the potential releases are within the requirements of 10 CFR100 with no operator action. The operator functions during these or similar events would likely be limited to initial communication and co-ordination activities. The tasks required would be no greater for a two unit site than for a single unit site. Therefore, two operations personnel would be adequate shift manning if such events were to occur.

The operational events described in Attachment A to this letter involve loss of cooling to the spent fuel pool at normal and reduced levels. As a result of the long heatup times for the spent fuel pool, there is ample time to restore cooling prior to reaching a limiting condition, even if additional personnel needed be called in to respond to these events. Therefore, the number or personnel needed on shift to respond to such events is not critical and is no greater for a two unit site than for a single unit site.

Based on the above, ComEd considers that the minimum shift crew at Zion need be no larger than at single unit permanently shutdown sites such as Trojan and Maine Yankee, and that as a licensee of multiple nuclear power stations, ComEd may be better able to augment the minimum shift crew for non-routine operations.

ZION STATION
LICENSE AMENDMENT REQUEST NUMBER 98-06;
PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

ATTACHMENT H

LIST OF COMMITMENTS IDENTIFIED IN THIS AMENDMENT REQUEST

The following table identifies those actions committed to by ComEd in this document. Any other actions discussed in this submittal represent intended or planned actions by ComEd. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify Mr. Robert Godley, Zion Station Regulatory Assurance Manager, of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed Date
The following current License Conditions and CTS requirements will be relocated to the DSAR: License Condition, 2.C.(7)(b) (Control Of Loads Over the SFP) CTS 3.24/4.24 (Sealed Source Contamination) CTS 5.1 Descriptions of the exclusion area and the low population zone CTS 5.5 Information on the capacity of the new fuel racks, the number of sections and rows, the distance between each section, the U235 gram/centimeter loading, and the vault drain CTS 5.5 Information concerning the SFP stainless steel liner and vertical array, the U235 gram/centimeter loading, and the figure showing a diagram of the SFP CTS 5.6 The seismic design features that are relevant with the units permanently shutdown and defueled	In the initial issue of the DSAR prior to implementation of the PDTS
The prohibition of temporary procedure changes will continue. (CTS 6.2.4)	At least until a change is made to the Quality Assurance Manual
The CTS 6.1.3 descriptions of responsibilities of the Decommissioning Plant Manager and the Decommissioning Operations Manager regarding the Fire Protection Program will be relocated to site documents.	Prior to implementation of the PDTS

ATTACHMENT H

Commitment	Committed Date
Adequate review requirements for changes to the PCP will be incorporated into the ODCM. (CTS 6.9)	Prior to implementation of the PDTS
The gas decay tanks will be vented and removed from service. (CTS 3.12/4.12)	Prior to implementation of the PDTS