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

August 7, 1991

Docket No. 50-409

Mr. William L. Berg
General Manager
Dairyland Power Cooperative
2615 East Avenue South
La Crosse, Wisconsin 54602-0817

Dear Mr. Berg:

SUBJECT: ORDER TO AUTHORIZE DECOMMISSIONING AND AMENDMENT NO. 66 TO POSSESSION ONLY LICENSE NO. DPR-45 FOR LA CROSSE BOILING WATER REACTOR

The Commission has issued the enclosed order to authorize decommissioning of the La Crosse Boiling Water Reactor (LACBWR) and has also issued Amendment No. 66 to Possession Only License No. DPR-45 for LACBWR. The order and amendment responds to your application of December 21, 1987 as revised February 22, 1988, September 9, 1988, September 30, 1988, January 26, 1989, March 28, 1989, June 6, 1989, October 3, 1989, July 25, 1990, May 10, 1991, and July 25, 1991.

A Notice of Consideration of Issuance of Amendment to License and Opportunity for Hearing related to the requested action was published in the FEDERAL REGISTER on April 8, 1988 (53 FR 11718). No comments or requests for hearing were received.

A copy of the related Safety Evaluation and Environmental Assessment supporting Amendment No. 66 are enclosed. Also enclosed is a copy of the Notice of Issuance of Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on August 7, 1991 (56 FR 37574).

Sincerely,

Peter B. Erickson

Peter B. Erickson, Senior Project Manager
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Order Authorizing Decommissioning
2. Amendment No. 66 to License No. DPR-45
3. Safety Evaluation
4. Environmental Assessment
5. Notice of EA

cc w/enclosures:
See next page

Mr. James W. Taylor
Dairyland Power Cooperative

Docket No. 50-409
La Crosse Boiling Water Reactor

cc:

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

DAIRYLAND POWER COOPERATIVE

DOCKET NO. 50-409

LA CROSSE BOILING WATER REACTOR (LACBWR)

AMENDMENT TO POSSESSION ONLY LICENSE

Amendment No. 66
License No. DPR-45

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

   A. The application for amendment filed by Dairyland Power Cooperative (the licensee), dated December 21, 1987 as revised February 22, 1988, September 9, 1988, September 30, 1988, January 26, 1989, March 28, 1989, June 6, 1989, October 3, 1989, July 25, 1990, May 10, 1991, and July 25, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Possession Only License No. DPR-45 is hereby amended by revising the indicated paragraphs as follows:

   Paragraph 2.C.(2) to read:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 66, are hereby incorporated in the license. The licensee shall maintain the facility in accordance with the Technical Specifications.

   Paragraph 2.E. to read:

   2.E. This license shall expire at midnight March 29, 2031.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Richard F. Dudley, Jr. for

Seymour H. Weiss, Director
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: August 7, 1991

ENCLOSURE TO LICENSE AMENDMENT NO. 66

POSSESSION ONLY LICENSE NO. DPR-45

DOCKET NO. 50-409

Replace all of the pages of the Appendix A Technical Specificiations with the enclosed pages.

Tech Spec

DOCKET NO. 50-409

APPENDIX A

LICENSE NO. DPR-45

TECHNICAL SPECIFICATIONS

FOR

LA CROSSE BOILING WATER REACTOR

Amendment No. 66

TABLE OF CONTENTS

1. SITE

1.1 LOCATION

The La Crosse Boiling Water Reactor (LACBWR) is located on the east bank of the Mississippi River, approximately 19 miles south of the city of La Crosse, Wisconsin, and 1 mile south of the Village of Genoa, Wisconsin.

1.2 PRINCIPAL ACTIVITIES

The principal activity carried on at the La Crosse Boiling Water Reactor shall be possession of the facility. The major activities shall include the monitoring of the reactor and plant equipment, storage and handling of the reactor fuel, maintenance of systems required for safe storage activities, monitoring effluents, and analyzing environmental samples to assure the health and safety of the public.

2. DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. When these terms appear in capitalized type, the following definitions apply in these Technical Specifications.

ACTION

ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a real or simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be isolated are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.

b. The freight door is closed,

c. Each air lock is OPERABLE,

d. The containment leakage rates are within the limit, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, o-rings) is OPERABLE.

CONTROLLED AREA

A Controlled Area is an area for which access shall be controlled because of actual presence of uncontained radioactive material. Controlled areas shall be designated by radiation control signs and markers in accordance with 10 CFR 20.203. These controlled areas shall include the Turbine Building beyond the change room door, the Containment Building, the Waste Treatment Building, and any other areas so marked by radiation signs and markers.

EFFLUENT RELEASE BOUNDARY

The Dairyland Power Cooperative property line within the 1109 ft. (338m) radius EXCLUSION AREA is the EFFLUENT RELEASE BOUNDARY. (See Figure 1, Sec. 4.)

EXCLUSION AREA

The EXCLUSION AREA is defined as the area within an 1109 ft. (338m) radius from the centerline of the Containment Building. This was the area established per 10 CFR 100 as the EXCLUSION AREA for plant siting and operation.

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table of Surveillance Frequency Notation.

FUEL HANDLING

FUEL HANDLING shall be the movement of any irradiated fuel within the Containment Building. Suspension of FUEL HANDLING shall not preclude completion of movement of the fuel to a safe conservative position.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include any person who is not occupationally associated with the utility. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or make deliveries. This category does include persons who use portions of the site for recreational or other purposes not associated with the utility.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used for the calculation of offsite doses due to radioactive gaseous and liquid effluents and for the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints. It shall describe the radiological environmental monitoring program.

OPERABLE-OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, a normal or an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61 and Federal and State regulations and other requirements governing the transportation and disposal of the radioactive waste.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 6.8.3. A Licensee Event Report shall be submitted for REPORTABLE EVENTS.

RESTRICTED AREA

A RESTRICTED AREA shall be any area within the EXCLUSION AREA, access to which is controlled by the licensee for purposes of protection of individuals from exposure to ionizing radiation and radioactive materials.

SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area not controlled by the licensee for purposes of protection of MEMBERS OF THE PUBLIC from exposure to ionizing radiation and radioactive materials.

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
A	At least once per 12 months.
R	At least once per 18 months.
P	Completed prior to each use or release.
N.A.	Not applicable.

3. APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the specified applicable condition for each specification.

3.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.3 Entry into specified applicability state shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted.

SURVEILLANCE REQUIREMENTS

3.4 Surveillance Requirements shall be applicable during the specified applicable conditions for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

3.5 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

3.6 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment or on equipment not required to be OPERABLE.

3.7 Entry into a specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 4/5.

3.1 This specification defines the applicability of each specification in terms of specified applicability conditions and is provided to delineate specifically when each specification is applicable.

3.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.3 This specification provides that entry into specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that activities are not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when performance of activities with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

3.4 This specification provides that surveillance activities necessary to ensure that the Limiting Conditions for Operation are met and will be performed during the specified applicability conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the specified applicability conditions are provided in the individual Surveillance Requirements.

3.5 The provision of this specification provides an allowable tolerance for performing surveillance activities beyond those specified in the nominal surveillance interval. This tolerance is necessary to provide flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency requirement does not negate these allowable tolerances for performing surveillance activities.

The tolerance value is sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

BASES - (Cont'd)

3.6 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Condition for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

3.7 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

4/5. PERFORMANCE REQUIREMENTS

4.1 FUEL STORAGE AND HANDLING

4.1.1 GENERAL FUEL STORAGE AND HANDLING REQUIREMENTS

4.1.1.1 Irradiated fuel elements shall be stored underwater in spent fuel storage racks that are positioned on the bottom of the Fuel Element Storage Well, in approved onsite dry storage containers, or in an approved shipping cask.

4.1.1.2 During the handling of irradiated fuel elements that have been operated at power levels greater than 1 Mwt, the depth of water in the reactor upper cavity and/or the Fuel Element Storage Well shall be at least 2 feet above the active fuel, and only one fuel element will be moved at a time.

4.1.1.3 With the exception of a spent fuel shipping cask, the core spray bundle, the transfer canal shield plug and the other components and fixtures that are normally located and used within the storage well, no objects heavier than a fuel assembly shall be handled over the Fuel Element Storage Well.

4.1.2 FUEL ELEMENT STORAGE WELL

LIMITING CONDITION FOR OPERATION

The Fuel Element Storage Well (FESW) shall meet the following requirements:

a. The Fuel Element Storage Well water level shall be at least 16 feet above any irradiated fuel stored in the spent fuel storage racks, and

b. Water in the storage well shall be maintained at a temperature $\leq$ 150°F.

APPLICABILITY: At all times.

ACTION:

a. With water level less than 16 feet above any irradiated fuel stored in the Fuel Element Storage Well storage racks, take immediate action to restore water level and suspend all operations involving FUEL HANDLING.

b. With water temperature in the storage well above 150°F, take actions to reduce water temperature to $\leq$ 150°F within 24 hours and suspend any evolutions involving FUEL HANDLING.

SURVEILLANCE REQUIREMENTS

5.1.2.1 The Fuel Element Storage Well water level and FESW System water temperature shall be monitored at least once per 24 hours.

5.1.2.2 The Fuel Element Storage Well water level indication channel shall be calibrated at least once per 18 months.

4.1.3 FUEL ELEMENT STORAGE WELL WATER CHEMISTRY

LIMITING CONDITION FOR OPERATION

The Fuel Element Storage Well water shall meet the following requirements:

a. Chloride concentration-------- $\leq$ 0.5 ppm
b. pH---------- 5.3 - 8.6
c. Conductivity---------- $\leq$ 10 umho/cm
d. Gross beta-gamma activity----- $\leq$ 0.1 uCi/ml

APPLICABILITY: Whenever fuel is stored in the Fuel Element Storage Well.

ACTION:

With the Fuel Element Storage Well water chemistry or radiochemistry limits exceeded, initiate action to restore water quality to within the limits.

SURVEILLANCE REQUIREMENTS

5.1.3.1 The Fuel Element Storage Well chemistry and radiochemistry parameters specified in this section shall be determined at least once every 7 days.

5.1.3.2 If gross beta-gamma activity exceeds the specified limit and cleanup efforts cannot restore the activity to below the specified limit within 7 days, a 30-day report shall be made to the NRC.

5.1.3.3 If the gross beta-gamma activity exceeds 1.0 uCi/ml at the time of sampling, an immediate report to the NRC shall be made.

4.1.4 FUEL ELEMENT STORAGE WELL WATER SUPPLY

LIMITING CONDITION FOR OPERATION

At least one of the following Fuel Element Storage Well water supplies shall be OPERABLE:

a. The Demineralized Water Tank with a minimum water level of two feet, (approximately 5000 gallons) or

b. The Overhead Storage Tank with a minimum contained water volume of 5,000 gallons.

APPLICABILITY: At all times.

ACTION:

With neither the Demineralized Water Tank nor the Overhead Storage Tank OPERABLE, restore level in at least one tank within 7 days.

SURVEILLANCE REQUIREMENTS

5.1.4.1 The Demineralized Water Tank shall be demonstrated OPERABLE by verifying the minimum water level in the tank at least once per 7 days.

5.1.4.2 The Overhead Storage Tank shall be demonstrated OPERABLE by verifying the minimum contained water volume at least once per 7 days.

BASES

4/5.1 Fuel Storage and Handling

Spent fuel storage is provided in the spent fuel storage racks located in the Fuel Element Storage Well within the Containment Building. The spent fuel storage racks are designed with a nominal 7.0 inch center-to-center distance between fuel assemblies in each individual rack assembly with a boron containing poison slab between each storage location to ensure K_{eff} of $\leq$ 0.95 when flooded with unborated water. Fuel stored in the storage well is restricted to fuel with stainless steel cladding which has a U-235 loading of $\leq$ 22.6 grams per axial centimeter. A fuel handling system is provided which is capable of remotely handling fuel assemblies one at a time.

The Fuel Element Storage Well System is capable of maintaining water temperature $\leq$ 150°F and maintaining water quality within the established limits.

Minimum water coverage limits above fuel stored in the storage racks and fuel being handled have been established to provide adequate shielding to protect personnel and to provide adequate cooling. Limits to the handling of heavy objects over the Fuel Element Storage Well have been established to reduce the probability of a heavy load drop into the storage well.

A minimum Fuel Element Storage Well water supply has been established to ensure that sufficient water is available onsite to provide short-term makeup to the storage well while an alternate supply is being established if needed.

4.2 CONTAINMENT BUILDING

4.2.1 CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION
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CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: During FUEL HANDLING.

ACTION:

If CONTAINMENT INTEGRITY does not exist, suspend all FUEL HANDLING within 2 hours.

SURVEILLANCE REQUIREMENTS
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5.2.1.1 Containment Building gasketed closures and ventilation system closures which have been subjected to maintenance, repair or other operations which might affect their performance shall, before any subsequent operation for which containment integrity is required, be tested for leak tightness using the soap-bubble technique (or other method of equivalent sensitivity). This test shall be performed using a pressure differential no less than 0.5 psi and the results shall be used as a guide in evaluating leakage.

5.2.1.2 Any new or existing containment penetrations which are placed in or removed from service shall be tested for leak tightness using the soap-bubble technique (or other method of equal sensitivity) at a pressure differential of at least 10 psi. The test shall be performed before any subsequent FUEL HANDLING is conducted.

5.2.1.3 The door seals on the containment personnel and emergency airlocks will be visually inspected for degradation prior to FUEL HANDLING if not inspected within the preceding 72 hours.

5.2.1.4 The door seals on the containment personnel and emergency airlocks will be replaced periodically in accordance with manufacturer's recommendations.

SURVEILLANCE REQUIREMENTS - (Cont'd)

5.2.1.5 Individual leak detection tests shall be performed on Containment Isolation Valves and the following penetrations at a pressure of at least 10 psig prior to FUEL HANDLING if not performed within the preceding 90 days, except for the freight door. The freight door shall be tested following each closure prior to FUEL HANDLING. The containment penetrations shall be tested using the soap-bubble technique (or other methods of equivalent sensitivity) or by determining the rate of pressure loss of pneumatically pressurized test chambers. The combined leakage rate for all penetrations and valves tested shall be no greater than 50 SCFH. The penetrations to be tested are:

- the electrical penetrations,
- the reactor building spray valve shaft penetration,
- the freight door,
- the main steam line penetration,
- the feedwater line penetration,
- the heating steam line and condensate return penetration,
- the containment building airlocks, and
- the flanges of the ventilation system inlet and exhaust ducts.

5.2.1.6 Containment Isolation Valves shall be tested prior to FUEL HANDLING if not tested within the preceding 60 days to demonstrate automatic closure on the conditions specified in Table 4.2.1 unless the Containment Isolation Valve is deactivated in its closed position or the penetration is isolated by at least one manual valve or blind flange.

TABLE 4.2.1

CONTAINMENT ISOLATION VALVES

VALVE	CHANNEL, SENSOR, OR CONDITION	SETPOINT
Containment Ventilation Inlet and Outlet Dampers	A	C
	B	D
	Loss of Control Air	NA
	Loss of Electrical Power Supply	NA
Containment Vent Header Valve	A	C
	B	D
Decay Heat System Blowdown Valve	A	C
	B	D
Heating Steam Condensate Return Valve	A	C
	B	D
Retention Tank Pump Discharge Valve	A	C
	B	D

TABLE NOTATION:

A. Reactor Containment Building Ventilation Particulate or Gaseous Activity Monitor
B. Fuel Element Storage Well Level *
C. $\leq$ Radiation Levels which Correspond to the limits of Specification 4.7.2.2.
D. $\geq$ 690' MSL

* This installation will be accomplished as soon as possible, but no later than six (6) months from the date of issue of the SAFSTOR T.S.

4.2.2 CONTAINMENT VENTILATION DAMPERS

LIMITING CONDITION FOR OPERATION

The Containment ventilation inlet and outlet dampers shall be OPERABLE with isolation times of less than or equal to 10 seconds.

APPLICABILITY: Whenever CONTAINMENT INTEGRITY (Specification 4.2.1) is required.

ACTION:

a. With one or more of the above ventilation damper(s) inoperable, suspend FUEL HANDLING or isolate the affected penetration, with a deactivated automatic valve secured in its closed position or with a blind flange, within 2 hours.

b. The provisions of Specification 3.3 are not applicable if the affected penetration is isolated.

SURVEILLANCE REQUIREMENTS

5.2.2.1 The ventilation dampers shall be demonstrated OPERABLE prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control, or power circuit by performance of a cycling test, and verification of isolation time.

5.2.2.2 The isolation time of each above damper shall be determined to be within its limit when tested pursuant to Specification 5.2.1.6.

5.2.2.3 The seat rings of the ventilation inlet and outlet dampers shall be replaced at least once per 5 years.

5.2.2.4 The Containment ventilation system dampers shall be subjected to quarterly leakage tests, in addition to the tests required by Specification 5.2.1.5. The tests shall be conducted at an initial pressure of at least 10 psig. Excessive degradation is determined not to exist and the isolation valve(s) is considered operable if pressure decreases by less than 5 psi in a ten minute test period.

4.2.3. CONTAINMENT VESSEL

LIMITING CONDITION FOR OPERATION

The Containment Vessel shall be protected by the following requirements:

a. At least one Containment Building vacuum breaker shall be set to relieve at a differential (external-over-internal) pressure not exceeding 0.5 psi.

b. The Containment Building steel shell temperature shall be greater than 0°F when the Containment Building pressure exceeds 10.4 psig.

APPLICABILITY: At all times.

ACTION:

a. With no vacuum breakers OPERABLE, restore at least one vacuum breaker to service within 24 hours.

b. With the Containment Building steel shell temperature less than 0°F with pressure exceeding 10.4 psig, restore the temperature and/or pressure to within the limits within 30 minutes and perform an engineering determination of the effects of the out-of-limit condition on the fracture toughness properties of the Containment Building.

SURVEILLANCE REQUIREMENTS

5.2.3.1 The Containment Building vacuum breakers shall be tested for proper operation at least once every 18 months.

5.2.3.2 Containment Building steel shell temperature shall be monitored every 4 hours when Containment Building pressure exceeds 10 psig.

4.2.4 VENTILATION SYSTEM EXHAUST

LIMITING CONDITION FOR OPERATION

The Containment Building Ventilation System exhaust shall be through particulate filters.

APPLICABILITY: Whenever the Containment Ventilation outlet dampers are open.

ACTION:

With Containment Building Ventilation System exhaust being discharged without filtration, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report which discusses the circumstances and what action will be taken to prevent a recurrence.

SURVEILLANCE REQUIREMENTS

5.2.4 Cumulative dose contributions from gaseous releases are calculated per Sections 5.7.2.3 and 5.7.2.4.

4/5.2.1 Containment Integrity

The containment vessel was originally designed to be capable of containing an internal pressure of 52 psig at 280°F. The established containment leakage rate corresponded to a leakage rate that, per 24-hour period, did not exceed 0.1 percent by weight of a steam-air mixture at 273°F, 28.5 psig, and a steam-to-initial-air ratio of 2.2. Leakage testing was conducted at 52 psig; the maximum allowable containment leakage was 50 SCFH.

With the plant in the SAFSTOR condition, there is no longer a postulated accident that would result in containment pressurization. The requirement for CONTAINMENT INTEGRITY during fuel handling has been retained. Leakage testing is conducted at a lower pressure, since there would be no pressure head during any postulated accident.

Containment isolation signals are based on high Containment Building ventilation activity and low Fuel Element Storage Well level since the irradiated fuel is stored in the storage well in the Containment Building and these conditions can be indicative of a postulated accident.

The design bases for containment vessel penetrations remains as originally established.

Personnel and Emergency Airlocks: The airlock doors shall be held closed by locking bars or locking rings which clamp against locking lugs on the doors, and these doors shall be sealed by means of gaskets. The doors of each airlock shall be mechanically interlocked so that one door must be completely closed and sealed before the other door can be opened. The door operation shafts shall be sealed by mechanical seals at both the inner and outer door frames.

Freight Door: The freight door shall be sealed by means of gaskets and shall be held closed by means of hinged bolts and lugs.

Electrical Penetrations: Each cable penetration shall have two gas-tight seals to prevent leakage from the containment vessel. Test chambers shall be provided to permit pressurized leak testing between the seals. Closures provided for electrical penetration chambers shall be welded or shall be bolted and sealed by gaskets.

Mechanical Penetrations: The handwheel shaft for the building spray system shall be provided with two gas-tight seals to prevent leakage from the containment vessel.

BASES - (Cont'd)

Piping and Ventilation Duct Penetrations: Piping and duct penetrations shall be sealed by means of welding directly to the containment vessel wall or by means of corrugated-type expansion joints which shall be welded to the pipe at one end and to a containment vessel nozzle at the other end. All piping and duct penetrations, other than spares which have welded pipe caps, shall conform to the following:

a. Two valves in series shall be provided at containment penetrations for piping connected directly to the primary system or for piping, other than vacuum breaker lines which is open both to the building atmosphere and to the outside atmosphere. At least one of these valves shall be capable of closing automatically, unless one valve is normally closed during plant operation. The second valve shall be operable either from the control room or from some other location which would be accessible after an accident.

b. Two dampers shall be provided on each of the Containment Building ventilation penetrations. The dampers shall be capable of being closed automatically.

Testing is required following any placement of new penetrations in service or after any work on an existing penetration that might affect its sealing capability prior to FUEL HANDLING.

4/5.2.2 Containment Ventilation Dampers

The Nuclear Regulatory Commission requested that similar Technical Specifications per Generic Item B-24 and NUREG-0737, Item II.E.4.2, be submitted to help assure operability of containment ventilation dampers. The ACTION statement has been modified due to the plant's permanent shutdown. DPC had previously committed to replace the containment ventilation inlet and outlet dampers' resilient sealing material at least once each 5 years until such time as additional in-situ data can be accumulated to justify a longer interval. If in-situ data is accumulated which supports a longer seal replacement interval, a change to Specification 5.2.2.3 may be requested. The additional leakage test was established to detect any gross degradation of the dampers.

4/5.2.3 Containment Vessel

The containment vessel was designed to the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII (1962 Edition) as modified by the nuclear code cases applicable as of June 1962. The vessel plate is ASTM A201, Grade B steel, conforming to the test requirements of ASTM A300. The temperature/pressure limit was established to assure the fracture toughness properties of the vessel are maintained. The containment vessel is capable of withstanding an external-over-internal pressure of 0.5 psi. Therefore, a vacuum relief system is provided which is capable of limiting the external-over-internal pressure to 0.5 psi.

BASES - (Cont'd)

4/5.2.4 Ventilation System Exhaust

Filtration of the Containment Building Ventilation System exhaust is required to reduce the amount of radioactive particulates being released to the environment. This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50.

4.3 ELECTRICAL POWER SYSTEMS

4.3.1 A.C. SOURCES

LIMITING CONDITION FOR OPERATION

As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. The physically independent circuit between the offsite transmission network and the onsite power distribution system and at least one electrical feeder from 2400-volt bus 1A or 1B energizing either, but not both, essential switchgear buses.

b. Diesel Generator 1A or 1B, aligned to, but not energizing, the same essential switchgear bus, with:

   1. A day tank containing a minimum of 80 gallons of fuel for diesel generator 1A or 170 gallons of fuel for Diesel Generator 1B.

   2. A fuel storage system containing a minimum of 200 gallons of fuel for Diesel Generator 1A or 2500 gallons of fuel for Diesel Generator 1B.

   3. A fuel transfer pump.

APPLICABILITY: At all times.

ACTION:

With less than the above required A.C. electrical power sources OPERABLE, suspend all operations involving FUEL HANDLING until at least the minimum required power sources are OPERABLE.

SURVEILLANCE REQUIREMENTS

5.3.1.1 The physically independent circuit between the offsite transmission network and the onsite power distribution system and the electrical feeds from 480-volt bus 1A and 1B to the associated essential switchgear bus shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability.

SURVEILLANCE REQUIREMENTS - (Cont'd)

5.3.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel generator starts from ambient condition, and reaches rated bus voltage in less than or equal to 20 seconds.

5. Verifying the diesel generator is loaded with the design test load and operates for greater than or equal to 60 minutes.

6. Verifying the diesel generator is aligned to provide emergency power to the associated essential buses.

b. At least once per 92 days by verifying that a sample of diesel fuel from within 3 inches of the bottom of each fuel storage tank, is within the acceptable limits specified in Table 1 of ASTM D975-81 when checked for water and sediment, and for viscosity.

c. At least once per 18 months by:

1. Subjecting the diesel to an inspection in accordance with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load greater than or equal to 41 kw load on Diesel Generator 1A and greater than or equal to 50 kw on Diesel Generator 1B without tripping.

3. Simulating a loss of offsite power and:

a) Verifying de-energization of the essential switchgear buses associated with Diesel Generators 1A and 1B.

b) Verifying the diesel generator starts from ambient condition on the auto-start signal, energizes the essential switchgear bus with permanently connected loads, and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads.

SURVEILLANCE REQUIREMENTS - (Cont'd)

c) Verifying that all Diesel Generator 1B trips, except engine overspeed, overcrank, and generator differential, are automatically bypassed upon loss of voltage on the essential bus.

4. Verifying that all available loads to each diesel generator do not exceed the continuous load rating of 250 kw for Diesel Generator 1A and 400 kw for Diesel Generator 1B.

5.3.1.3 The starting and control power battery and battery charger of each diesel generator shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates,

2. The pilot cell specific gravity, corrected to 77°F and normal electrolyte level, is greater than or equal to 1.180, and has not decreased more than 0.04 from the value observed during the previous test, and

3. The overall battery voltage is greater than or equal to 24 volts for Diesel Generator 1A and greater than or equal to 32 volts for Diesel Generator 1B.

b. At least once per 18 months by verifying that:

1. The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. The battery-to-battery terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

4.3.2 ONSITE A.C. POWER DISTRIBUTION SYSTEMS

LIMITING CONDITION FOR OPERATION

As a minimum, the following A.C. electrical buses shall be OPERABLE and energized from sources of power other than a diesel generator but aligned to an OPERABLE diesel generator.

a. 480-Volt Essential Switchgear Bus 1A or 480-Volt Diesel Building Essential Switchgear Bus 1B.

b. 120-Volt A.C. Non-Interruptible Bus 1B.

c. 120-Volt Turbine Building Regulated Bus.

APPLICABILITY: At all times.

ACTION:

With less than the above required A.C. distribution system buses OPERABLE, suspend all operations involving FUEL HANDLING within 8 hours.

SURVEILLANCE REQUIREMENTS

5.3.2 Each of the above required A.C. distribution system buses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.3.3 ONSITE D.C. POWER DISTRIBUTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

The following D.C. buses shall be energized and OPERABLE:

a. Generator Plant 125-volt D.C. bus, a full capacity charger, and a 125-volt battery bank, and

b. Diesel Building 125-volt D.C. bus, a full capacity charger, and a 125-volt battery bank or the Reactor Plant 125-volt D.C. bus, a full capacity charger, and a 125-volt battery bank.

APPLICABILITY: At all times.

ACTION:

With less than the above required 125-volt DC buses OPERABLE, suspend all operations involving FUEL HANDLING within 8 hours.

SURVEILLANCE REQUIREMENTS

5.3.3.1 Each of the above required D.C. distribution system buses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

5.3.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks,

2. The pilot cell specific gravity, corrected to 77°F and normal electrolyte level, is greater than or equal to 1.200,

3. The pilot cell voltage is greater than or equal to 2.0 volts, and

4. The overall battery voltage is greater than or equal to 120 volts.

SURVEILLANCE REQUIREMENTS - (Cont'd)

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.0 volts under float charge and has not decreased more than 0.3 volts from the value observed during the original acceptance test,

2. The specific gravity, corrected to 77°F, of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.04 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.

2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

3. The battery capacity is adequate to supply and maintain in OPERABLE status a load equal to or greater than all of the emergency loads or all of the actual emergency loads for 7 hours when the battery is subjected to a battery service test.

4. At the completion of the above test, the battery charger shall be demonstrated capable of recharging its battery while supplying normal D.C. loads. The battery shall be charged to at least 90% capacity in less than or equal to 24 hours.

4/5.3 Electrical Power Systems

The OPERABILITY of the normal and backup electrical power systems ensures that sufficient power will be available to supply necessary plant equipment normally and following a loss of offsite power event.

Normally, all power is supplied by a connection to the external grid via the reserve auxiliary transformer.

The onsite power distribution system is divided between two independent 2400 volt buses and each component of duplicate equipment is supplied by a different bus. Each 2400 volt bus also supplies power to a 480-volt bus. If the supply to either 480 volt bus fails, a connection may be manually made from the other 480 volt bus using a control switch in the control room. Duplicate equipment is also supplied by the 480 volt buses.

Each 480 volt bus normally supplies power to its associated essential bus. The 480 Volt Bus 1A supplies the 480 Volt Essential Switchgear Bus 1A. The 480 Volt Essential Switchgear Bus 1A supplies Turbine Building Motor Control Center 1A which supplies the Turbine Building 120 Volt Bus which in turn supplies the Turbine Building 120 Volt Regulated Bus. Diesel Generator 1A serves as the backup power supply for the 480 Volt Essential Switchgear Bus 1A.

Turbine Building Motor Control Center 1A supplies power for the Reactor Plant Battery Charger. The Reactor Plant Battery Charger carries the load and charges the batteries on the Reactor Plant 125 Volt DC Bus. The Reactor Plant 125 Volt DC Bus normally supplies power for Static Inverter 1A which supplies power to instrumentation and control circuits connected to 120 Volt AC Non-Interruptible Bus 1A. In case of Static Inverter 1A trouble, a transfer switch located in Static Inverter 1A automatically transfers to its alternate power supply which is Turbine Building 120 Volt Regulated Bus.

The 480 Volt Bus 1B supplies power to the 480 Volt Turbine Building Motor Control Center 1D. Turbine Building Motor Control Center 1D is the normal supply for the Generator Plant 125 Volt DC Bus through the Generator Plant Battery Charger. The Generator Plant 125 Volt DC Bus normally supplies power to a Generator Plant 125 Volt DC distribution panel which supplies power to instrumentation and control circuits connected to 120 Volt AC Non-Interruptible Bus 1C. The static switch of the Static Inverter 1C is capable of supplying power from Static Inverter 1C or its reserve 120 volt AC power source. In case of Static Inverter 1C trouble, the static switch will automatically transfer from the output of Static Inverter 1C to the reserve transformer which is supplied from Turbine Building Motor Control Center 1A. The Reactor Plant 125 Volt DC Bus is capable of providing power to the Generator Plant 125 Volt DC Bus or the Diesel Building 125 Volt DC Bus through tie breakers.

BASES - (Cont'd)

The Diesel Building 480 Volt Essential Switchgear Bus 1B supplies Reactor Building Motor Control Center 1A and Diesel Building 480 Volt Motor Control Center. Diesel Generator 1B is the emergency power supply for the Diesel Building 480 Volt Essential Switchgear Bus 1B. The Diesel Building 480 Volt Motor Control Center supplies power for the Diesel Building Battery Charger. The Diesel Building Battery Charger carries the load of and charges the batteries on the Diesel Building 125 Volt DC Bus.

The Diesel Building 125 Volt DC Bus is the normal power supply for Static Inverter 1B which supplies power to instrumentation and control circuits connected to 120 Volt AC Non-Interruptible Bus 1B. In case of Static Inverter 1B trouble, its static switch can automatically transfer from the output of Static Inverter 1B to the output of its alternate supply which is a transformer supplied by the Diesel Building 480 Volt Motor Control Center. The Turbine Building 120 Volt Regulated Bus is the reserve feed for the 120 Volt AC Non-Interruptible Bus 1B.

4.4 FIRE PROTECTION

4.4.1 FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 4.4.1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With one or more of the fire detection instruments shown in Table 4.4.1 inoperable:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instruments:

   1) at least once per four hours in the Reactor Containment Building and High Radiation areas;

   2) at least once per hour in any other area.

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of a Licensee Event Report, prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

SURVEILLANCE REQUIREMENTS

5.4.1.1 Each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

5.4.1.2 The supervised circuits associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

FIRE DETECTION INSTRUMENTATION

TABLE 4.4.1

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Containment Building	
Zone 19 Dome Level	2
Zone 19 Mezzanine Level	2
Zone 19 Grade Level	2
Zone 19 Basement Level	2
2. Electrical Equipment Room	2
3. Crib House	
Zone 4	1
4. Diesel Generator Rooms	
1A Zone 17	1
1B Zone 16	1
5. 1B Emergency Diesel Generator Building Battery Room	
Zone 16	1
6. Oil Storage Room	
Zone 13	1
7. Control Room and Office Area	
Zone 8	2

4.4.2 FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

The Fire Suppression Water System shall be OPERABLE with:

a. At least one high pressure diesel driven pump with a capacity of 750 gallons per minute with its discharge aligned to the fire suppression header.

b. An OPERABLE flow path capable of taking suction from the Mississippi River and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser required to be OPERABLE per Specification 4.4.3.

APPLICABILITY: At all times.

ACTION:

With the Fire Suppression Water System inoperable:

1. Establish a backup fire suppression water system within 24 hours, and

2. Submit a Special Report in accordance with Specification 6.8.2 within 30 days outlining the action taken, the cause of inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

5.4.2.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by performance of a system flush.

c. At least once per 18 months by cycling each manual valve in the flow path through at least one complete cycle of full travel.

d. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

SURVEILLANCE REQUIREMENTS - (Cont'd)

1. Verifying that the pump develops at least 750 gpm at a system head of 320 feet, and

2. Verifying that the high pressure pump starts to maintain the Fire Suppression Water System pressure $\geq$ 60 psig.

e. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11, of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

5.4.2.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the fuel storage tank for each diesel contains at least 150 gallons of fuel.

b. At least once per 31 days by starting the diesel from ambient conditions and operating for at least 20 minutes.

c. At least once per 92 days by verifying that a sample of diesel fuel from within three inches of the bottom of each fuel storage tank is within acceptable limits specified in Table 1 of ASTM D975-81 for water and sediment, and or viscosity.

d. At least once per 18 months, by:

1. Subjecting the diesel to the annual inspection in accordance with its manufacturer's recommendations for the class of service, and

2. Verifying the diesel starts from ambient conditions on the auto-start signal and operates for $\geq$ 20 minutes while loaded with the fire pump.

FIRE SUPPRESSION WATER SYSTEM

SURVEILLANCE REQUIREMENTS - (Cont'd)

5.4.2.3 Each fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each battery is above the plates, and

   2. The overall battery voltage is $\geq$ 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

   1. The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

   2. The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

4.4.3 SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

The following spray and/or sprinkler systems shall be OPERABLE:

a. The turbine lube oil reservoir.
b. The turbine lube oil storage room.
c. The reserve auxiliary transformer.
d. 1A Diesel Generator room.
e. Electrical penetration area.
f. Crib house diesel fire pump area.

APPLICABILITY: At all times.

ACTION:

With any of the above required spray and/or sprinkler systems inoperable, establish a 1-hour fire watch within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of a Licensee Event Report, prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 within the next 30 days outlining the action taken, the cause of the inoperability and plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

5.4.3.1 The above required spray and/or sprinkler system shall be demonstrated OPERABLE:

a. At least once per 18 months by:

   1. Cycling each manual valve in the flow path through at least one complete cycle of full travel.

   2. Performing a system functional test which includes simulated automatic actuation of the reserve transformer deluge system.

   3. Inspecting the spray headers to verify their integrity.

   4. Inspecting each nozzle to verify no blockage.

4.4.4. CHEMICAL EXTINGUISHING

LIMITING CONDITIONS FOR OPERATION

The chemical extinguishing systems located in the following areas shall be OPERABLE:

a. 1B Emergency Diesel Generator, with a minimum of 4 high-pressure CO_2 bottles weighing at least 190 lbs. each.

b. Electrical equipment room, with a minimum of 1 Halon 1301 bottle weighing at least 270 lbs.

APPLICABILITY: At all times.

ACTION:

With either of the above required chemical extinguishing systems inoperable, establish a 1-hour fire watch within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of a Licensee Event Report, prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

5.4.4 Each of the above required chemical extinguishing systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying tank weight of each of the required bottles and pressure of the Halon bottle.

b. At least once per 18 months by:

1. Verifying each system, including associated ventilation dampers, actuates manually and automatically with the solenoid valve removed from the respective bottles, upon receipt of a simulated test signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

4.4.5 FIRE HOSE STATIONS

LIMITING CONDITIONS FOR OPERATION

The fire hose stations in the following locations shall be OPERABLE:

1. 1B Emergency Diesel Generator Building.
2. Turbine Building Grade Floor by turbine lube oil reservoir.
3. Main Floor Turbine Building outside Control Room.
4. Containment Building Basement.
5. Waste Treatment Building.

APPLICABILITY: At all times.

ACTION:

With a hose station inoperable, establish a 1-hour fire watch, or route an additional hose of equivalent capacity to the unprotected area within one hour.

SURVEILLANCE REQUIREMENTS

5.4.5 Each of the fire hose stations shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and reracking, and
   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.
   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

4.4.6 YARD FIRE HYDRANTS AND OUTSIDE HOSE HOUSES

LIMITING CONDITION FOR OPERATION

The five yard fire hydrants and four outside hose houses shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With one or more of the yard fire hydrants or outside hose houses inoperable, verify sufficient lengths of 2-1/2 inch diameter hose are available to provide service to the unprotected area within one hour. Restore the inoperable hydrant(s) and/or hose houses(s) to OPERABLE status within 14 days or, in lieu of a Licensee Event Report, prepare and submit a Special Report to the Commission pursuant to Specification 6.8.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

5.4.6 Each of the yard fire hydrants and outside hose houses shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the outside hose house to assure all required equipment is at the hose house.

b. At least once per 6 months, by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any yard fire hydrant.

   2. Replacement of all degraded gaskets in couplings.

4/5.4 Fire Protection

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where equipment needed for safe storage is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a report to the Commission provides for evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear facility.

4.5 INSERVICE INSPECTION

4.5.1 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

4.5.1.1 The structural integrity of ASME Code Class 3 components associated with fuel storage shall be maintained in accordance with Specification 5.5.1.1.

4.5.1.2 The structural integrity of the LACBWR ventilation stack and the smoke stack of the conventional steam power generating station, Genoa 3, adjacent to the LACBWR facility, shall be maintained in accordance with Specification 5.5.1.2.

APPLICABILITY: At all times.

ACTION:

a. With the structural integrity of any ASME Code Class 3 component(s) associated with fuel storage not conforming to the above requirement, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

b. With the structural integrity of the LACBWR or Genoa 3 stack not maintained, perform an engineering evaluation.

SURVEILLANCE REQUIREMENTS

5.5.1.1 Inservice inspection of ASME Code Class 3 components associated with fuel storage shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, Summer 1975 Addenda. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

5.5.1.2 The exterior surfaces of the LACBWR ventilation stack and the Genoa 3 stack shall be inspected for structural integrity at least once every 5 years.

INSERVICE INSPECTION

4.5.2 PUMPS AND VALVES

LIMITING CONDITION FOR OPERATION

The OPERABILITY of ASME Code Class 3 pumps and valves associated with fuel storage shall be maintained in accordance with Section 5.5.2.

APPLICABILITY: At all times.

ACTION:

With a pump or valve falling into its Action range, declare the affected pump or valve INOPERABLE.

SURVEILLANCE REQUIREMENTS

5.5.2 Inservice testing of ASME Code Class 3 pumps and valves associated with fuel storage shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, Summer 1975 Addenda. Nothing in the ASME Boiler Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

BASES

4/5.5 Inservice Inspection

This section ensures that inservice inspection of ASME Code Class 3 components associated with fuel storage and inservice testing of ASME Code Class 3 pumps and valves associated with fuel storage will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. The edition specified is that used at the time of plant shutdown.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 3.7 to perform surveillance activities prior to entry into specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And, for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

This section also ensures that the structural integrity of the LACBWR and Genoa 3 stacks are maintained.

4.6 RADIATION MONITORING

4.6.1 RADIATION MONITORS

LIMITING CONDITION FOR OPERATION
==========

Area radiation monitors and portable radiation detectors shall be demonstrated OPERABLE in accordance with Specification 5.6.1.1 for area radiation monitors and Specification 5.6.1.2 for portable radiation detectors.

APPLICABILITY: At all times.

ACTION:

With a monitor/detector not satisfactorily meeting its surveillance criteria or if its surveillance requirement is not performed, declare the monitor/detector INOPERABLE.

SURVEILLANCE REQUIREMENTS
==========

5.6.1.1 Each area radiation monitor shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least daily, a CHANNEL FUNCTIONAL TEST at least quarterly, and a CHANNEL CALIBRATION at least once per 18 months.

5.6.1.2 Each portable radiation detector shall be demonstrated OPERABLE by performance of a SOURCE CHECK at least once every 2 weeks and calibrated at least semiannually.

4.6.2 POST-ACCIDENT RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

The following post-accident radiation monitoring instrumentation channels shall be OPERABLE with their alarm setpoints within the specified limits.

a. A High Range Containment Building Area Radiation Monitor, with its high alarm setpoint at $\leq$ 100 R/hr and

b. A Stack Midrange Noble Gas Effluent Monitor with its high alarm setpoint at $\leq$ 1.0 E-1 uci/cc or equivalent counting rate to Kr-85.

APPLICABILITY: At all times.

ACTION:

a. With a post-accident radiation monitoring channel alarm setpoint exceeding the above value, adjust the setpoint to within the limit within 24 hours or declare the channel INOPERABLE.

b. With less than the required post-accident radiation monitoring instrumentation channels OPERABLE, restore the INOPERABLE channel(s) to OPERABLE status within 7 days.

SURVEILLANCE REQUIREMENTS

5.6.2 Each post-accident radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK daily, CHANNEL FUNCTIONAL TEST monthly, and CHANNEL CALIBRATION at least once per 18 months.

4.6.3 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting materials or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

SURVEILLANCE REQUIREMENTS

5.6.3.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

   a. The licensee, or

   b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

5.6.3.2 Test Frequencies - Each category of sealed sources, excluding startup sources, fission detectors previously subjected to core flux, and sources as specfied in 10 CFR 30.15 shall be tested at the frequency described below.

   a. Sources in Use - At least once per six months for all sealed sources containing radioactive materials;

      1. With a half-life greater than 30 days, excluding Hydrogen 3, and

      2. In any form other than gas.

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b. Stored Sources Not in Use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

5.6.3.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

BASES

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4/5.6.1 Radiation Monitors

The surveillance requirements specified will ensure that area radiation monitors and portable radiation detectors being used are functional.

4/5.6.2 Post-Accident Radiation Monitoring Instrumentation

The operability of the post-accident radiation monitoring instrumentation is established so that if an accident does occur, it can be monitored. The setpoints are based on analyses of the radiological aspects of selected worst case accident scenarios involving the release of Kr-85 from spent fuel assemblies and the direct radiation from exposed irradiated components.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Calibrations of Containment Building Area High Range Radiation Monitors may be performed under the requirements of NUREG-0578.

4/5.6.3 Sealed Source Contamination

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into two groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

4.7 RADIOACTIVE EFFLUENTS

4.7.1 RADIOACTIVE LIQUID EFFLUENTS

4.7.1.1 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

The following radioactive liquid effluent monitoring instrumentation channels shall be OPERABLE, with their alarm setpoints set to ensure that the limits of Specification 4.7.1.2 are not exceeded.

a. Liquid Radwaste Effluent Line Monitor or Turbine Condenser Cooling Water Monitor, and

b. Liquid Radwaste Effluent Line Flow Meter.

The alarm setpoints for these monitors will be determined and adjusted using methodology in the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: At all times when releasing liquid radioactive effluents.

ACTION:

a. With the Liquid Radwaste Effluent Line Monitor or Turbine Condenser Cooling Water Monitor channel alarm/trippoint setpoint less conservative than that required by the above specification, immediately suspend the release or declare the channel inoperable or change the setpoint so that it is acceptably conservative.

b. With both channels not OPERABLE, or if both alarm setpoints are found to be less conservative than required, suspend release of liquid radioactive effluent without delay. Effluent releases may be resumed with neither activity monitor OPERABLE, provided that at least two independent samples are analyzed and that at least two technically qualified members of the staff independently verify the release rate calculations. If channels are not operable for more than 30 continuous days, explain in the next Semi-Annual Effluent Reports pursuant to Specification 6.8.1.2.

c. With the flow meter not OPERABLE, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

SURVEILLANCE REQUIREMENTS

5.3.1.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown on Table 5.7.1.1.

TABLE 5.7.1.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	SURVEILLANCE REQUIREMENT CONDITIONS
a. Liquid Radwaste Effluent Line Monitor	P	P*	Q(1)	R(3)	**
b. Turbine Condenser Cooling Water Monitor	P	M*	Q(1)	R(3)	**
c. Liquid Radwaste Effluent Line Flow Meter	D(2)	NA	NA	R(4)	**

* Background radiation may be used for the source check.
** During applicable conditions.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels at the alarm setpoint.
b. Instrument indicates a downscale (circuit failure) failure.

(2) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days in which continuous, periodic, or batch releases are made.

(3) The CHANNEL CALIBRATION shall include the use of a known liquid radioactive source positioned in a reproducible geometry with respect to the sensor. The source will have the gamma emitting radionuclide mixture and activity concentration which would normally be measured by the channel during batch discharges.

(4) The CHANNEL CALIBRATION will be in accordance with plant procedure.

RADIOACTIVE EFFLUENTS

RADIOACTIVE LIQUID EFFLUENTS

4.7.1.2 CONCENTRATION

LIMITING CONDITION FOR OPERATION

The concentration of radioactive material released in liquid effluents at any time to areas beyond EFFLUENT RELEASE BOUNDARY shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, issue of December 30, 1982, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 6×10^{-4} uci/ml total activity concentration.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released beyond the EFFLUENT RELEASE BOUNDARY exceeding the above limits, without delay restore concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

5.7.1.2 The radioactivity content of each batch of radioactive liquid waste to be discharged shall be determined prior to release by sampling and analysis in accordance with Table 5.7.1.2. The results of pre-release analyses shall be used in accordance with the ODCM methodology to assure that the concentration at the point of release is maintained within the limits of Specification 4.7.1.2.

RADIOACTIVE LIQUID EFFLUENTS

4.7.1.3 DOSE

LIMITING CONDITION FOR OPERATION
==========

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to areas beyond EFFLUENT RELEASE BOUNDARY shall be limited:

a. During any calendar quarter to ≤ 1.5 mrem to the total body and to ≤ 5 mrem to any organ, and
b. During any calendar year to ≤ 3 mrem to the total body and to ≤ 10 mrem to any organ.

APPLICABILITY At all times.

ACTION:

With the calculated dose from the release of radioactive materials in liquid effluents exceeding the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions which have been or will be taken to assure that subsequent releases shall be in compliance with the above limits.

SURVEILLANCE REQUIREMENTS
==========

5.7.1.3 Dose Calculations: Cumulative dose contributions from liquid effluents shall be determined at least once per calendar quarter in accordance with the ODCM.

TABLE 5.7.1.2

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS[d]
Waste Tank Batch Releases[a]	P	P	Principal Gamma Emitters[c]
	One Batch/M	M	Dissolved and Entrained Gases (gamma emitters)
	P	M Composite[b]	H-3
			Gross Alpha
	P	Q Composite[b]	Sr-90
			Fe-55

a. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, to assure representative sampling.

b. A composite sample is one made up of individual samples which are proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquid release.

c. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Co-60, Zn-65, Cs-134, Cs-137, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

d. Methods of calculating the Lower Limits of Detection (LLD) shall be contained in plant procedures and are calculated in accordance with criteria of NUREG-0473, Revision 2.

4.7.2 RADIOACTIVE GASEOUS EFFLUENTS

4.7.2.1 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 4.7.2.1 shall be OPERABLE with their alarm and/or trip setpoints set to ensure that the limits of Specification 4.7.2.2 are not exceeded. The stack noble gas instrumentation alarm setpoint will be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 4.7.2.1

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm and/or trip setpoint less conservative than that required by the above specification, declare the channel inoperable or change the setpoint so that it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION required by Table 4.7.2.1. Exert best efforts to return the instruments to OPERABLE status within 30 days, and if unsuccessful, explain in the next Semi-Annual Radioactive Effluent Release Report pursuant to 6.8.1.2 why the inoperability was not corrected in a timely manner.

SURVEILLANCE REQUIREMENTS

5.7.2.1 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 5.7.2.1.

TABLE 4.7.2.1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION[A]

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ACTION
1. Reactor Containment Building Ventilation Monitor System			
a. Particulate Activity Monitor	1	*	B
b. Gaseous Activity Monitor	1	*	B
c. Sampler Flow Rate Measuring Device	1	*	C
2. Stack Monitor System			
a. Noble Gas Activity Monitor	1	**	D
b. Particulate Activity Monitor	1	**	E
c. Sampler Flow Rate Measuring Device	1	**	C

* When Containment Building Ventilation System is in operation.
** At all times, unless alternate monitoring is available.

A. For post-accident instrumentaion, refer to Specification 4.6.2.

B. With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases through this pathway may continue as long as stack monitors are OPERABLE and CONTAINMENT INTEGRITY is not required; otherwise, isolate Containment Building Ventilation.

C. With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 24 hours.

D. With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the Containment Building Monitor Gaseous Activity Monitor is OPERABLE; otherwise, isolate the Containment Building.

E. With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided continuous collection of samples with auxiliary sampling equipment is initiated within 12 hours.

TABLE 5.7.2.1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL (4) CALIBRATION	SURVEILLANCE REQUIREMENT CONDITIONS
1. Reactor Containment Building Ventilation Monitor System					
a. Particulate Activity Monitor	D	M	Q(1)	R	*
b. Gaseous Activity Monitor	D	M	Q(1)	R	*
c. Sampler Flow Rate Measuring Device	D	N/A	Q(3)	R	*
2. Stack Monitor System					
a. Noble Gas Activity Monitor	D	M	Q(2)	R	*
b. Particulate Activity Monitor	D	N/A	Q(2)	R	*
c. Sampler Flow Rate Measuring Device	D	N/A	Q(3)	R	*

* During applicable conditions per Table 4.7.2.1.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway (if CONTAINMENT INTEGRITY is required) and control room alarm annunciation occurs if any of the following conditions exist:
a. Instrument indicates measured levels at or above the alarm setpoint.
b. Instrument indicates a downscale failure (provides control room annunciation alarm only).
c. Instrument indicates a circuit failure (provides control room annuniciation alarm only).

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:
a. Instrument indicates measured level above the alarm setpoint on one channel.
b. Instrument indicates a failure by a Low Flow and Low Count Rate signal.
c. Instrument controls in Maintenance mode.

(3) The CHANNEL FUNCTIONAL TEST shall also demonstrate that the control room local alarm occurs if the flow instrument indicates measured levels below the minimum and/or above the maximum alarm setpoint.

(4) The CHANNEL CALIBRATION shall be conducted in accordance with plant procedures.

RADIOACTIVE GASEOUS EFFLUENTS

4.7.2.2 INSTANTANEOUS DOSE RATE

LIMITING CONDITION FOR OPERATION

The dose rate due to radioactive materials released in gaseous stack effluents to areas beyond the EFFLUENT RELEASE BOUNDARY shall be limited to the following:

a. The dose rate limit for noble gases shall be ≤ 500 mrem/year to the total body and ≤ 3000 mrem/year to the skin, and

b. The dose rate limit for H-3 and for all radionuclides in particulate form with half lives greater than 8 days, shall be ≤ 1500 mrem/year to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, without delay decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

5.7.2.2.1 The dose rate due to noble gases in gaseous stack effluents shall be determined to be within the above limits in accordance with the ODCM.

5.7.2.2.2 The dose rate due to H-3 and for all radioactive materials in particulate form with half lives > 8 days in gaseous stack effluents shall be determined to be within the above limits in accordance with the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 5.7.2.2.

TABLE 5.7.2.2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis [d][e]
A. Stack Effluents	Continuous[b]	W[a] Composite Particulate Sample	Principal Gamma Emitters[c]
	Continuous[b]	Q Composite Particulate Sample	Sr-90
	Continuous[b]	Q Composite Particulate Sample	Gross Alpha
	Continuous[b]	Noble Gas Monitor	Noble Gases Gross Beta and Gamma
	M Grab Sample	M	H-3[d]

TABLE NOTATION:

(a) Filter samples shall be changed at least weekly, and filter analyses shall be completed within 7 days.

(b) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 5.7.2.2.1, 5.7.2.2.2, 5.7.2.3 and 5.7.2.4.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Co-60, Zn-65, Cs-134, Cs-137, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable and measurable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.8.1.2.

(d) When upper cavity is flooded or FUEL HANDLING is being performed, stack tritium grab samples will be taken at least once per 7 days.

(e) Lower Limits of Detection (LLD) are determined in accordance with plant procedures and are calculated in accordance with criteria of NUREG-0473, Revision 2.

RADIOACTIVE GASEOUS EFFLUENTS

4.7.2.3 DOSE, NOBLE GASES

LIMITING CONDITION FOR OPERATION

The air dose to a MEMBER OF THE PUBLIC due to noble gases released in gaseous effluents to areas beyond EFFLUENT RELEASE BOUNDARY shall be limited to the following, (See Figure 4/5.7):

a. During any calendar quarter, to ≤ 5 mrad for gamma radiation and ≤ 10 mrad for beta particle radiation; and

b. During any calendar year, to ≤ 10 mrad for gamma radiation and ≤ 20 mrad for beta particle radiation.

APPLICABILITY: At all times.

ACTION:

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions which have been taken or will be taken to reduce the releases of radioactive noble gases in gaseous effluents so that the cumulative dose during each subsequent quarter is within 5 mrad for gamma radiation and 10 mrad for beta radiation, and during the calendar year is within 10 mrad for gamma radiation and 20 mrad for beta radiation.

SURVEILLANCE REQUIREMENTS

5.7.2.3 Dose Calculations: Cumulative dose contributions shall be determined in accordance with the ODCM at least quarterly.

4.7.2.4 DOSE, RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

The dose to a MEMBER OF THE PUBLIC from H-3, and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents released to areas beyond EFFLUENT RELEASE BOUNDARY shall be limited to the following (See Figure 4/5.7):

a. During any calendar quarter to $\leq$ 7.5 mrem to any organ, and
b. During any calendar year to $\leq$ 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

With the calculated dose from the release of H-3 and all radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.8.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions which have been taken or will be taken to reduce these releases in gaseous effluents during remaining quarters so that the cumulative dose during each subsequent quarter is within 7.5 mrem and for the calendar year is within 15 mrem to any organ.

SURVEILLANCE REQUIREMENTS

5.7.2.4 Dose Calculations: Cumulative dose contributions shall be determined in accordance with the ODCM at least quarterly.

4.7.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

Solid radioactive wastes shall be handled in accordance with a PROCESS CONTROL PROGRAM in order to meet shipping and burial ground requirements.

APPLICABILITY: At all times when processing solid radioactive wastes for shipment and disposal.

ACTION:

With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

SURVEILLANCE REQUIREMENTS

5.7.3 The PROCESS CONTROL PROGRAM shall be used to assure the appropriate form for packaging each type of radioactive waste (e.g., filter sludges, spent resins, tank bottoms, dry active wastes).

4.7.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

The dose equivalent to any MEMBER OF THE PUBLIC due to releases of radioactivity and radiation, shall be limited to $\leq$ 25 mrem to the total body or any organ (except the thyroid, which is limited to $\leq$ 75 mrem) over a period of one calendar year.

APPLICABILITY: At all times.

ACTION:

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limit of Specifications 4.7.1.3.b, 4.7.2.3.b or 4.7.2.4.b, a determination should be made, including direct radiation from reactor containment and radioactive waste storage tanks to determine if limits of Specification 4.7.4 have been exceeded. If the limits of Specification 4.7.4 have been exceeded, prepare and submit a Special Report (including an analysis which estimates the radiation exposure to a MEMBER OF THE PUBLIC for the calendar year) to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 4.7.4. If the release condition resulting in violation of Specification 4.7.4 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190. Submittal of the Special Report is considered a timely request, and a variance is granted until staff action on the request is complete.

SURVEILLANCE REQUIREMENTS

5.7.4.1 Dose Calculations: Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 5.7.1.3, 5.7.2.3, 5.7.2.4, and in accordance with the ODCM once per year.

5.7.4.2 Dose Determination: Cumulative dose contributions from direct radiation from the reactor containment or radioactive waste storage tanks shall be determined in accordance with the methodology and parameters of the ODCM once per year.

BASES

4/5.7.1 RADIOACTIVE LIQUID EFFLUENTS

4/5.7.1.1 Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents with the alarm setpoints set to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20.

4/5.7.1.2 Concentration

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. The concentration limit for noble gases is based upon the assumption that Kr-85 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

4/5.7.1.3 Dose

This specification is provided to implement the requirements of Sections II.A, III.A, IV.a and Annex of Appendix I, 10 CFR Part 50. The dose calculations in the ODCM implement the requirement in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated.

4/5.7.2 RADIOACTIVE GASEOUS EFFLUENTS

4/5.7.2.1 Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The only significant noble gas remaining is Kr-85. The alarm setpoints for these instruments shall be set to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20.

4/5.7.2.2 Instantaneous Dose Rate

This specification is provided to ensure that the dose rate at any time at the EFFLUENT RELEASE BOUNDARY from gaseous effluents from LACBWR will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, outside the EFFLUENT RELEASE BOUNDARY to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the EFFLUENT RELEASE BOUNDARY, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the EFFLUENT RELEASE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the EFFLUENT RELEASE BOUNDARY to $\leq$ 500 mrem/year to the total body or to $\leq$ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding organ dose rate above background to an infant via the inhalation pathway to $\leq$ 1500 mrem/year.

4/5.7.2.3 Dose, Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated.

4/5.7.2.4 Dose, Radionuclides Other Than Noble Gases

This specification is provided to implement the requirements of Sections II.C, III.A, IV.A and Annex of Appendix I, 10 CFR Part 50. The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated.

4/5.7.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and applicable portions of 10 CFR Part 61.

BASES - (Cont'd)

4/5.7.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. The Special Report will describe a course of action which should result in the limitation of dose to a real individual for 12 consecutive months to within the 40 CFR 190 limits.

4.8 RADIOLOGICAL ENVIRONMENTAL MONITORING AND INTERLABORATORY COMPARISON

LIMITING CONDITION FOR OPERATION

4.8.1 The Radiological Environmental Monitoring Program, shall be conducted as specified in Table 4.8.1-1. An Interlaboratory Comparison Program for annual analyses of radioactive materials shall be conducted.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 4.8.1-1, prepare and submit to the Commission, in the Radiological Environmental Monitoring Report pursuant to Specification 6.8.1.1.c, a description of the reasons for not conducting the program as required, analysis of the causes of unexpected results, and the plans for preventing a recurrence.

b. With radiological environmental sample analyses in excess of the reporting levels in Table 4.8.1-2, when averaged over any calendar quarter, prepare and submit to the Commission a Special Report within 30 days, pursuant to Specification 6.8.2, with a description of the reasons for exceeding the reporting levels.

c. With Interlaboratory Comparisons not being performed, report the corrective actions taken to prevent a recurrence to the Commission in the Radiological Environmental Monitoring Report, pursuant to Specification 6.8.1.1.c.

SURVEILLANCE REQUIREMENTS

5.8.1.1 The radiological environmental monitoring samples shall be collected in accordance with Table 4.8.1-1 from locations specified in the ODCM, and shall be analyzed in accordance with requirements listed in Tables 4.8.1-1 and 4.8.1-2.

5.8.1.2 A summary of the results obtained from the above required Interlaboratory Comparison Program shall be included in the Radiological Environmental Monitoring Report pursuant to Specification 6.8.1.1.c.

TABLE 4.8.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND SAMPLING	NUMBER OF SAMPLES[A]	SAMPLING FREQUENCY	TYPE AND FREQUENCY[B] OF ANALYSIS
(1) Airborne Particles	3	Continuous operation of sampler with sample collected as determined by dustloading but at least weekly.	Analyze for gross beta activity ≥ 24 hours following filter change. Gamma isotopic analysis of each particulate sample which exhibits >10 times the mean beta activity of control station. Gamma isotopic analysis of at least a quarterly composite from each location.
(2) Direct Radiation	8 At least 2 monitors at each location.	SA	Gamma dose - at least semi-annually.
(3) Waterborne River Water	2	M	Gamma isotopic analysis on each sample monthly. Tritium analysis on composite of monthly samples at least quarterly.
(4) River Sediment	2	SA	Gamma isotopic analysis of each sample.
(5) Ingestion			
(a) Milk	1	At least monthly when animals are in pasture (May-Oct.).	Gamma isotopic analysis on each sample.
(b) Fish	1	At least semi-annually. One sample of two different species in the area that are important as a game or commercial species.	Gamma isotopic analysis edible portions of each.
(c) Vegetation	1	At time of harvest.	Gamma isotopic analyses of edible portions of each sample.

[A] Locations are specified in ODCM.
[B] LLD's are calculated in accordance to criteria of, and are essentially the same as those found in, NUREG-0473, Rev. 2.

TABLE 4.8.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting levels

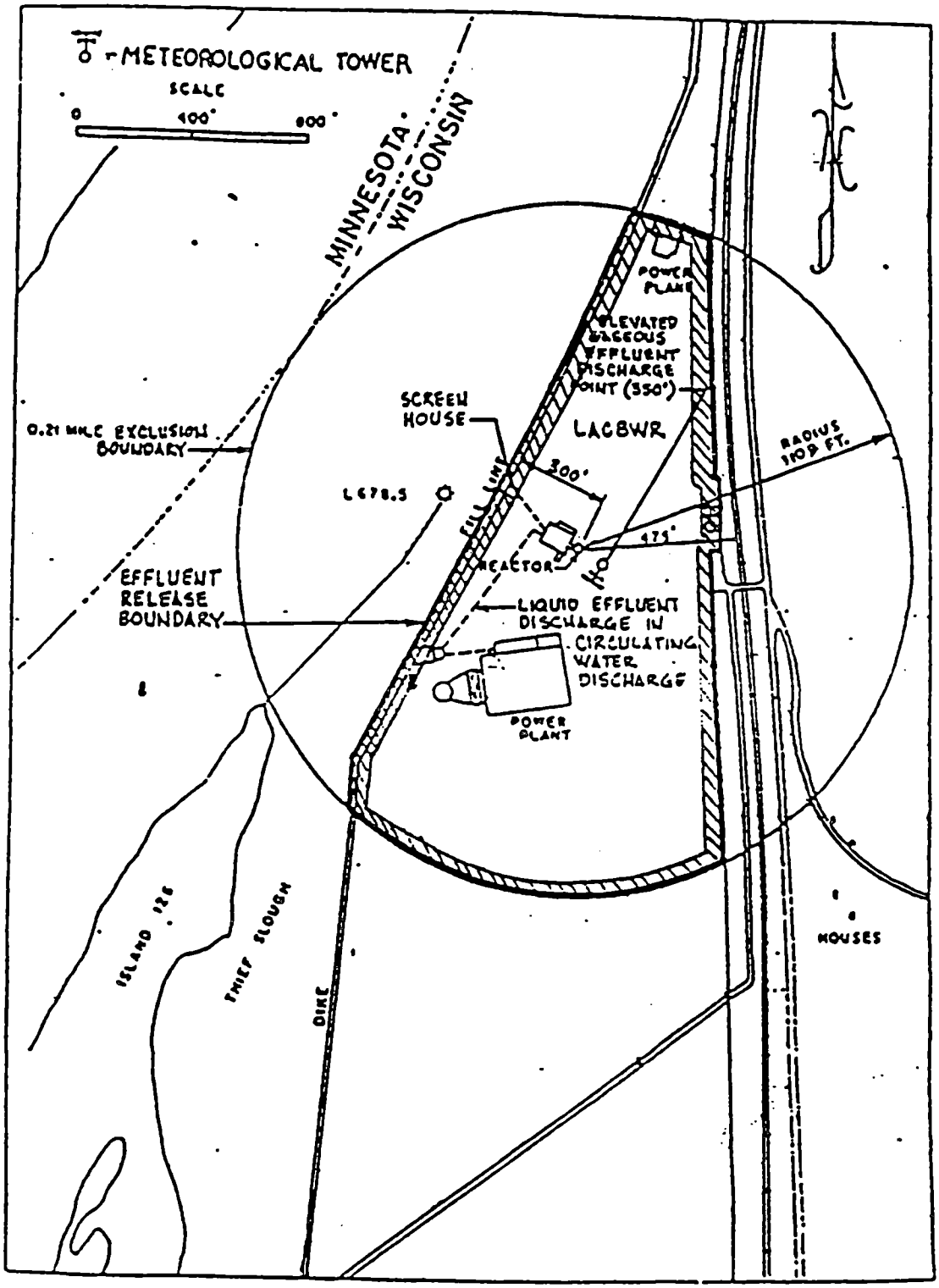
Analysis	Water pCi/l	Airborne Particulate (pCi/m^3)	Fish (pCi/Kg, Wet)	Milk (pCi/l)
H-3	2×10^4			
Mn-54	1×10^3		3×10^4	
Co-60	3×10^2		1×10^4	
Zn-65	3×10^2		2×10^4	
Cs-134	30	10	1×10^3	60
Cs-137	50	20	2×10^3	70

BASES

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from plants effluents. This monitoring program theory supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental samples are performed to demonstrate that the results are reasonably valid.

FIGURE 1



Effluent Release Boundary

6. ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or, during his absence from the Control Room, a designated individual) shall be responsible for Control Room command function.

6.2 ORGANIZATION

6.2.1 FACILITY STAFF

6.2.1.1 The facility organization shall be as follows:

a. Each on-duty shift shall be composed of at least one Shift Supervisor, who is a Certified Fuel Handler, and one qualified Control Room Operator.*

b. A qualified Control Room Operator or Shift Supervisor shall be within visual and/or audio distance of the Control Room annunciators.

c. All FUEL HANDLING shall be directly supervised by a Certified Fuel Handler.

d. "An individual qualified in radiation protection procedures shall be on site when there is fuel on site or there is a potential for release of radioactive materials." At least one additional Operator and one Health Physics Technician shall be on site when spent fuel or a spent fuel shipping cask is being handled or when any evolutions are being conducted in or above the Fuel Element Storage Well.

e. A Fire Brigade of at least 3 members shall be maintained on site at all times.**

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* Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. This provision does not permit any shift crew position to be unfilled upon shift change due to an oncoming shift crew member being late or absent.

** Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements. This provision does not permit any Fire Brigade position to be unmanned upon shift change due to an oncoming Brigade member being late or absent.

6.2.1.2 OVERTIME POLICY

The working hours of Operators, the Duty Shift Supervisor, Mechanical Maintenance and Instrument & Electrical Technicians when performing duties which may affect nuclear safety, and Health Physics Technicians, when performing radiation protection duties which may affect the safety of the public, shall be limited.

In the event overtime must be used, the following restrictions shall be followed:

(1) The specified personnel shall not be permitted to work more than 16 hours straight, excluding shift turnover time.

(2) The specified personnel shall not be permitted to work more than 16 hours in any 24-period, more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period.

(3) A break of at least 8 hours shall be allowed following overtime before the next scheduled shift for the specified personnel, if the above limits are exceeded.

In the event overtime must be used in excess of the above restrictions, the Plant Manager or his designate, must authorize the deviation and the cause must be documented.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions except for the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained which shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971.

6.4.1.1 A training program for "Certified Fuel Handler" shall be completed, as per the accredited operator training program procedure, prior to performing fuel handling. General Employee Training, Health Physics Technician training and retraining shall be completed as per their respective Training Plan Procedures (TPP's).

6.4.2 A training program for the Fire Brigades shall meet the requirements of Section 27 of the NFPA Code-1975, except for the frequency of Fire Brigade training sessions, which shall be held at least once per 92 days.

6.5 REVIEW AND AUDIT

6.5.1 OPERATIONS REVIEW COMMITTEE (ORC)

6.5.1.1 FUNCTION

The Operations Review Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

6.5.1.2 COMPOSITION

The Operations Review Committee shall be composed of the following:

Chairman: Plant Manager

Members: LACBWR Department Supervisors
LACBWR Staff Engineers
LACBWR Shift Supervisors
LACBWR Management Personnel

6.5.1.3 ALTERNATES

All alternate members shall be appointed in writing by the ORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ORC activities at any one time.

6.5.1.4 MEETING FREQUENCY

The ORC shall meet at least once per calendar quarter and as convened by the ORC Chairman or his designated alternate.

6.5.1.5 QUORUM

The minimum quorum of the ORC necessary for the performance of the ORC responsibility and authority provisions of the Technical Specifications shall consist of the Chairman, or his designated alternate, and 3 members, including alternates.

6.5.1.6 RESPONSIBILITY

The Operations Review Committee shall be responsible for:

a. Review of (1) all procedures required by Specification 6.6 and changes thereto, (2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to facility systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the General Manager and to the Safety Review Committee (SRC).

f. Review of all REPORTABLE EVENTS.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or SRC.

i. Review of the Contingency Plan, the Plant Security Plan, and implementing procedures.

j. Review of the Emergency Plan and implementing procedures.

k. Review the waste management process control program, which includes the transportation packaging program.

l. Review of the Decommissioning Plan.

6.5.1.7 AUTHORITY

The Operations Review Committee shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under 6.5.1.6a through d above.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a through e above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the General Manager and the SRC of disagreement between the ORC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

6.5.1.8 RECORDS

The Operations Review Committee shall maintain written minutes of each ORC meeting that, at a minimum, document the results of all ORC activities performed under the responsibility and authority provision of these Technical Specifications. Copies shall be provided to the General Manager and the SRC.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

6.5.2.1 FUNCTION

The SAFETY REVIEW COMMITTEE shall function to provide independent review and audit of designated activities in the areas of:

a. FUEL HANDLING operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering,

h. Quality assurance practices, and

i. Waste management.

6.5.2.2 COMPOSITION

The SRC shall be composed of the:

Chairman:	Consultant
Members:	Consultant
	Plant Manager
	Director of External Relations
	General Manager

6.5.2.3 ALTERNATES

All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRC activities at any one time.

6.5.2.4 CONSULTANTS

Consultants shall be utilized as determined by the SRC Chairman to provide expert advice to the SRC.

6.5.2.5 MEETING FREQUENCY

The SRC shall meet at least once per six months.

6.5.2.6 QUORUM

The minimum quorum of the SRC necessary for the performance of the SRC review and audit functions of these Technical Specifications shall consist of the Chairman, or his designated alternate, and at least 3 SRC members, including alternates. No more than a minority of the quorum shall have line responsibility for the facility.

6.5.2.7 REVIEW

The Safety Review Committee shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provisions of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in section 50.59, 10 CFR.

d. Proposed changes to Appendix "A" Technical Specifications of this license.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant deviations from normal and expected performance of facility equipment that affects nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meeting minutes of the Operations Review Committee.

j. Changes to the Contingency Plan and Plant Security Plan.

k. Changes to the Emergency Plan.

l. Changes to the Decommissioning Plan.

6.5.2.8 AUDITS

Audits of facility activities shall be performed under the cognizance of the SRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Appendix "A" Technical Specifications and applicable license conditions at least once per 24 months.

b. The performance, training and qualifications of the entire facility staff at least once per 24 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 12 months.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 24 months.

f. The Contingency Plan, the Security Plan and implementing procedures, at least once per 24 months.

g. The Fire Protection Program and implementing procedures at least once per 24 months.

h. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 24 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

i. The Radiological Environmental Monitoring Program and results at least once per 24 months.

j. The OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM and implementing procedures at least once per 24 months.

k. Any other area of facility operation considered appropriate by the SRC or the General Manager.

6.5.2.9 AUTHORITY

The SRC shall report to and advise the General Manager on those areas of responsibility listed in Specifications 6.5.2.7 and 6.5.2.8.

6.5.2.10 RECORDS

Records of SRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved and forwarded to the General Manager within 20 days following each meeting.

b. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the General Manager and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 PROCEDURES *

6.6.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978, for a plant in the SAFSTOR condition.

b. FUEL HANDLING operations.

c. Surveillance and test activities of equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM and OFFSITE DOSE CALCULATION MANUAL implementation.

6.6.2 Each procedure of Specification 6.6.1, and changes thereto, shall be reviewed by the ORC and approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

* In the period immediately following issuance of the SAFSTOR T.S., many of the procedures currently in use will have references to the previous T.S. These procedures will still be correct in their other aspects however; and, as each comes due for its annual or biennial review, these references will be updated to include the references to the SAFSTOR T.S.

6.6.3 Temporary changes to procedures of Specification 6.6.1 may be made, provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the facility management staff.

c. The change is documented, reviewed by the ORC and approved by the Plant Manager within 30 days of implementation.

6.6.4 SPECIAL PROCEDURE REQUIREMENTS

6.6.4.1 PROCESS CONTROL PROGRAM (PCP)

a. The PCP shall be maintained on site and will be available for NRC review.

b. Licensee-initiated changes to the PCP shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

- Information to support the rationale for the change;
- A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and
- Documentation of the fact that the change has been reviewed and found acceptable by the ORC.

6.6.4.2 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall be maintained by the licensee. Changes to the ODCM will be outlined in the Semi-Annual Radioactive Effluent Release Report per Specification 6.8.1.2.

This submittal shall contain:

(1) Detailed information to support the rationale for the change. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s) and

(2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations.

6.7 CONTROL OF MAINTENANCE AND TESTING ACTIVITIES

6.7.1 Maintenance operations and routine tests shall be performed in conformance with these specifications.

6.7.2 Maintenance operations shall be performed as authorized by the Shift Supervisor. Maintenance involving the opening of systems containing radioactive materials shall be conducted under the surveillance of a Health Physics representative.

6.7.3 Components which have been repaired, replaced, or otherwise subjected to temporary or permanent modification shall be tested in accordance with procedures which are appropriate in view of the nature of the repair, replacement or modification, and in view of the condition of the system.

6.7.4 Key switches shall permit operational, maintenance, and test bypass of the safety instrumentation only with the approval of the Shift Supervisor.

6.8 REPORTING REQUIREMENTS

6.8.1 ROUTINE REPORTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

6.8.1.1 Reports required on an annual basis shall be submitted by March 1 of each year and shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel, including contractors, receiving exposures greater than 100 mRem/yr and their associated man rem exposure according to work and job functions, e.g., plant operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and fuel handling.* The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling 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.

b. A report containing a brief description of any changes, testing and experiments conducted under the criteria of 10 CFR 50.59, including a summary of the safety evaluations of them.

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* This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

c. An Annual Radiological Environmental Monitoring Report which shall include summarized and tabulated results, including interpretations and analysis of data trends, of environmental samples taken during the previous calendar year. In the event that some 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 as soon as possible in a supplementary report.

The report shall also include the following: a summary description of the Radiological Environmental Monitoring Program; a map of all sampling locations keyed to a table giving distances and directions from the plant, the results of the Interlaboratory Comparison Program, and a discussion of all analyses in which the LLD was not achievable.

6.8.1.2 Semi-Annual Radioactive Effluent Release Report

Paragraph (a)(2) of Part 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," of 10 CFR Part 50 requires that a report be made to the Commission within 60 days after January 1 and July 1 of each year. The report shall specify the quantity of each of the principal radionuclides released to unrestricted areas by liquids and gaseous effluents during the previous 6 months. The information submitted shall be in accordance with Appendix B of Regulatory Guide 1.21 (Revision 1) dated June 1974 with data summarized on at least a quarterly basis.

An annual summary of (hourly) meteorological data collected over the previous year in accordance with Regulatory Guide 1.21 (Rev. 1) will be included with the Semi-Annual Radioactive Effluent Release Report, which is submitted 60 days after January 1 of each year. This same report shall include an assessment of radiation doses to MEMBERS OF THE PUBLIC from radioactive liquid and gaseous effluents released beyond the EFFLUENT RELEASE BOUNDARY performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM). This report will contain any changes made to the ODCM during the previous twelve months.

6.8.2 SPECIAL REPORTS

Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.8.3 LICENSEE EVENT REPORTS

6.8.3.1 A Licensee Event Report shall be submitted to the Nuclear Regulatory Commission if any REPORTABLE EVENT of the type listed in Specification 6.8.3.3 occurs. The report shall meet the requirements of 10 CFR 50.73.

6.8.3.2 Each REPORTABLE EVENT shall be reviewed by the ORC and the Licensee Event Report shall be submitted to the SRC and General Manger.

6.8.3.3 The following types of events are REPORTABLE EVENTS:

a. Any violation of or condition prohibited by Technical Specifications.

b. Any deviation from Technical Specifications authorized by 10 CFR 50.54(x).

c. Serious degradation of nuclear fuel.

d. Loss of CONTAINMENT INTEGRITY when it is required to exist.

e. Any event or condition that resulted in the plant being in an unanalyzed condition that significantly compromised plant safety.

f. Any event or condition that resulted in the plant being in a condition outside its design basis.

g. Any event or condition that resulted in the plant being in a condition not covered by normal or emergency procedures.

h. Any natural phenomenon, other external phenomenon, or event such as fire, toxic gas release, or radioactive release, that posed an actual threat to plant safety or significantly hampered site personnel in the performance of duties necessary for plant safety.

i. Any event or condition that resulted in unplanned and unexpected manual or automatic actuation of the Emergency Diesel Generators or Containment Isolation.

j. Any airborne radioactivity release that exceeded 2 times the applicable concentrations of the limits specified in Appendix B, Table II of 10 CFR Part 20 in unrestricted areas, when averaged over a time period of one hour.

k. Any liquid effluent release that exceeded 2 times the limiting combined Maximum Permissible Concentration (MPC) (See Note 1 of Appendix B to 10 CFR Part 20) at the point of entry into the receiving water (i.e., EFFLUENT RELEASE BOUNDARY) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour.

6.8.4 IMMEDIATE NOTIFICATION REQUIREMENTS

Notification of declaration of an Emergency Class listed in the LACBWR Emergency Plan shall be made within 1 hour.

6.9 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.9.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.6.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.9.2 The following records shall be retained for the duration of the LACBWR License:

a. Facility design modification packages.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs, and records of analyses required by the Radiological Environmental Monitoring Program.

e. Records of reactor tests and experiments.

f. Records of training and qualification for current members of the facility staff.

g. Records of in-service inspections performed pursuant to these Technical Specifications.

h. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

i. Records of meetings of the ORC and the SRC.

6.10 RADIATION PROTECTION PROGRAM

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

6.11 HIGH RADIATION AREA

6.11.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mRem/hr but less than 1000 mRem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Special Work Permit (SWP).* Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

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

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.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device and who is responsible for providing positive exposure control over the activities within the area and who will perform periodic radiation surveillance at the frequency which will be established by the Health and Safety Supervisor.

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* Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the SWP issuance requirement during the performance of their assigned radiation protection duties, provided they are following plant radiation protection procedures for entry into high radiation areas.

6.11.2 For each area with radiation levels greater than 1000 mRem/hr, the control of Specification 6.11.1 shall be implemented and

(1) Each entrance or access point to the area shall be maintained locked except during periods when access to the area is required, with positive control over each individual entry, or

(2) Each entrance or access point to the area shall be equipped with a control device which shall energize a conspicuous visible or audible alarm signal in such a manner that the individual entering the high radiation area and the licensee or a supervisor of the activity are made aware of the entry.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
DAIRYLAND POWER COOPERATIVE	)	Docket No. 50-409
	)	Facility License No. DPR-45
La Crosse Boiling	)	
	)	
Water Reactor (LACBWR)	)	

ORDER AUTHORIZING DECOMMISSIONING OF FACILITY

By application dated December 21, 1987, as revised February 22, 1988, September 9, 1988, September 30, 1988, January 26, 1989, March 28, 1989, June 6, 1989, October 3, 1989, July 25, 1990, May 10, 1991, and July 25, 1991, Dairyland Power Cooperative (DPC) requested approval of its proposed Decommissioning Plan for the La Crosse Boiling Water Reactor (LACBWR) and an amendment to License No. DPR-45. A Notice of Consideration of Issuance of Amendment and Opportunity for Hearing was published in the FEDERAL REGISTER on April 8, 1988 (53 FR 11718). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Nuclear Regulatory Commission (the Commission) has reviewed the application with respect to the provisions of the Commission's rules and regulations and has found that decommissioning as stated in the licensee's Decommissioning Plan will be consistant with the regulations in 10 CFR Chapter I, and will not be inimical to the common defense and security or to the health and safety of the public. The basis for these findings is set forth in the concurrently issued Safety Evaluation by the Office of Nuclear Reactor Regulation.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact for the proposed action. Based on that Assessment, the

Commission has determined that the proposed action will not result in any significant environmental impact and that an environmental impact statement need not be prepared. The Notice of Issuance of Environmental Assessment was published in the FEDERAL REGISTER on August 7, 1991 (56 FR 37574).

Accordingly, the license is hereby ordered to decommission the reactor facility in accordance with its Decommissioning Plan and the Commission's rules and regulations.

For further details with respect to this action, see: (1) the licensee's application for authorization to decommission the facility, dated December 21, 1987, as revised February 22, 1988, September 9, 1988, September 30, 1988, January 26, 1989, March 28, 1989, June 6, 1989, October 3, 1989, July 25, 1990, May 10, 1991, and July 25, 1991; (2) Amendment No. 66 to License No. DPR-45; (3) the Commission's related Safety Evaluation; and (4) the Environmental Assessment and Finding of No Significant Impact. These documents are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555, and at the La Crosse Public Library, 800 Main Street, La Crosse, Wisconsin 54601. Copies of items (2), (3), and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Advanced Reactors and Special Projects.

Dated at Rockville, Maryland this 7th day of August 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Director
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

ENVIRONMENTAL ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION REGARDING ORDER AUTHORIZING FACILITY DECOMMISSIONING AND THE RENEWAL AND AMENDMENT OF POSSESSION ONLY LICENSE NO. DPR-45 DAIRYLAND POWER COOPERATIVE LA CROSSE BOILING WATER REACTOR DOCKET NO. 50-171

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action:

The La Crosse Boiling Water Reactor (LACBWR) is a 165 MW-thermal, boiling water reactor that operated from November 1, 1969 to its final shutdown on April 30, 1987. LACBWR is located 19 miles south of La Crosse, Wisconsin on the Mississippi River adjacent to a coal fired generating unit that is also owned and operated by Dairyland Power Cooperative (DPC). DPC has proposed to decommission LACBWR, and to renew and amend License No. DPR-45.

All spent fuel has been removed from the reactor and placed in the spent fuel storage wells. The spent fuel will remain on site until a Federal repository is available to receive it.

Need for Proposed Action:

The granting of the proposed action would allow DPC to retain LACBWR in a SAFSTOR status until March 29, 2031. The proposed delay would significantly reduce the gamma exposure rate to workers involved in final decontamination and dismantling.

Environmental Impact of the Proposed Action:

We have evaluated the proposed decommissioning and the renewal and amendment of the LACBWR possession only license with respect to 10 CFR 51.45. Environmental considerations are discussed in the following sections.

Unavoidable Impacts:

During the SAFSTOR period, LACBWR will continue to occupy a small (less than 2.0 acre) restricted area that is adjacent to a coal fired power plant.

Alternatives Comparison:

The three alternatives for decommissioning are SAFSTOR, ENTOMB and DECON. Each alternative as it relates to LACBWR is discussed below:

1. SAFSTOR

SAFSTOR is the alternative in which the facility is placed and maintained in a condition that allows it to be safely stored and subsequently decontaminated to levels that permit release of the property or area for unrestricted use. DPC has proposed the SAFSTOR alternative for LACBWR because spent fuel must remain on site until a Federal repository is available to accept it. No such Federal repository space is expected until after 2010. In addition, potential radiation exposure to workers will be significantly reduced with the proposed 30 to 50 year delay in dismantling. With repsect to on site storage of spent fuel, 10 CFR 51.23 states: "The Commission has made a generic determination that for at least 30 years beyond the expiration of reactor operating licenses no significant environmental impacts will result from the storage of spent fuel in reactor facility storage pools."

2. ENTOMB

ENTOMB is the alternative in which radioactive contaminants are encased in a structurally long-lived material such as concrete. The entombed structure is maintained and surveillance continued until the radioactivity decays to a level that permits unrestricted release. Long-lived radionuclides present at LACBWR such as Nickel-63 and Niobium-94 in the reactor vessel and vessel internals would not decay to levels acceptable for release to unrestricted access in any reasonable (100 year) period of time. Also, the storage of spent fuel on site prevents this option as spent fuel could not reasonably be entombed and would have to be removed when a Federal repository is available.

3. DECON

DECON is the alternative in which the equipment, structures and portions of the facility containing radioactive contaminants are dismantled and removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after final shutdown of the reactor.

This alternative could not be selected because of on site storage of spent fuel. Also it would result in higher exposures to workers doing the final dismantling. A larger volume of radioactive waste would likely result with prompt dismantling.

Local Short-Term Uses Verses Long-Term Productivity:

The site is now being used for power production with the continued operation of an adjacent coal fired power plant. The licensee has stated that there are no plans for this site other than electrical power production for the SAFSTOR period and there is no advantage gained by making this small area of land available earlier. Therefore, there is no conflict between short-term uses versus long-term productivity of the site.

Irreversible and Irretrievable Commitments of Resources:

The proposed SAFSTOR period followed by dismantling would not involve the commitment of any significant amount of resources. Conversely, there would be less volume of radioactive waste to dispose of at the end of the SAFSTOR period than with immediate DECON because of radioactive decay. With less volume of radioactive waste, the required burial space at a low-level waste burial site would be reduced.

Access Control to Radiation Areas:

All buildings and structures at LACBWR that retain spent fuel or residual radioactivity above levels acceptable for release to unrestricted access (Regulatory Guide 1.86, Table I) are within a protected area. Access to the spent fuel and this protected area is controlled by use of security guards, security fences, locked doors, and radiological procedures. In addition, access to the residual high level radiation in the reactor vessel is prevented by the shielded and sealed primary system. Requirements for access are specified in the Technical Specifications.

Potential Exposure to Workers:

DPC has provided an estimate of radiation exposures to workers during the SAFSTOR period. DPC estimated the exposures at about 46 person-rem/yr for the first 3 years that decreases to 5.7 person-rem/yr for the 27th through 30th year of SAFSTOR.

The requested period of SAFSTOR will reduce dose rates from activated components by a significant amount in the decontamination/dismantling phase since the activation is primarily Cobalt-60 which has a 5.2 year half-life.

Alternative to the Proposed Action:

An alternative to approval of the SAFSTOR option would be to require prompt dismantlement of LACBWR. Prompt dismantlement could not be easily accomplished because of the spent fuel remaining on site. The licensee could propose to construct an on site Independent Spent Fuel Storage Installation (ISFSI). Fuel transfer to the ISFSI could then occur but spent fuel handling equipment would also have to be constructed to allow eventual transfer of fuel to shipping containers for shipment to a Federal repository.

Considering the spent fuel disposal situation, the radiation exposure to workers and the greater radioactive waste volume, prompt dismantling would have a greater environmental impact.

Alternative Use of Resources:

This action involves no use of resources not previously considered in the Commission's Final Environmental Statement for LACBWR dated April, 1980 (NUREG-0191).

Agencies and Person Consulted:

The NRC staff reviewed the licensee's request and did not consult other agencies or persons.

FINDING OF NO SIGNIFICANT IMPACT

Based upon the foregoing environmental assessment, the Commission has concluded that the proposed action will not have a significant effect on the quaility of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for this proposed action.

For further details with respect to this action, see the licensee's request for a license amendment dated December 21, 1987 as revised February 22, 1988, September 9, 1988, September 30, 1988, January 26, 1989, March 28, 1989, June 6,

1989, October 3, 1989, July 25, 1990, May 10, 1991, and July 25, 1991. These documents are available for inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555 and at the La Crosse Public Library, 800 Main Street, La Crosse, Wisconsin 54601.

FOR THE NUCLEAR REGULATORY COMMISSION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ORDER AUTHORIZING DECOMMISSIONING AND

AMENDMENT NO. 66 TO POSSESSION ONLY LICENSE NO. DPR-45

DAIRYLAND POWER COOPERATIVE

LA CROSSE BOILING WATER REACTOR

DOCKET NO. 50-409

1.0 INTRODUCTION

The La Crosse Boiling Water Reactor (LACBWR) is a 165 MW-thermal BWR located on the east bank of the Mississippi River, 19 miles south of La Crosse, Wisconsin. LACBWR, which is permanently shut down, is owned and licensed by Dairyland Power Cooperative (DPC or the licensee).

On April 27, 1987, DPC announced that LACBWR would be permanently shut down and decommissioned because of economic reasons and on April 30, 1987, the shutdown was completed. On June 12, 1987, all fuel had been removed from the reactor and stored in the Fuel Element Storage Well (FESW). By letter dated May 22, 1987, DPC requested that Provisional License No. DPR-45 for LACBWR be amended to a possession only status. Amendment No. 56 issued on August 4, 1987, established the possession only license status.

By letter dated December 21, 1987, as revised February 22, 1988, September 9, 1988, September 30, 1988, January 26, 1989, March 28, 1989, June 6, 1989, October 3, 1989, July 25, 1990, May 10, 1991, and July 25, 1991, DPC requested approval of its proposed Decommissioning Plan and corresponding renewal and amendment of License No. DPR-45. The plan involves long term storage (SAFSTOR) of LACBWR followed by dismantling and removal of residual radioactivity.

With the proposed Decommissioning Plan, DPC also proposed SAFSTOR Technical Specifications (TS) and a renewal of License No. DPR-45 to accommodate the proposed SAFSTOR period.

Following permanent shutdown of LACBWR and the issuance of the possession only license, DPC requested early approval of the Emergency Plan and a number of amendments to reduce TS requirements consistent with the possession only, defueled status of LACBWR. A revised emergency plan was approved on July 8, 1988, (Revision 10) and TS requirements were amended as follows:

1. Amendment No. 57 dated September 15, 1987, deleted certain inservice inspection requirements that were necessary for power operations only;

2. Amendment No. 58 dated January 4, 1988, approved a reduction in the required number of fire brigade members from five members to three;

3. Amendment No. 59 dated March 15, 1988, and an associated exemption deleted requirements for type A integrated leak rate testing of the containment building;

4. Amendment No. 60 dated April 11, 1988, reduced shift staffing requirements consistent with the possession only status of the LACBWR license;

5. Amendment No. 61 dated May 18, 1988, reduced security requirements;

6. Amendment No. 62 dated May 31, 1988, deleted TS definitions related to power operations and operator requirements that are needed only when fuel is in the reactor core;

7. Amendment No. 63 dated August 18, 1988, revised Possession Only License No. DPR-45 to a full term possession only license that expires on March 29, 2003. This amendment was issued in response to the licensee's application and a May 13, 1988 order from the Atomic Safety and Licensing Board, No. 78-368-05-OL;

8. Amendment No. 64 dated December 22, 1988, deleted TS requirements for Iodine-131 monitoring since with its short, 31 day, half-life no measurable quantities of this radionuclide are left on the LACBWR site; and

9. Amendment, No. 65 dated April 26, 1989, revised the TS requirements for storage of diesel fuel for the fire pump diesel engines. The amendment accommodated the installation and use of two new fuel tanks to replace older less reliable fuel tanks.

This safety evaluation considers the proposed Decommissioning Plan, the additional changes in the TS and the renewal of License No. DPR-45. DPC proposed a 30 to 50 year SAFSTOR period in the Decommissioning Plan but 10 CFR 50.51 restricts a license renewal to 40 years from the date of issuance. Therefore, to remain within this 40 year restriction, we are considering a license renewal to March 29, 2031. The present possession only license expires on March 29, 2003.

2.0 EVALUATION

DPC selected the SAFSTOR alternative for decommissioning because the spent fuel must remain on site until a Federal repository is available for it and because radiation exposure for workers will be significantly reduced with the delay in final dismantling. The on site storage of spent fuel is the major safety issue that is considered in this Safety Evaluation.

2.1 Spent Fuel Storage

All of the La Crosse spent fuel (333 assemblies) is stored in the fuel element storage well (FESW). The FESW is a stainless steel lined concrete structure that has inside measurements of 11 feet by 11 feet and is about 42 feet deep. Spent fuel assemblies are stored under water in two-tiered racks. The racks are designed with a nominal 7.0 inch center-to-center distance between fuel assemblies with boron slabs between each storage location to insure that fuel will remain subcritical in all situations.

FESW cooling is provided by a heat exchanger and two storage well pumps. The fission products in the fuel have, however, decayed to the point that natural circulation will provide adequate spent fuel cooling without either cooling pump running for seven days or more. An ion exchanger is provided for cleanup of the water.

The proposed fuel storage TS add two new specifications and bases to assure that the FESW water quality, depth, and temperature are maintained. The new FESW cooling water quality specifications are identical to the existing primary coolant purity requirements for cold shutdown except that a gross beta-gamma activity limit is added to detect any fuel cladding degradation. If the activity limits are exceeded and cannot be restored within seven days, a 30 day report will be made. Any serious degradation of the fuel cladding is considered to be a reportable event with immediate notification to the NRC. Weekly sampling and analysis is specified. A minimum FESW water level and maximum water temperature are specified with determinations to be made at least daily.

Specifications for a FESW backup water supply from either the demineralized water tank or the overhead storage tank are not changed.

The staff has reviewed the proposed TS for the FESW and has determined that the requirements are sufficient to assure that the spent fuel will be retained safely and that any deterioration of the fuel will be detected and corrected.

2.2 Containment Building

The LACBWR containment building remains unchanged for the SAFSTOR period. This building is a right circular steel cylinder with an internal height of 144 feet and an inside diameter of 60 feet. The steel shell thickness is 1.16 inches except for the upper hemispherical dome which is 0.60 inches thick.

The containment building contains the FESW and most of the equipment associated with the nuclear steam supply system. The TS requirement for containment integrity during any fuel handling activity remains the same as in the existing possession only TS. The definition of containment integrity also remains unchanged but some changes in containment TS have been made and are proposed as discussed below.

Amendment No. 59 to Possession Only License No. DPR-45, dated March 15, 1988, and an associated exemption relieved the licensee from 10 CFR 50.54(0) and Part 50, Appendix J requirements for integrated leakage rate testing. This was approved because the calculated worst case SAFSTOR accident was shown to result in less than one percent of the 10 CFR Part 100 whole body dose limit of 25 rem at the exclusion area boundary over a two hour release period.

In the SAFSTOR mode, there is no postulated accident that can produce pressure inside the containment building. Therefore, the need to test penetrations and isolation valves at the designed pressure of 52 psig is no longer necessary. The licensee has arbitrarily chosen 10 psig as a reasonable test pressure, which would give reliable indication that the containment components are intact.

Proposed TS require that all penetrations and containment isolation valves be leak tested at least once every 18 months if containment integrity is required for fuel handling. This specified frequency is the same as that required for LACBWR when it was an operating reactor. The combined leakage rate limit for all penetrations and valves is proposed to be 50 standard cubic feet per hour. This limit is equivalent to the Appendix J requirements for operating reactors. This is a very conservative leakage rate since the LACBWR containment would not be pressurized during any postulated accident.

Proposed TS requirements for isolation signals to the automatic containment isolation valves would require closure of the valves on high activity of the containment building air or on low FESW water level. Conservative setpoints for the air particulate or gaseous activity and FESW level have been included. When containment integrity is required for fuel handling, the automatic valve closures must be tested every 60 days. In addition, the containment ventilation inlet and outlet dampers are required to close within ten seconds and are subjected to quarterly leakrate testing whenever containment integrity is required.

By letter dated July 25, 1991, DPC indicated that hardware and wiring for the new containment isolation signal for water level in the Fuel Element Storage Well (page 4/5-8 of TS) would be installed as soon as possible, but no later than 6 months from the date of issuance of this amendment. We have determined that this is acceptable because, as stated in DPC letter, containment isolation from radiation levels continues in service and the containment will be manually isolated during any fuel handling. An appropriate notation has been made in the TS (page 4/5-8).

The SAFSTOR TS requires one of the two containment building vacuum breakers to be operable to relieve at a 0.5 psig negative differential pressure and the TS retain the requirement that the containment ventilation system be exhausted through particulate filters.

The Staff finds that the reduced leak testing pressure of 10 psig, the automatic containment isolation signal source changes, and the proposed surveillance test frequencies are appropriate for the LACBWR SAFSTOR period.

2.3 Inservice Inspection

Proposed TS delete surveillance requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components since with the reactor shutdown there are no longer any components of these classes at LACBWR. However, several systems still in use for the storage of spent fuel have been designated Class 3 systems in internal procedures and these systems will be maintained and inspected in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, Summer 1985 Addenda. The four designated Class 3 systems used to protect the spent fuel are the overhead storage tank system, component cooling system, demineralized water system, and fuel element storage well and cooling system.

Since the ASME Code classes address only those systems for the safe operation and shutdown of a reactor, the TS has been modified to include only those Class 3 components associated with the safe storage of spent fuel. The ASME Code requires that the structural integrity and operability of systems be maintained and tested according to code requirements.

The existing TS requirements for the structural integrity and inspection of the LACBWR ventilation stack and the adjacent coal plant, Genoa 3, smokestack remain unchanged.

The staff finds the proposed inservice inspection and testing adequate for LACBWR in its permanently shut down, defueled status.

2.4 Electrical Power Systems

The proposed SAFSTOR TS for the LACBWR electrical systems remains essentially the same as the current TS for cold shutdown conditions. The requirements for AC sources is identical except that only one emergency diesel generator (EDG) is required to be operable. The EDGs must be tested on a 31 day frequency and the time allowed for EDG voltage to reach the rated bus voltage has been increased from 10 seconds to 20 seconds. The ECCS actuation test start, the 24 hour test run at load, and the 10 year simultaneous test for EDG independence have been eliminated. The 120-Volt AC non-interruptible Bus 1A has been removed as an on site AC distribution system since it does not supply any required SAFSTOR equipment. TS have added the generator plant and reactor plant 125-Volt DC buses, battery banks, and chargers as required SAFSTOR systems. All operational TS remain except for the four hour battery charger capacity test and 60 month 80 percent battery discharge test. These surveillances have been eliminated since the battery loads in SAFSTOR have been significantly reduced.

The proposed TS reductions in the required electrical systems and their surveillances are acceptable since the most critical SAFSTOR requirement is the maintenance of spent fuel cooling and shielding. With complete loss of coolant flow at least seven days are available to make any repairs or other adjustments before the FESW coolant may begin to boil. This is sufficient time to repair at least one EDG or to obtain electrical power from another source.

2.5 Fire Protection

The proposed SAFSTOR TS for the fire protection systems at LACBWR remain essentially the same as for the operational TS with the following exceptions: TS requirements for fire detectors in the turbine exciter room and turbine building air intake louver have been deleted; only one of the two diesel driven fire pumps is required to be operable; TS requirements for sprinkler systems for the alternate core spray valve area and main transformer sprinkler systems have been deleted; requirement for a fire hose station in the turbine building main steam stop valve area has been deleted; and the TS for penetration fire barriers has been removed. The Waste Treatment Building hose station has been added to the TS requirements.

Except for the fire detection instrumentation section, the applicability statements for the fire protection sections have been changed to "at all times" instead of "whenever equipment in that area is required to be operable." The action statements have also been modified to agree with the above changes. The continuous fire watch action for the inoperable sprinklers and/or chemical extinguishers has been changed to a one hour fire watch. Fire protection surveillance requirements have been modified to agree with the above changes.

The above proposed changes only remove the TS requirements for those areas that are not now safety related. Also, since the diesel fire pumps are no longer required for the alternate core spray backup water supply, the staff agrees that only one diesel fire pump is needed to assure adequate fire protection backup in the SAFSTOR mode. Therefore, the staff finds the proposed fire protection TS acceptable.

2.6 Post Accident Radiation Monitoring Instrumentation

The proposed SAFSTOR TS deletes radioiodine monitoring from the post-accident monitoring system because radioiodine is no longer present in any significant quantity. The stack gas alarm setpoint for the noble gas effluent monitor has been reduced from 1.25×10^2 μ Ci/cc to 0.1 μ Ci/cc for Krypton-85. The high range monitor setpoint in the containment building has been reduced from 10^4 R/hr to 10^2 R/hr. These instruments are required to be operable at all times. Action statements have been relaxed from 4 hours to 24 hours for declaring a channel inoperable. The staff finds that these proposed TS provide adequate requirements for post accident radiation monitoring and are consistent with the permanent shutdown/defueled, SAFSTOR status of LACBWR.

2.7 Radioactive Effluents

The following significant changes have been made in the radioactive effluents section of the SAFSTOR TS:

(1) The TS concentration limit for dissolved or entrained noble gases in liquid effluents has been changed from 2×10^{-4} μCi/ml to 6×10^{-4} μCi/ml because the only remaining noble gas present is Krypton-85 which has a reduced biological impact;

(2) The cumulative dose contributions from liquid, gaseous, and noble gaseous wastes will be determined quarterly instead of 31 day intervals;

(3) Iodine-131 and other short half-life isotopes have been deleted from the liquid gaseous waste analysis program since they are no longer present in significant quantities; and

(4) The containment building sampling systems are now required to be operable only when containment integrity is required for fuel handling.

The licensee stated that the approved Offsite Dose Calculation Manual (ODCM) and Health and Safety Procedures will be revised to delete reference to short half-life isotopes and to include changes being made to the TS.

The staff finds that the proposed TS for the Radioactive Effluent section are acceptable since the reduction in action timing and reporting frequency are commensurate with the shutdown, SAFSTOR status of the facility.

2.8 Radiological Environmental Monitoring Program and Interlaboratory Comparison

The proposed SAFSTOR TS for the radiological environmental monitoring program are essentially the same as for the current TS and the commitment for annual analysis in an interlaboratory comparison program remains unchanged.

The number and frequencies of airborne particulates, direct radiation, and milk samples have been reduced as follows:

	Number of Samples	Frequency
Airborne particulates	5 to 3	Weekly (No Change)
Direct Radiation (TLDs)	35 to 8	Quarterly to Semiannually
Milk	3 to 1	Biweekly to Monthly

The analysis and reporting levels for iodine and other short half-life isotopes have been eliminated as has the biennial land use census requirement. Since the reactor has been shutdown, Iodine-131 and other short half-life isotopes are no longer present in significant quantities.

These changes are acceptable to the staff since the reactor has been shutdown for over 36 months and essentially all the Iodine-135 and other short lived fission products have decayed to stable isotopes. The reduced frequency and number of environmental sample are commensurate with the reduced possibility of release of radioactivity while the facility is in the SAFSTOR mode.

2.9 Radiation Protection

2.9.1 Radiological Safety Organization

The Health and Safety Supervisor is responsible for the radiological health and safety of the general public in the area surrounding the plant and is also responsible for the safety of the staff and all visitors to the plant. The Health and Safety Supervisor ensures that all long-term radiological and environmental surveillance programs in the SAFSTOR operation are carried out and that proper reports are maintained on radiation exposure throughout the facility. This individual ensures that all radiation exposure controls are in place and that contamination and daily, monthly, and annual exposure limits on personnel are complied with. The Health and Safety Supervisor is responsible for the ALARA program and will ensure that all personnel stationed at or visiting LACBWR comply with its regulations. This supervisor will also assign the day-to-day duties of the Health Physics Technicians.

The Health Physics Technicians are responsible for the radiation protection and chemistry programs at LACBWR. They maintain the exposure records of personnel as required, perform the necessary radiation surveys to control the spread of contamination and provide input to the long-term radionuclide inventory program.

The qualifications of the incumbent Radiation Protection Engineer at LACBWR meet the requirements of Regulatory Guide 1.8, "Personnel Selection and Training," for the Radiation Protection Manager (RPM) position. His duties include supervising the activities of the health physics group, as well as implementing and enforcing the plant's health physics program. In addition, the plant RPM is responsible for ensuring that contract personnel have enough training and experience in plant radiation protection procedures. This engineer is responsible for working with the Health and Safety Supervisor to prepare the procedures necessary for decontamination, waste management, chemical control, and fuel shipment. The RPE reports directly to the Plant Superintendent and is also a member of the Operations Review Committee and the Safety Review Committee.

The staff finds that the qualifications of the LACBWR health physics management meet the acceptance criteria of NUREG-0800 and Regulatory Guide 8.8 (Revision 3) and are acceptable.

2.9.2 Dose Assessments

DPC has estimated that the collective dose equivalent to the plant staff for the 30 years of SAFSTOR period followed by delayed DECON is not expected to exceed 500 person-rem. The licensee has provided a dose assessment, including a completed summary table of occupational radiation exposure estimates and sufficient detail to explain how the assessment process was performed. A graph is also included showing annual personnel whole body doses expected during SAFSTOR from 1988 to 2014. The staff considers this to be a reasonable estimate and acceptable.

2.9.3 Radiation Protection Program

The radiation protection program used during the SAFSTOR period is an extension of the program that was used during the period of reactor operations at LACBWR. The program adequately addresses the essential elements of 10 CFR Part 20. Implementation of the radiation protection program at LACBWR will be accomplished through health and safety procedures. The licensee has committed to using contract personnel when additional health physics help is required. The staff finds this to be reasonable and acceptable. The responsibilities of the health physics group includes performing area and airborne radioactivity surveys, administering the respiratory protection program, maintaining a supply of radiation protection equipment and instrumentation, personnel dosimetry, and maintaining radiation records.

2.9.4 Access Control and Postings

Personnel access is controlled in areas where radiation exposure is possible. This control consists of a system of physical barriers, warning signs, and signals in accordance with 10 CFR Part 20.

A Radiation Work Permit (RWP) is issued to authorize personnel to perform work of a non-routine nature that involves radiation hazards. RWPs are used to inform personnel of these hazards and the required protective measures.

2.9.5 Radiation Monitoring

A program for routine surveys and monitoring is continued during the SAFSTOR period at LACBWR. This program will continue to ensure that all personnel are aware of the possible hazards involved before entering a potential radiation area or a potentially contaminated area. This program ensures that the potential hazards are adequately defined and that adequate controls are instituted to minimize radiation exposure of personnel working in radiation areas or with radioactive materials.

Radiation surveys are conducted by the Health and Safety Department to determine dose rates in general areas. The Health and Safety Department also monitors areas to locate any radiological hot spots. Surveys are performed on a routine basis as established by procedures.

Special radiation surveys of particular items or areas will be performed on an "as needed" basis. Examples of special radiation surveys are the removal of equipment or materials from a controlled area, leak testing of sealed radioactive sources, or the shipment or receipt of radioactive material packages. Radiation surveys to determine area contamination levels will be conducted routinely by the Health and Safety Department as established by procedures.

2.9.6 Health Physics Equipment and Facilities

The radiation protection equipment and facilities in place at LACBWR during SAFSTOR are similar to these that existed during plant operation. These facilities include a counting room, radiation laboratory, calibration facility, personnel decontamination room, and a respirator cleaning facility. Equipment to be used for radiation protection purposes includes portable radiation survey instruments, personnel monitoring equipment, fixed and portable area and airborne radioactivity monitors, laboratory equipment, air samplers, respiratory protective equipment, and protective clothing. The number and types of equipment to be used are adequate and provide reasonable assurance that the licensee will be able to maintain occupational exposure ALARA during SAFSTOR.

2.9.7 Health Physics Procedures

The proposed TS require dosimeters for all plant personnel entering radiation areas. Whole-body counts of all plant personnel who enter restricted areas will be conducted on a scheduled basis and other bioassays will be provided when deemed necessary by the plant's health physics staff, using the guidance of Regulatory Guide 8.26. All radiation exposure information will be processed and recorded in accordance with 10 CFR Part 20.

2.9.8 Summary (Radiation Protection)

On the basis of the plant's health physics organization, equipment, and procedures, staff positions and Regulatory Guides, the staff concludes that the radiation protection program for SAFSTOR will maintain in-plant radiation exposures within the limits of 10 CFR Part 20 and ALARA in accordance with Regulatory Guide 8.8 (Revision 3), and is, therefore, acceptable.

2.10 Waste Management

2.10.1 Liquid and Gaseous Waste

Radioactive liquid waste generated during the SAFSTOR period is processed and disposed of. The normal and potential sources of liquid wastes during SAFSTOR are the following:

(1) Turbine building floor drains and sumps (laundry water, decontamination liquids);

(2) Waste treatment building sump (decontamination liquids, resin dewatering liquids); and

(3) Containment building sumps and retention tanks (fuel element storage well leakage, air conditioner condensate, decontamination liquids, and system draining evolutions).

After liquid waste is collected in a tank, the tank's contents are recirculated. After recirculation, a sample is withdrawn and analyzed for radioactivity concentrations prior to discharge. The tank's contents are discharged through a backwashable filter into the circulating water system. The discharged water is diluted by the Genoa 3 circulating water discharge and the waters of Thief Slough prior to mixing with the Mississippi River.

Ventilation exhausts from the turbine building, waste treatment building, and containment building join at the stack plenum prior to discharge up the stack. One or two stack blowers, rated at 35,000 cfm each, act as a driving force for stack releases and add dilution air within the stack prior to discharge. The off gas system from the condenser is no longer in operation, therefore, the majority of gaseous releases are from the various ventilation exhaust systems. The containment building and waste treatment building exhaust systems are equipped with full flow HEPA filter banks.

The low-level waste storage building is not equipped with a ventilation system. All waste stored in this building is normally sealed and the outer surfaces of the containers are free of significant removable contamination.

The only radionuclides expected to be released in liquid effluents in quantities which may have significant contributions to calculated off site doses are the following (the annual activities estimated to be released, in curies, are in parentheses): Cobalt-60 (0.2), Iron-55 (0.05), Cesium-134 (0.001), Cesium-137 (0.02), and Tritium (1). No radionuclides are expected to be released in gaseous effluents in quantities which may have significant contributions to calculated off site doses.

Proposed TS require that liquid and gaseous effluents be controlled to limit concentrations of radioactive materials in areas beyond the site property line such that the concentrations and dose rates are within the limits of 10 CFR Part 20. TS also require that radioactive material in liquid and gaseous effluents be limited to maintain off site doses within the 10 CFR Part 50, Appendix I, design objectives for keeping these releases "as low as is reasonably achievable." The 1990 Radioactive Effluent Report for LACBWR* showed that the calculated exposures for both liquid and gaseous effuents were less than one percent of the Appendix I TS limits. Also, off site monitoring results as tabulated in the report indicated no significant radioactivity from LACBWR.

In its evaluation of the liquid and gaseous radioactive waste management systems, the staff considered (1) the capability of the systems to maintain releases to unrestricted areas below the limits in 10 CFR Part 20, (2) the capability of the systems to control releases so that doses to members of the public are within the design objectives of 10 CFR Part 50, Appendix I, to meet the criteria of "as low as is reasonably achievable," and (3) the design features that are incorporated to control the releases of radioactivity material in accordance with GDC 60.

The staff concludes that the design of the liquid and gaseous radioactive waste management systems are acceptable and meet the requirements of 10 CFR 20.106 and GDC 60. This conclusion is based on the following:

(1) The licensee has satisfied the requirements of 10 CFR 20.106. The staff has determined that the concentrations of radioactive materials in liquid and gaseous effluents in unrestricted areas can be maintained to a small fraction of the limits of 10 CFR Part 20, Appendix B, Table II;

(2) The staff has determined that releases of radioactive materials can be maintained to keep doses to members of the public to within the 10 CFR Part 50, Appendix I, design objectives; and

(3) The licensee has satisfied the requirements of GDC 60 with respect to controlling the releases of radioactive materials to the environment.

2.10.2 Solid Waste

During the SAFSTOR period, the radioactive solid wastes which are processed for shipment to a suitable disposal facility are low-level radioactive wastes principally with radioactivity content less than Class C (10 CFR Part 61) wastes. This will include:

(1) Dry active wastes;

(2) Dewatered spent demineralizer resins and filtration media; and

*Berg, W. L., DPL, letter to A. B. Davis, NRC, "Radiological Effluent Report and Environmental Monitoring Report for LACBWR 1/1/90 to 12/31/90," dated February 20, 1991.

(3) Contaminated or irradiated plant system components.

All solid radioactive wastes are processed to meet shipping and burial ground requirements in accordance with the Process Control Program, as required by the TS.

Solid radioactive waste material is classified in accordance with 10 CFR 61.55 by periodically sampling and analyzing the various waste streams. The criteria of 10 CFR 61.56 is used to process and package solid radioactive waste. For example, dewatered spent demineralizer resins and filtration media is packaged in containers that provide structural stability for waste disposal.

The licensee estimates that the quantity of dry active waste and dewatered resins will not exceed 10 cubic meters per year.

The staff concludes that the provisions for the processing and packaging of solid radioactive waste to meet the requirements of 10 CFR Parts 20, 61, and 71 are acceptable and, therefore, the design of the solid waste management system is acceptable.

2.11 Administrative Controls

The SAFSTOR TS in the Administrative Controls section remain essentially the same as those in the existing possession only TS. Responsibilities of facility staff, overtime policy, and staff qualifications are unchanged. The training section has been modified by deleting the reference to 10 CFR Part 55 and removing the Technical Support Engineer as the director of the Fire Brigade Training program. A Certified Fuel Handler training program taken from INPO's accredited operator training program procedure has been included in the TS as well as General Employee Training and Health Physics Technician Training and Retraining.

The off site and unit organization charts (Figures 6.2.1-1 and 6.2.2-1 in the possession only TS) have been deleted. This is acceptable since these charts are part of the approved Quality Assurance Plan which is updated annually and reviewed by the NRC regional office.

The Operations Review Committee (ORC) composition has been modified, specifying membership by staff positions and not by specific position titles and the minimum quorum has been reduced from the chairman and four members to the chairman and three members, due to the small organization of experienced personnel. There are only three departmental supervisors and three staff engineers remaining and at the most two shift supervisors would be available for ORC meetings. The elimination of the TS for specific positions will allow position title changes or further organization consolidation without a TS amendment. The minimum meeting frequency has been reduced from monthly to quarterly. This is in line with the reduced activities at LACBWR in the SAFSTOR mode. The Process Control Program, Transportation Packaging Program, and the Decommissioning Plan have been added to the review responsibilities of the ORC. The other responsibilities remain unchanged.

The composition, function, meeting frequency, and quorum requirements of the Safety Review Committee (SRC) remains unchanged. Waste Management has been added to the SRC's independent review and audit functions. The review of changes to the Plant Security Plan, Emergency Plan, and the Decommissioning Plan have been added to the SRC responsibilities. Other review responsibilities remain unchanged as well as the responsibility for the Audit Program. The audit areas remain the same except that the three year audit of the fire protection program has been deleted. Two other biennial audits of the fire prevention program remain, one for review of program and procedures, the other an independent inspection and audit of the fire protection program. The audit of the Emergency Plan and the Security Plan remain at an annual frequency. The frequencies of several of the audits have been decreased. The audit frequency for corrective actions has been decreased from every six months to annually. The audit frequency for conformance to TS and other license conditions has been decreased from annually to biennially as have audits of the Fire Protection-Prevention Plan, and the Radiological Environmental Monitoring Plan. The frequency of audits of Training and Staff Performance, the QA plan, Fire Prevention Program, and Off site Dose Calculation Manual and Process Control Program remains biennial.

These audit frequencies agree with Section 4.5, Audit Program of ANSI Standard 18.7-1976/ANS-3.2 which states audits of all safety-related functions be completed within a two year period. The audit frequencies do not agree with Regulatory Guide 1.33, QA Program Requirements for Operating Nuclear Power Plants, which recommends increased frequencies for these areas. The staff agrees that the reduced risks and activities in the SAFSTOR mode do not warrant the more frequent audits recommended by Regulatory Guide 1.33.

Reporting requirements have been modified to eliminate the startup reporting requirements and other references to operations but retaining all other semiannual and annual reporting requirements.

Since LACBWR is no longer an operating reactor, the immediate reporting requirements of 10 CFR 50.72 no longer apply to them; however, most of the Licensee Event Reports (LER) of 10 CFR 50.73, that are not concerned with an operating reactor have been included in the LACBWR reporting requirements. The licensee has committed to four hour reports to the NRC Operations Center if any of the events listed occur and one hour reports whenever a listed Emergency Class in the approved Emergency Plan has been declared.

Procedures currently in use will continue with the exception that references must be changed to indicate the new TSs. As stated in DPC letter of July 25, 1991, these procedure references will be updated to reference the new SAFSTOR TS as each procedure comes due for its annual or biennial review. We have determined that this schedule for updating of reference numbers is acceptable. An appropriate notation has been made in the TS (page 6-8).

The staff has reviewed the proposed Administrative TS and finds that they provide essentially the same requirements as those for an operating reactor as specified in 10 CFR 50.36 and, therefore, they are acceptable.

2.12 Financial Considerations

DPC's plan for financing LACBWR's period of safe storage and eventual decontamination and dismantlement is contained in Section 6.7 of the "Decommissioning Plan." DPC provided additional information in (1) its September 30, 1988 response to NRC letter dated July 7, 1988, and (2) its January 26, 1989 response to the NRC letter dated January 7, 1989. DPC provided the LACBWR decommissioning funding report on July 25, 1990.

From these documents, the staff has determined that DPC intends to fund from current income the expenses needed to maintain LACBWR in a safe storage mode. Annual costs during the SAFSTOR period were approximately $2.5 million in 1989 and are projected to escalate at 4.8 percent per year. Annual costs would be significantly reduced after spent fuel shipment, which is projected to occur before 2010. DPC also has established an external sinking fund to which they will contribute $1.3 million annually until 1999. By January 1, 2010, DPC projects this fund, including accumulated earnings, will accrue to $92 million. The $15.9 million already collected since 1983 has been deposited in this external sinking fund. DPC estimates that this amount will be sufficient to dismantle LACBWR in 2010 to the point of terminating the license and releasing the site for unrestricted use. However, DPC intends to retain LACBWR in a SAFSTOR status beyond 2010 to allow additional radioactive decay. DPC has committed to reevaluating decontamination and dismantlement cost estimates every five years and DPC's Board of Directors has resolved to increase the amount of contributions based on these reevaluations.

Although DPC did not provide a detailed cost analysis for final dismantling of LACBWR, DPC concluded that it would cost $91 million to DECON LACBWR in 2010, based on studies by Nuclear Energy Services, Inc. Since the DPC external fund has been established to accumulate $92 million by 2010 we have used that figure as the estimated cost for LACBWR DECON. The estimate of $92 million in 2010 dollars for final dismantling seems reasonable when compared to the cost of dismantling the Pathfinder Facility, a 190 MWt BWR. Our cost comparison is shown below.

	Initial DECON $ Spent	Final DECON Cost	Total DECON Cost*	MWD of Oper.	Years in SAFSTOR
Pathfinder 190 MWt BWR	$1.9 M (1971 $)	$13 M (1991 $)	$19.4 M (1991 $)	16.6 K Thermal	24 yr actual
LACBWR 165 MWt BWR	None	$92 M (2010 $)	$30.3 M (1991 $)	560 K Thermal	30/50 yr proposed

*The total DECON cost for Pathfinder includes $1.9 million spent in 1971 for initial DECON as increased by inflation. Based on the "Consumers Price Index" costs increased by a factor of 3.31 from 1971 to 1991. DPC assumed a 6 percent cost escalation factor for LACBWR and that factor was used in converting 2010 dollars to 1991 dollars.

The above table compares decommissioning costs for the Pathfinder BWR with estimated costs by DPC for decommissioning LACBWR. Since the Pathfinder decommissioning/DECON is nearing completion, its cost estimates are reasonably accurate. Although the total MW days (MWD) of operation for LACBWR are significantly greater than for Pathfinder, the reactors are both Allis-Chalmers BWR's, both are approximately the same size, and the additional $10.9 million (1991 dollars) set aside for LACBWR would reasonably be expected to cover the additional cost of DECON for LACBWR in 2010 or later. Furthermore, if DPC elected a 50 year SAFSTOR period for LACBWR, the inventory of Cobalt-60 would decrease by a factor of 32 (5 more half-lives) over a DECON starting in 2010 (23 years after shutdown). Cobalt-60 is the dominant radionuclide with respect to worker exposure, radioactive waste volume, and costs for both plants.

In its review, the staff has determined that DPC's funding plan provides reasonable assurance of the decommissioning of LACBWR. This conclusion is based on the assumption that DPC will follow its funding plan as it outlined above and will adhere to its commitment to adjust its contributions to an external fund as decommissioning cost estimates change. The Commission's new decommissioning regulations (53 FR 24018, June 27, 1988 at p. 24027, col. 1) state that for reactors already shut down, "details concerning financial assurance ..... would be decided on a case-by-case basis."

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The state official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in TS requirements to reflect the permanent shutdown, defueled status of LACBWR. An Environmental Assessment has been completed by the NRC staff. A Notice of Issuance of Environmental Assessment and Finding of No Significant Impact has been published in the FEDERAL REGISTER. Based on the Environmental Assessment the Commission concluded that the proposed amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSIONS

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: August 7, 1991

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DAIRYLAND POWER COOPERATIVE

LA CROSSE BOILING WATER REACTOR (LACBWR)

DOCKET NO. 50-409

NOTICE OF ISSUANCE OF ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is considering the issuance of an order authorizing decommissioning and issuance of a renewal and amendment of Possession Only Licensee No. DPR-45 for Dairyland Power Cooperatives (DPC's) La Crosse Boiling Water Reactor (LACBWR). The Decommisioning Plan involves long-term storage of the facility (SAFSTOR) followed by dismantlement.

Description of Proposed Action

LACBWR has been shut down since April 30, 1987. All spent fuel has been removed from the reactor and is stored in the LACBWR Fuel Element Storage Well. Approval of the Decommissioning Plan and renewal of License No. DPR-45 will allow storage of spent fuel on site until a Federal repository is available and SAFSTOR of LACBWR until March 29, 2031.

Environmental Impacts

The NRC staff has reviewed the proposed decommissioning, the proposed renewal and amendment of the possession only license and DPC's Supplemental Environmental Report with respect to 10 CFR 51.53(b). To document its review, the staff has prepared an Environmental Assessment. The proposed SAFSTOR of LACBWR will prevent unrestricted use of a small area of land for the SAFSTOR period but will result in a reduced exposure for workers that do final dismantling and also result in a reduced volume of radioactive waste.

Finding of No Significant Impact

The staff has reviewed the proposed decommissioning, license renewal and amendment relative to the requirements set forth in 10 CFR Part 51. Based upon the Environmental Assessment, the staff concluded that there are no significant environmental impacts associated with the proposed action and that the proposed action will not have a significant effect on the quality of the human environment. Therefore, the Commission had determined, pursuant to 10 CFR 51.31, not to prepare an environmental impact statement for the proposed amendment.

For further details with respect to this action, see: (1) the licensee's application for authorization to decommission the facility, dated December 21, 1987, as revised February 22, 1988, September 9, 1988, September 30, 1988, January 26, 1989, March 28, 1989, June 6, 1989, October 3, 1989, July 25, 1990, May 10, 1991, and July 25, 1991; (2) Amendment No. 66 to License No. DPR-45; (3) the Commission's related Safety Evaluation; and (4) the Environmental Assessment and Finding of No Significant Impact. These documents are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555, and at the La Crosse Public Library, 800 Main Street, La Crosse, Wisconsin 54601. Copies of items (2), (3), and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Advanced Reactors and Special Projects.

Dated at Rockville, Maryland this 31st day of July 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard F. Dudley, Jr.

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Non-Power Reactors, Decommissioning and
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