

June 19, 1987

In reply, please
refer to LAC-12265

DOCKET NO. 50-409

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Gentlemen:

SUBJECT: Dairyland Power Cooperative
La Crosse Boiling Water Reactor (LACBWR)
Provisional Operating License DPR-45
Application for Amendment to License

REFERENCES: (1) 10 CFR 50, Section 50.90
(2) 10 CFR 170.12
(3) DPC Letter, Taylor to Docket Control Desk,
LAC-12234, dated May 22, 1987

Dairyland Power Cooperative requests revisions to LACBWR Technical Specifications. Therefore, in accordance with the provisions of Reference 1, an application to amend Provisional Operating License No. DPR-45 for the La Crosse Boiling Water Reactor is hereby filed.

Reference 3 contained a request to amend the plant's license to a possession-only license. LACBWR was permanently shut down on April 30, 1987. Reactor defueling was completed June 11, 1987. Due to the permanent shutdown, changes to LACBWR Technical Specifications are needed. This letter requests the first group of changes which are needed.

The revisions requested at this time fall into 3 categories: Inservice Inspection (ISI), non-operation of the reactor, and operability of Nuclear Instrumentation. The proposed deletion of Sections 3.0.10:f and 3.0.11 eliminates the need to perform ISI on high pressure primary system piping and the requirement to use NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," to govern repair and replacement of piping. The proposed revision to Section 5.2.2.23 limits the performance of ISI to demonstration of component structural integrity for those ASME Code Class 1, 2 and 3 components utilized in connection with fuel storage. These changes support the ALARA concept by eliminating inspection of systems which will no longer be required to be operable. ISI typically accounts for 10% of the annual radiation exposure at LACBWR.

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The proposed revisions to Sections 1.3 and 4.2.4.1 reflect that the reactor will not be operated. DPC also proposes to delete Section 4.1.3 which requires Nuclear Instrumentation Channel 1 or 2 to be in operation, because there is no fuel in the reactor and therefore no neutrons to be monitored.

Attachment 1 to this letter contains the current and proposed versions of the involved specifications and justifications for each individual change. Attachment 2 contains the proposed revised pages.

Finding of No Significant Hazards

We have reviewed the hazards considerations referenced in 10 CFR 50 Sections 91 and 92 and have determined that, with these criteria, no significant hazards considerations result from this proposed amendment.

1. Operation of LACBWR in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Eliminating the requirement to keep a Nuclear Instrumentation channel in operation will have no effect on accident probability or consequences, since the reactor has been permanently shut down and defueled. Removing from the Technical Specifications that the reactor can be operated is strictly an administrative change to support the proposed license amendment contained in Reference 3. The probability of most previously analyzed accidents is decreased with the reactor shut down. Terminating inspection of piping that will no longer be used as a pressure boundary and components which will no longer need to be utilized cannot increase the probability or consequences of an accident, since there will be no significant adverse consequences if these components fail. Additionally, removing requirements dealing with NUREG-0313, which applies to BWR Coolant Pressure Boundary piping above 200°F, will have no effect on the probability or consequences of an accident, since the plant will not be in the condition for which the NUREG was written. This, too, is therefore an administrative change. For these reasons, the proposed Technical Specification amendment will not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of LACBWR in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The only method of operation being affected by this change is that the reactor will not be able to operate. This was previously proposed in the license amendment contained in Reference 3. Non-operation will not

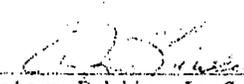
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cc: Mr. A. Bert Davis, Region III
NRC Resident Inspector
Mr. Peter Erickson, LACBWR Project Manager

Mr. L. L. Smith, Director
Electric and Water Bureau
Wisconsin Public Service Commission
P. O. Box 7854
Madison, WI 53707

STATE OF WISCONSIN)
)
COUNTY OF LA CROSSE)

Personally came before me this 23rd day of June, 1987,
the above named, James Taylor, to me known to be the person who executed the
foregoing instrument and acknowledged the same.


Notary Public, La Crosse County Wisconsin

My commission expires March 12, 1987

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create the possibility of a new type of accident. Eliminating the need to monitor a defueled reactor with Nuclear Instrumentation cannot cause a new type of accident. Terminating inspection of equipment not required to be operable cannot cause a new type of accident, since unneeded equipment poses no new accident threat. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Operation of LACBWR in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

Cessation of plant operation will not cause a significant reduction in the margin of safety. Eliminating neutron monitoring of a reactor with no fuel assemblies in it cannot reduce the margin of safety. Termination of Inservice Inspection on piping and components associated with the defueled reactor and related systems will have no effect on the margin of safety. Deletion of requirements pertaining to NUREG-0313 will have no effect on the margin of safety, since the NUREG applies to a reactor coolant pressure boundary above 200°F. For these reasons, the proposed amendment will not involve a significant reduction in the margin of safety.

As determined by the analysis above, this amendment has no significant hazards consideration.

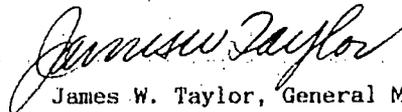
The license amendment has been reviewed by the appropriate review committees.

The application fee required by Reference 2 for processing a license amendment is enclosed.

If there are any questions, please contact us.

Sincerely,

DAIRYLAND POWER COOPERATIVE



James W. Taylor, General Manager

JWT: LSG: dh

Enclosures

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ATTACHMENT 1

CURRENT AND PROPOSED SPECIFICATIONS

PAGE 1

CURRENT

1. SITE

1.3 PRINCIPAL ACTIVITIES

The principal activities carried on within the exclusion area shall be the operation of the reactor and associated power-generating equipment, and the operation of conventional steam electric plants and electrical transmission switching and distribution centers, all of which shall be located within the site boundary.

PROPOSED

1. SITE

1.3 PRINCIPAL ACTIVITIES

The principal activities carried on within the exclusion area shall be the possession of the reactor and associated power-generating equipment, and the operation of conventional steam electric plants and electrical transmission switching and distribution centers, all of which shall be located within the site boundary.

JUSTIFICATION

The reactor and associated power-generating equipment would be possessed by the licensee but not operated.

CURRENT

3.0.10 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

- f. Inservice inspection of all non-conforming service sensitive lines, identified per NUREG-0313, shall be performed at 36-month intervals (plus or minus by as much as 12 months) coinciding with a refueling or other outage. In the event these examinations reveal no unacceptable indications within three successive inspections, the schedule may revert to examinations per Specifications 5.3.1, with the exception that the required examinations should be completed during each 80-month period.

PROPOSED

Delete paragraph 3.0.10.f.

JUSTIFICATION

The inservice inspection of nonconforming service sensitive high pressure piping is not required, as the reactor portion of the plant will not be pressurized and, therefore, such inservice inspection is not necessary.

CURRENT

3.0.11 All replacement of and repairs to ASME Code class 1, 2, and 3 lines shall conform to the guidelines stated in the NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", July 1980.

PROPOSED

Delete Section 3.0.11.

JUSTIFICATION

NUREG-0313 applies only to BWR Coolant Pressure Boundary Piping above 200°F. Since the reactor will not be operated and therefore the reactor coolant piping will not be above 200°F, NUREG-0313 does not apply.

PAGE 27c

CURRENT

BASES

3.0.11 This specification ensures that replacement piping and repairs to ASME Code Class 1, 2, and 3 lines conform to the guidelines stated in NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", July 1980. Application of these guidelines will help reduce intergranular stress corrosion cracking (IGSCC) susceptibility.

PROPOSED

Delete the bases for Section 3.0.11.

JUSTIFICATION

Section 3.0.11 is being deleted.

PAGE 28

CURRENT

4.1.3 During periods when the reactor is in Condition 3, 4 or 5, either Channel 1 or 2 of the Nuclear Instrumentation System shall be in operation and shall be monitored by the operator.

PROPOSED

Delete this section.

JUSTIFICATION

Once the fuel is removed from the reactor, the Nuclear Instrumentation System has no function and there will not be sufficient neutrons present to demonstrate operability. The status of the facility when fuel has been removed to the storage well is not clearly defined under Operational Condition 3, 4 or 5. If Condition 4 is considered to apply, this specification would be applicable, but not achievable or necessary.

PAGE 32e(1)

CURRENT

5.2.2.23 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated per the requirements of Specification 3.0.10.

PROPOSED

5.2.2.23 The structural integrity of ASME Code Class 1, 2 and 3 components utilized in connection with fuel storage shall be demonstrated per the requirements of Specification 3.0.10.

JUSTIFICATION

Only those systems utilized for fuel storage require maintenance of structural integrity.

PAGE 32f(1)

CURRENT

BASES

4/5.2.2.23 STRUCTURAL INTEGRITY

The inspection and testing program conforms to the requirements of 10 CFR 50, Section 50.55a(g). To the extent practical, the inspection and testing of components classified into Class 1, 2, and 3 will conform to the requirements for ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code, "Inservice Inspection of Nuclear Reactor Coolant Systems," 1974 Edition, Summer 1975 Addenda.

PROPOSED

BASES

4/5.2.2.23 STRUCTURAL INTEGRITY

The inspection and testing program conforms to the requirements of 10 CFR 50, Section 50.55a(g). To the extent practical, the inspection and testing of components utilized in connection with fuel storage classified into Class 1, 2, and 3 will conform to the requirements for ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code, "Inservice Inspection of Nuclear Reactor Coolant Systems," 1974 Edition, Summer 1975 Addenda.

JUSTIFICATION

Only those systems utilized for fuel storage require maintenance of structural integrity.

PAGE 32s

CURRENT

4.2.4 Reactor Core

4.2.4.1 The reactor shall not be operated at steady power levels in excess of its rated power of 165 Mwt.

PROPOSED

4.2.4 Reactor Core

4.2.4.1 The reactor shall not be operated.

JUSTIFICATION

DPC Management decision.

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PROPOSED REVISED PAGES

ATTACHMENT 2

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 EXCLUSION AND RESTRICTED AREAS

1.2.1 The distance from the centerline of the reactor building to the boundary of the exclusion area, as defined in 10 CFR 100, shall be at least 1109 feet. Any changes in occupancy of the exclusion area which lead to residential uses shall be noticed to the NRC.

1.3 PRINCIPAL ACTIVITIES

The principal activities carried on within the exclusion area shall be the possession of the reactor and associated power-generating equipment, and the operation of conventional steam-electric plants and electrical transmission switching and distribution centers, all of which shall be located within the site boundary.

2. DESIGN AND PERFORMANCE REQUIREMENTS

2.1 REACTOR BUILDING

2.1.1 Containment Vessel

2.1.1.1 The containment vessel shall be capable of containing an internal pressure of 52 psig at 280°F, and it shall be capable of withstanding an external-over-internal pressure of 0.5 psi.

2.1.1.2 The leakage rate from the containment vessel shall correspond to a leakage rate that, per 24-hour period, does 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.

3.0 APPLICABILITY

SURVEILLANCE REQUIREMENT - (Continued)

ASME Boiler and Pressure Vessel
Code and applicable Addenda
Terminology for inservice in-
spection and testing activities

Required frequencies
for performing inservice
inspection and testing
activities

Weekly
Monthly
Quarterly or every 3 months
Semiannually or every 6 months
Every 9 months
Yearly or annually

At least once per 7 days
At least once per 31 days
At least once per 92 days
At least once per 184 days
At least once per 276 days
At least once per 366 days

- c. The provisions of Specification 3.0.7 are applicable to the above required frequencies for performing inservice inspection and testing activities.
- d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.
- e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.
- f. Deleted.

3.0.11 Deleted.

3.0 APPLICABILITY

BASES - (Continued)

3.0.10 This specification ensures that inservice inspection of ASME Code Class 1,2, an 3 components and inservice testing of ASME Code Class 1,2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies of performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

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 3.0.9 to perform surveillance activities prior to entry into an OPERATIONAL CONDITION or other 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.

3.0.11 Deleted.

4.1 GENERAL

4.1.3 Deleted.

4.1.4 Whenever the reactor contains one or more fuel elements, any operations from points outside the control room of equipment which may affect the reactor shall be conducted under the direction, or with the knowledge, of the control room operator.

4.1.5 If the plant is operational during a tornado warning, the Shift Supervisor on duty shall keep informed of the actual tornado activity which may approach the plant. In the event that reports indicate an imminent tornado strike at or near the LACBWR plant, the Shift Supervisor shall reduce reactor power to a level which permits prompt reduction of power generation to station load. However, the Shift Supervisor shall be instructed to discontinue plant operation if, in his judgment, this action is required to ensure plant safety.

4.1.6 If the plant is in CONDITION 1, 2, or 3 and the Mississippi River level adjacent to the plant reaches 639.2' and is predicted to exceed 640', commence reactor shutdown and be in CONDITION 4 prior to the river level exceeding 640'.

4.2 OPERATING LIMITS

4.2.1 Reactor Building

4.2.1.1 CONTAINMENT INTEGRITY shall be maintained in Conditions 1, 2, 3 and during:

- a. CORE ALTERATIONS,
- b. handling of irradiated fuel, or
- c. there is fuel in the reactor and any control rod is withdrawn.

4.2.1.2 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.

REACTOR COOLANT SYSTEM

STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

4.2.2.23 The structural integrity of the ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 5.2.2.23.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5

ACTION:

- a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.
- b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 212°F.
- c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.
- d. The provisions of Specification 3.0.4 are not applicable.
- e. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

5.2.2.23 The structural integrity of ASME Code Class 1, 2 and 3 components utilized in connection with fuel storage shall be demonstrated per the requirements of Specification 3.0.10.

The applicable surveillance requirements are set forth in Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, Summer 1975 Addenda, except where specific written relief has been granted from the Nuclear Regulatory Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

BASES

4/5.2.2.23 STRUCTURAL INTEGRITY

The inspection and testing program conforms to the requirements of 10 CFR 50, Section 50.55a(g). To the extent practical, the inspection and testing of components utilized in connection with fuel storage classified into Class 1, 2, and 3 will conform to the requirements for ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code, "Inservice Inspection of Nuclear Reactor Coolant Systems," 1974 Edition, Summer 1975 Addenda.

Using Regulatory Guide 1.26, Revision 3, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants" as a guide, LACDWR components have been classified in "Classes." This classification serves as the basis for determining which ASME Code Class inspection and testing requirements are applicable to a given component. 10 CFR 50, Section 50.55a(g) requires components which are part of the reactor coolant pressure boundary and their supports to meet the inservice inspection and testing requirements applicable to components classified as ASME Code Class 1. Other safety-related components must meet the inservice inspection and testing requirements applicable to components classified as ASME Code Class 2 or 3.

The inservice inspection and testing program must be updated at 120-month intervals in accordance with 10 CFR 50, Section 50.55a(b)(2). A description of the updated programs should be submitted to the NRC for review at least 3 months before the start of each interval.

The inservice inspection and testing program must, to the extent practical, comply with the requirements in editions and addenda to the ASME Code that are "in effect" more than twelve months before the start of the period covered by the updated program. The term "in effect" means both having been published by the ASME, and having been referenced in paragraph (b) of 10 CFR 50, Section 50.55a. If a code required inspection or test is impractical, requests for deviations are submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i). Deviation requests should, if possible, be submitted to the NRC for review at least 90 days before the start of each period, however, deviations identified during an inspection period may be requested.

4.2.4 Reactor Core

4.2.4.1 The reactor shall not be operated;