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Docket No. 50-409
LS05-80-058

Mr. Frank Linder
General Manager
Dairyland Power Cooperative
2615 East Avenue South
La Crosse, Wisconsin 54601

Dear Mr. Linder:

RE: SEP TOPIC III-8.C IRRADIATION DAMAGE, USE OF SENSITIZED STAINLESS STEEL AND FATIGUE RESISTANCE

Enclosed is a copy of our evaluation of Systematic Evaluation Program Topic III-8.C Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance This assessment compares your facility, as described in Docket No. 50-409, with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment within 60 days of receipt of this letter.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic is modified before the integrated assessment is completed.

Sincerely,

/S/

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

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Enclosure:
Completed SEP
Topic III-8.C

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DATE	12/1/80	12/15/80				

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

December 15, 1980

Docket No. 50-409
LS05-80-058

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General Manager
Dairyland Power Cooperative
2615 East Avenue South
La Crosse, Wisconsin 54601

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Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Completed SEP
Topic III-8.C

cc w/enclosure:
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SYSTEMATIC EVALUATION PROGRAM

LACROSSE BOILING WATER REACTOR

Topic III-8.C Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance

The safety objective of this review is to determine whether the integrity of the internal structures of operating reactors has been degraded through the use of sensitized stainless steel.

The effect of neutron irradiation and fatigue resistance on materials of the internal structures was eliminated from the safety objective of Topic III-8.C. in memorandum to D. G. Eisenhut from D. K. Davis and V. S. Noonan dated December 8, 1979. The memorandum concluded that operating experience indicated that no significant degradation of the materials of the reactor internal structures had occurred as a result of either irradiation or fatigue. Furthermore, the Standard Review Plan (Section 4.5.2) does not address neutron irradiation nor fatigue resistance of the materials of the reactor internal structures.

As a result of incidents of intergranular stress corrosion cracking in piping in the BWR system, special study groups were formed by NRC and industry to evaluate the cause, extent and safety implications of the use of sensitized stainless steel in the nuclear steam supply systems. The study groups identified the incidents with the recirculation system bypass lines, the core spray lines, and the control rod drive return lines. It was concluded that the problem was caused by a combination of high total stresses, sensitization of the austenitic stainless steel in the heat affected zones of welds, and the relatively high oxygen content of the coolant.

The NRC study group recommended an augmented inservice inspection program for stainless steel piping, more stringent monitoring of the leak detection system, modification of plant operating practice, and the use of alternate materials immune to intergranular stress corrosion cracking. The study group concluded in NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," that intergranular stress corrosion cracking in piping would be detected prior to unstable crack growth because of the adequacy of the inservice inspection program and the leak detection system. Reactor operating experience has validated the leak-before-break concept of piping integrity, and, it was concluded, that through-wall cracks in the piping systems would be detected before they presented a hazard to the health and safety of the public.

The regulatory position on the use of sensitized stainless steel in reactor internal materials is addressed in the Standard Reveiw Plan Section 4.5.2, "Reactor Internal Materials." The areas currently reviewed in the applicant's SAR are materials specification and the controls imposed on the reactor coolant chemistry, fabrication practices and examination and protection procedures. The materials specification should comply with Section III of the ASME Boiler and Pressure Vessel Code and the components should satisfy the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal" and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel".

The reactor vessel for the LaCrosse Boiling Water Reactor was designed, fabricated and tested in accordance with the requirements of Section VIII of the ASME Boiler and Pressure Vessel Code, 1962 Edition, and Nuclear Code Case 1270N. Stresses were calculated by the methods described in the U. S. Department of Commerce Bulletin PB-151987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Associated Components". The primary stresses do not exceed the values of Table UCS-23 and UHA-23 of Section VIII.

The reactor internal structure is described in the Technical Section of the Application for Operating License, Hazards and Safety Analyses, and other Special Reports for the LaCrosse Boiling Water Reactor. The internal components were designed to provide support for the fuel and maintain the required configuration and clearances during normal and accident conditions. In addition, the internal components provide passageways for the coolant to cool the fuel and means for adequately separating the steam from the coolant water. The primary criteria for material selection for the reactor internal components were the mechanical properties, the material stability and corrosion resistance in the reactor environment. The materials used for fabricating the reactor internal components were identified as Types 304 and 348 stainless steel, Inconel 600, and minor quantities of special purpose alloys, such as Stellite.

Experience has shown that at least three elements in combination are necessary to cause cracking in sensitized stainless steel components. These are material susceptibility, an oxygenated water environment, and a threshold total stress. We assume for this evaluation that the LaCrosse Boiling Water Reactor internal components contain sensitized stainless steel in contact with an oxygen saturated coolant water environment. However, the calculated stresses on the reactor components do not exceed the threshold stress values generally associated with intergranular stress corrosion cracking. The threshold stress values are near or greater than the 0.2% off-set yield stress at temperature. Further, in the reactor environment, stress relaxation may occur due to irradiation and temperature effects.

The Monthly Operating Reports, Licensee Event Reports and the <u>BWR Operating Experience</u> were reviewed for the LaCrosse Boiling Water Reactor in order to correlate reactor internal material failure and the use of sensitized stainless steel in internal reactor components. The events related to the failure of internal components are summarized as follows:

> Failure of fittings and parts of the Core Spray Bundle and Fuel Leakage Detection System was traced to the use of cold worked stainless steel fittings and parts in the system. The mode of failure was intergranular stress corrosion cracking. The cold worked material was replaced with fully annealed parts and fittings. Similar events were reported in December 1971 and April 1973.

Roller Nut Guide failure in the Control Rod Drive Mechanism was reported in March 1969 and February 1976. Fracture of the cast Stellite material occurred at the end of the scram stroke, where the higher mechanical stress is involved. Since scram capability is not impaired by this type failure, the failures did not contribute adversely to reactor safety.

Degradation of a neutron source in November 1971 and several fuel rods in June 1973 and 1977, and May 1979 was attributed to intergranular stress corrosion cracking. Increased pressure in the neutron source from neutron reactions with the beryllium and the fuel pellet/clad interaction in the fuel rods resulted in failure in the respective events. The defective elements were removed and replaced.

During remote inspection of the reactor internals in June 1977, defects were observed in the peripheral shroud. Subsequent investigation showed that the defects were caused by fatigue initiated in weld defects. The shroud was redesigned and replaced with heavier gauge sheet material.

We conclude from our review of the reported events that the integrity of the reactor internals for the LaCrosse Boiling Water Reactor was degraded through the use of stainless steel which was subject to intergranular stress corrosion. The failures were detected by the inservice inspection and testing procedures. The failures did not adversely affect reactor safety, and were corrected either by component removal and replacement or with material immune to intergranular stress corrosion.

The inservice inspection program for the reactor internal components is being conducted during the current interval to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, including Summer 1975 Addenda. The program is in compliance with paragraph (g) of Section 50.55a of 10 CFR Part 50. It will assure that the integrity of the components is maintained during reactor operation.

We conclude from our review of the information submitted by the licensee that the materials in the reactor internal components are sensitized and will be operated in an oxygen saturated water environment. The incidents of stress corrosion cracking are rare because the total stress level is relatively low, not exceeding the 0.2% offset yield strength at operating temperature. In the unlikely event that intergranular stress corrosion cracking should occur, cracks in the components will be detected by the inservice inspection and testing procedures prior to component failure. We conclude that the integrity of the reactor internal components will be assured by the inservice inspection program conducted to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, including Summer 1975 Addenda, in compliance with paragraph (g) of Section 50.55a of 10 CFR Part 50. Further, we conclude that intergranular stress corrosion cracking in the reactor internal components is not a hazard to the health and safety of the public.