DAIRYLAND POWER COOPERATIVE • P.O. BOX 817 • 2615 EAST AV SOUTH • LA CROSSE, WISCONSIN 54601
(608) 788-4000

November 12, 1980

In reply, please
refer to LAC-7213

DOCKET NO. 50-409

Mr. James G. Keppler
Regional Director
U. S. Nuclear Regulatory Commission
Directorate of Regulatory Operations
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
REPORTABLE OCCURRENCE NO. 80-11

References: (1) LACBWR Technical Specifications, Section 3.9.2.a.(2)
(2) LACBWR Technical Specifications, Section 4.2.2.3

Dear Mr. Keppler:

In accordance with Reference (1), this is to inform you of operation during which a limiting condition was exceeded.

LACBWR Technical Specifications (Reference 2) limits the reactor vessel cooldown rate to 150°F/hr. during shutdown operations of the reactor.

On November 10, 1980, during the plant cooldown at the commencement of the 1980 refueling outage, the cooldown rate at 5 points on the reactor vessel exceeded 150°F/hr. The maximum cooldown rate was approximately 367°F/hr., during a 7-minute period, where the initial temperature was 357°F and the later temperature was 315°F. The rate was measured at a thermocouple located on the outside of the reactor vessel flange.

The other four excessive cooldown rates measured were 358°F/hr. over a 7-minute period, approximately 9 feet below the top surface of the flange; 332°F/hr. over a 7-minute period, approximately 5 feet below the flange; 326°F/hr. over a 3-minute period, approximately 7 feet below the flange; and 234°F/hr. over a 30-minute period, approximately 3 feet below the flange. At no point on the reactor vessel did the maximum change over a 1-hour period exceed 150°F.

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Following a similar experience in May 1970 where the reactor vessel cooldown rate exceeded Technical Specification limits, a detailed engineering analysis was performed ("Reactor Vessel Stresses, Fuel Temperature and Cladding Stress Calculations Following Main Steam Bypass Valve Malfunction" - United Nuclear Corporation report SS-588 dated June 8, 1970). The cooldown rate analyzed in this report was greater than the cooldown rate recently experienced. The report concluded that the vessel stresses during the 1970 incident were well within the allowable stress intensities of Section III, ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

Prompt notification of this event was made to the U. S. Nuclear Regulatory Commission Operations Center at 1715 on November 10, 1980.

If there are any questions concerning this report, please contact us.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder, General Manager

FL:LSG:af

cc: Director, Office of Management Information
and Program Control
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC Resident Inspectors

NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 LICENSEE CODE (2) LICENSE NUMBER (3) LICENSE TYPE (4) CAT (5)

CON'T
01 REPORT SOURCE (6) 0 5 0 0 0 4 0 9 DOCKET NUMBER (7) 1 1 1 0 8 0 EVENT DATE (8) 1 1 1 2 8 0 REPORT DATE (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02
03 SEE Letter
04
05
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09 SYSTEM CODE (11) CAUSE CODE (12) CAUSE SUBCODE (13) COMPONENT CODE (14) COMP. SUBCODE (15) VALVE SUBCODE (16)

(17) LER/RO REPORT NUMBER: EVENT YEAR 8 0 — SEQUENTIAL REPORT NO. 0 1 1 / OCCURRENCE CODE REPORT TYPE — REVISION NO.

ACTION TAKEN (18) FUTURE ACTION (19) EFFECT ON PLANT (20) SHUTDOWN METHOD (21) HOURS (22) ATTACHMENT SUBMITTED (23) NPRD-4 FORM SUB. (24) PRIME COMP. SUPPLIER (25) COMPONENT MANUFACTURER (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10
11
12
13
14

15 FACILITY STATUS (28) % POWER (29) OTHER STATUS (30) METHOD OF DISCOVERY (31) DISCOVERY DESCRIPTION (32)

16 ACTIVITY RELEASED (33) CONTENT OF RELEASE (34) AMOUNT OF ACTIVITY (35) LOCATION OF RELEASE (36)

PERSONNEL EXPOSURES
17 NUMBER (37) TYPE (38) DESCRIPTION (39)

PERSONNEL INJURIES
18 NUMBER (40) DESCRIPTION (41)

LOSS OF OR DAMAGE TO FACILITY (43)
19 TYPE (42) DESCRIPTION

PUBLICITY
20 ISSUED (44) DESCRIPTION (45)

NRC USE ONLY

NAME OF PREPARER ________ PHONE: ________

GPO 917-926