DAIRYLAND POWER COOPERATIVE

La Crosse, Wisconsin

54601

November 27, 1979

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In reply, please
refer to LAC-6660

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. 79-17

Reference: (1) LACBWR Technical Specifications, Section 3.9.2.a(2)
(2) LACBWR Technical Specifications, Section 4.2.2.3
(3) DPC Letter, LAC-6643, Linder to Keppler, dated November 13, 1979.

Dear Mr. Keppler:

In accordance with the provisions of Reference (1), this submittal constitutes the required follow-up report describing the occurrence which was initially reported in Reference (3). The initial report discusses the cooldown rate of the reactor vessel following a scram on November 9, 1979.

A complete explanation of the circumstances surrounding this event is as follows:

At approximately 0252 on November 9, 1979, with the reactor in Operating Condition I, at 85% Rated Thermal Power and steam flow at 500×10^3 lb/hr., the turbine governor initial pressure regulator system experienced an unexpected reduction in freedom of movement.

The Shift Supervisor took corrective action on the turbine governing system in accordance with written instructions. Following this action, the turbine governor valves, which are the main steam inlet valves, closed unexpectedly from 76% open to 60% open and then re-opened to 76% almost instantaneously. As a result of this rapid valve movement, reactor pressure quickly increased from 1248 psi to 1276 psi, causing neutron flux spikes to occur on wide range channels,

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Nos. 5 and 6, and power range channels, Nos. 7 and 8, with 124% on Channel 7 and 108% on Channel 5, representing the highest values observed. The remaining channels indicated values of 87% and 89%. The flux spike on Channel No. 7 caused a scram due to "Power Recirculation Flow Abnormal" at about 0255. The operators commenced implementation of the full scram procedure of the LACBWR Operating Manual.

Reactor pressure decreased from 1276 psig to 1040 psig in less than one minute after the scram causing the Main Steam Isolation Valve to close automatically and the shutdown condenser to operate. Shutdown condenser operation contributed to reactor pressure decreasing ultimately to 280 psig.

While the shutdown condenser was in operation and the Main Steam Isolation Valve was closed, the output of the operating reactor feed pump was manually reduced, however, the reactor water level increased to 223 in. on the wide range water level recorder before the feed and condensate pumps were tripped and the Feedwater Stop Valve (65-24-001) was closed.

Following the scram, the reactor vessel cooled down from 560°F. to 405°F. during a 22 minute period, a rate equivalent to 423°F/hr. Two factors are believed to have contributed to the increased cooldown rate of the reactor vessel. They were: 1) nearly continuous operation of the shutdown condenser for approximately ten minutes, and 2) increased water level in the reactor vessel due to delay in tripping the reactor feedwater pump. LACBWR Technical Specifications (Reference 2) limits the reactor vessel cooldown rate to 150°F/hr. during shutdown operations of the reactor. The reactor vessel stresses resulting from this cooldown have been evaluated in the attached report, which concluded that the stresses were well within the allowable ASME Code requirements. An independent study has also been performed by Nuclear Energy Services resulting in similar conclusions (a draft copy of this report is attached).

The turbine governor hydraulic control system was inspected and flushed. The turbine manufacturer has been consulted for assistance in the conduct of an investigation concerning the behavior of the turbine governor hydraulic control system, which occasionally has experienced reduction in freedom of movement. The system will be disassembled, thoroughly examined and overhauled during the spring 1980 refueling outage.

A thorough investigation and review of the incident was conducted by the LACBWR operating staff. The need for an additional alarm to show more directly the operational status of the shutdown condenser is being studied. At present, only Alarm C2-4 (Shutdown Condenser Outlet Valve Air Pressure) provides indication of shutdown condenser operation.

Also, a thorough review of the incident has been conducted with all members of the Operations Department with special emphasis being given to the recommended actions to be taken during turbine governor control problems, particularly after a resultant scram. It has been stressed that operators should take sufficient time to evaluate overall plant conditions prior to initiating corrective actions in accordance with the Operating Manual. An instructive memo has been issued to all members of the operating department providing guidance for actions following unexpected or abnormal events.

A Licensee Event Report (Reference: Appendix A, Regulatory Guide 1.16, Revision 4) is enclosed.

Authorization for this report to be submitted beyond the fourteen day reporting period was granted to L. Goodman by Mr. Ken Ridgway on Wednesday, November 21, 1979.

Should you have any questions regarding this report, please contact us.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder, General Manager

FL:LG:af

Attachments

cc: Director, Office of Inspection and Enforcement (40)
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Director, Office of Management Information and (3)
Program Control
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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REACTOR VESSEL STRESS EVALUATION DURING COOLDOWN TRANSIENT OF NOVEMBER 7, 1979 INCIDENT REPORT NO. 79-69 (RO # 79-17)

INTRODUCTION

A reactor vessel cooldown occurred as a result of conditions reported in the incident referenced above. During a cooldown transient, it is necessary to determine approximate reactor vessel and coolant temperature rate of change versus time to ensure that vessel stresses produced by the temperature-pressure transient remain within acceptable limits.

COOLANT AND VESSEL COOLDOWN RATES

Coolant and vessel cooldown rates for the November 7, 1979, incident are shown in Table I. The values listed were taken from Control Room chart records of reactor pressure and reactor vessel temperatures. Included for comparison purposes are data from similar type transients of May 1970 and August 1974.

It can be seen that the maximum coolant saturation cooldown rate for the November transient was 94°F in 4 minutes (1410°F/hr) and the vessel metal cooldown rate was 35°F in approximately 4 minutes (525°F/hr). These cooldown rates are greater than those reported in August 1974[(1)(2)] but less than the saturation and metal cooldown rates of 1620°F/hr and 825°F per hour respectively, reported in May 1970[(3)]. In the 1970 incident, it was determined that vessel stresses were less than one-half the allowable stresses in the most highly stressed regions (head shell to flange) of the pressure vessel. Therefore, since the cooldown rates of the present transient were less than the rates previously analyzed, the pressure vessel stresses are also less than those previously analyzed.

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(1) Reactor Vessel Stresses During Cooldown - Incident of 8/28/74 memo, C. Angle, August 28, 1974.

(2) Stress Analysis of LACBWR Reactor Vessel and Recirculation Piping System for Thermal Transients Produced by the Unplanned Reactor Scram of August 28, 1974, Nuclear Energy Services - NES 81A0014, September 13, 1974.

(3) Reactor Vessel Stresses, Fuel Temperature and Cladding Stress Calculation Following Main Steam Bypass Valve Malfunction, UNC, SS-588, June 8, 1970.

In addition, the forced circulation loop temperature change was 185°F in 16 minutes during the November 1979 transient. This coolant ΔT is also less than the value of ΔT used to analyze limiting circulation loop pipe and reactor inlet nozzle stresses in the August 1974 report[2].

In that report, a 250°F step temperature change was assumed to occur. Stresses were conservatively analyzed for the reactor recirculation loop nozzles and piping system.

The stress levels using a 250°F step change were found to be acceptable resulting in a calculated allowable number of stress cycles of 2,000 for the reactor inlet nozzles. Since the most recent transient rate change in temperatures was less than that previously analyzed, the stresses that resulted from this transient were also less.

CONCLUSION

It can be concluded that the magnitude of reactor vessel and piping stresses that occurred during this transient have been conservatively enveloped by previous calculations and that the values are within the allowable ASME code requirements for primary and secondary stresses. Furthermore, the incident will have a negligible effect on vessel and piping usage as determined by previous analyses.

CWA:abs

CC: Files: Reading
R3, T5C, R5G, R5H
ORC & SRC

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C. Angle
11/13/79
Rev. 1

TABLE I

VESSEL COOLDOWN TRANSIENT

VARIABLES			Nov. 1979, Inc. Rpt.	
Reactor Pressure	May 1970 (SS-588)	Aug. 1974 (P5101-50)	Avg. Rate	Max. Rate
P_{Sat}-1, psig	1280	1235	1265	1265
P_{Sat}-2, psig	200	585	280	560
ΔP_s, psi	1080 psi	650 psi	985 psi	705 psi
Δt, min.	~ 7 Min.	14 Min.	22 Min.	4 Min.
Reactor Coolant Saturation Temperatures				
T_{Sat}-1, °F	575	571	575	[illegible]
T_{Sat}-2, °F	382	[illegible]	411	[illegible]
ΔT_s, °F	193 °F	95 °F	164 °F	94 °F
Δt_s, Min.	~ 7 Min.	14 Min.	20 Min.	4 Min.
ΔT_s/hr	1620 °F/hr	365 °F/hr	494 °F/hr	1410 °F/hr
Reactor Vessel Temperatures				
Vessel T-1, °F	575 (Est)	575	565	565
Vessel T-2, °F	410 (Est)	515	405	530
ΔT_v, °F	165 °F	60 °F	160 °F	35 °F
Δt_v, Min.	12 Min.	15 Min.	22 Min.	4 Min.
ΔT_v/hr	825 °F/hr	240 °F/hr	436 °F/hr (Max)	525 °F/hr
Forced Recirculation Loop Coolant Temperatures (Measured at FCP Suction)				
T-1		565	555	555
T-2		315	370	465
ΔT, °F		250°F(Assumed)	185°F	90
Δt, min.		0	16 Min.	4.6
ΔT/hr		-	694°F/hr	1174°F/hr

EVALUATION OF STRESS IN LACBWR REACTOR

VESSEL AND RECIRCULATION PIPING SYSTEM

FOR THERMAL TRANSIENTS PRODUCED BY

NOVEMBER 7, 1979 INCIDENT

A reactor vessel cooldown transient occurred at LACBWR on November 7, 1979. The coolant and vessel cooldown rates for the November 7, 1979, incident are summarized and compared with the transients that occurred on May 15, 1970, and August 1974 in Table 1. The evaluation of stresses in the LACBWR reactor vessel and recirculation piping system for the thermal transient resulting from November 7, 1979 incident are described below.

From Table 1, it can be seen that the maximum coolant saturation cooldown rate (94°F in 4 minutes or 1410°F/hr.) for the November 7, 1979, incident is less than that of May 1970 transient ([illegible]°F in 7 minutes or [illegible]°F/hr.) and greater than that of August 1974 transient ([illegible]°F in [illegible] minutes or [illegible]°F/hr.). Similarly, the reactor vessel metal cooldown rate (35°F in 4 minutes or 525°F/hr.) for the November 7, 1979 incident is less than that of May 1970 transient (165°F in 12 minutes or 825°F/hr.) and greater than that for the August 1974 transient (60°F in 15 minutes or 240°F/hr.). The thermocouple located at the forced recirculation pump suction indicated maximum cooldown rate of 90°F in 4.6 minutes or 1174°F/hr. and average rate of cooldown of 185°F in [illegible] minutes. The maximum cooldown rate (1174°F/hr.) and the average cooldown rate (185°F) are less severe than the step temperature change of 250°F which was conservatively assumed for the stress analysis of August 1974 transient.

Stress analysis for the May 1970 incident was performed by United Nuclear Corporation (Reference 1). The result of the stress analysis is summarized in Table 2, from which it can be seen that the maximum stress in the critical region of the reactor vessel (Junction of Closure Head Spherical Shell to Flange Region) are considerably lower ([illegible] psi.) than the ASME Code allowable stress value (80,100 psi.). Since the cooldown rates (Table 1) of the November 7, 1979 incident are lower than that of the May 1970 transient, the stresses in the pressure vessel will be smaller than those shown in Table 2. Therefore, it can be concluded that the stresses in the reactor vessel resulting from November 7 transient are well within the ASME code allowable values.

Stress analysis for the August 28, 1974 incident was performed by Nuclear Energy Services (NES), Inc. (Reference 2). In the stress analysis, NES conservatively assumed a step temperature change of 100°F in the reactor outlet nozzle and a step temperature of 250°F at the reactor inlet nozzle, recirculation piping header elbow and pump suction piping/casing transition. The results of the stress analysis for August 28, 1974 incident are summarized in Table 3. From Table 3, it can be seen that primary and secondary stresses in the critical regions satisfy ASME Code Section III requirements. The fatigue analysis shows that the recirculation piping and vessel could tolerate conservatively assumed transients (250°F step change in temperature) about

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2,000 times. For the November 7, 1979 incident, since the ΔT (185°F in 16 minutes) is less than the step temperature change of 250°F assumed in the August 28, 1974 transient, the primary and secondary stresses will also be less than those shown in Table 3, and the allowable number of stress cycles will be greater than 2000.

Conclusions

The results of the stress evaluation show that the thermal transients experienced during the incident on November 7, 1979 did not result in pressure vessel and recirculation piping stresses exceeding ASME code requirements for the primary and secondary stresses. Furthermore, the incident has negligible effect on vessel or piping usage since the maximum usage factor was greater than 0.0005 (1/2000 Cycles).

With respect to the vessel lamination, five in-service inspections of the lamination have been performed and no change has been detected. These inspections span the Main Steam Bypass Valve incident (May 15, 1970) which produced a cooldown transient very comparable to the transients analyzed in this study. This fact coupled with the low usage factors associated with these incidents leads to the conclusion that plant integrity has not been degraded and that examination of the lamination can be deferred until the next regularly scheduled inspection.

VARIABLES

Reactor Pressure	May 1970 (SS-588)	Aug. 1974 (P5101-50)	Nov. 1979, Inc. Rpt. Avg. Rate	Max Rate
P_{Sat}-1, psig	1280	1235	1265	1265
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ΔP_s, psi	1080 psi	650 psi	985 psi	705 psi
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Reactor Coolant Saturation Temperatures

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ΔT_s, °F	193 °F	85 °F	164 °F	94 °F
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Reactor Vessel Temperatures

Vessel T-1, °F	575 (Est)	575	565	565
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ΔT_v, °F	165 °F	60 °F	160 °F	35 °F
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ΔT_v/hr	825 °F/hr	240 °F/hr	436 °F/hr (Max)	525 °F/hr

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T-1		565	555	555
T-2		315	370	465
ΔT, °F		250 °F (Assumed)	185 °F	90
Δt, min.		0	16 min.	4.6
ΔT/hr		∞	694 °F/hr	1174 °F/hr

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TABLE 2 (REFERENCE 1)

SUMMARY OF STRESS ANALYSIS (May 15, 1970 Incident)

LOCATION	MAXIMUM STRESS INTENSITY	ALLOWABLE
1. Junction of Closure Head Spherical Shell to Flange Region	30,782 psi	$3\ S_m = 80,100$ psi
2. Area in Vicinity of the Vessel Lamination	5,606 psi 650 psi (radial)	No Code Requirement

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SUMMARY OF STRESS ANALYSIS RESULTS (August 28, 1974 Incident)

Region	Primary Stress Intensity (Equation 9 of ASME Code) (psi)	Primary Plus Secondary Stress Intensity (Equation No. 10 of ASME Code) Sn (psi)	Peak Stress Intensity (Equation 11 of ASME Code) Sp (psi)	Simplified Elastic Plastic Discontinuity Analysis. (Equation 12 of ASME Code) (psi)	(Equation 13 of ASME Code) (psi)	Alternating stress intensity S_{alt} = ½Sp or, Equation 14 of ASME Code (psi)	Allowable number of stress cycles (N)
Reactor Outlet Nozzle	5,269 (40,050)*	36,264 (80,100)	60,000			30,000	20,000
Reactor Inlet Nozzle	5,014 (40,050)	72,596 (80,100)	123,305			61,653	2,000
Recirculation Piping Header Elbow	9,832 (27,150)	63,642 (54,300)	73,522	18,517 (54,300)	14,496 (54,300)	49,410	4,500
Pump Suction Piping/Casing Transition	5,583 (25,425)	57,044 (50,850)	95,403	291 (50,850)	26,615 (50,850)	59,322	2,500

1* Number in Parenthesis indicates ASME Code allowable stress values.

Note 1: According to Section NB-3653.6 of ASME Code, a component satisfies the design requirements of the Code if it satisfies equations 9, 10 and 11 or 9, 12, 13 and 14.