Commonwealth Edison
Dresden Nuclear Power Station
R.R. #1
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May 8, 1991

EDE LTR #91-280

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

Licensee Event Report #91-008, Docket #050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(i)(B).

E. D. Eenigenburg
Station Manager
Dresden Nuclear Power Station

EDE/ade

Enclosure

cc: A. Bert Davis, Regional Administrator, Region III
File/NRC
File/Numerical

(ZDVR/205)

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	1 of 0 4

Title (4) Violation of Core Thermal Power Limit Due to 2A Reactor Feed Pump Seal Failure

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
									N/A	
0 4	1 5	9 1	9 1	0 0 8	0 0	0 5	0 8	9 1	N/A	

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 9

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
Bill Keller, Technical Staff System Engineer Ext. 2546	8 1 5	9 4 2 - 2 9 2 0

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	S K	S E A L	B 5 8 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 1043 hours on April 15, 1991, with Unit 2 at 99% power, an unplanned power increase above the Unit 2 License Condition 2.G limit occurred as the result of a minor reactor water level excursion which occurred while securing a reactor feed pump due to failure of an inboard seal. An approximate eight inch increase in reactor water level occurred, resulting in a core thermal power increase to a peak of 104%. This event was of minimal safety significance because this type of transient had been previously demonstrated by analysis to have no adverse affect on the core. Corrective actions will include evaluations of reactor feed pump seal maintenance activities, potential reactor feed pump minimum flow line capacity enhancements, and potential operating practice changes. A previous event involving violation of the core thermal power limit License Condition was reported by LER 91-007/050237.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power

Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX)

EVENT IDENTIFICATION:

Violation of Core Thermal Power Limit Due to 2A Reactor Feed Pump Seal [SJ] Failure.

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: April 15, 1991 Event Time: 1043 Hours

Reactor Mode: N Mode Name: Run Power Level: 99%

Reactor Coolant System (RCS) Pressure: 1004.1 psig

B. DESCRIPTION OF EVENT:

With Unit 2 in steady state operation at 99% power, the reactor recirculation [AD] system master controller was disabled at 1020 hours on April 15, 1991 by locking out the 2A and 2B recirculation pump motor-generator (MG) set scoop tubes to facilitate repair of the MG set speed control error network by the Instrument Maintenance Department (IMD). At 1040 hours, an Operations Shift Supervisor was notified by a Maintenance Supervisor that the 2A reactor feed pump (RFP) inboard seal had failed, and was spraying significant amounts of water in the RFP room. The Operations Shift Supervisor immediately reported to the area and notified the Nuclear Station Operator (NSO), requesting that the 2A RFP be secured in order to prevent possible damage to the 2B RFP motor. Due to the reactor recirculation master controller being disabled, control of recirculation pump speed from the Control Room was not possible. Upon consultation with the Station Control Room Engineer (SCRE), the NSO started the standby 2C RFP and secured the 2A RFP. This prompt Operator action was deemed prudent to prevent a low reactor water level scram due to loss of two operating RFPs. During this evolution, Average Power Range Monitor (APRM) [IG] indication exceeded 102% power for 20 seconds, peaking at 105%. Heat balance calculations concluded that core thermal power peaked at 104%, which exceeded the Unit 2 License Condition 2.G. steady state thermal power limit of 2527 MWt. Power quickly stabilized as the reactor level returned to normal. No safety system actuations were required. The MG set scoop tubes were returned to normal at 1155 hours, returning the reactor recirculation system master controller to normal operation.

C. APPARENT CAUSE OF EVENT:

This event is being reported in accordance with Unit 2 License (DPR-19) Condition 2.G, which requires the reporting of any event contained in Section 2.C of the License, except as otherwise provided in the Technical Specifications. Section 2.C.(1) of the License prohibits exceeding the licensed steady state thermal power limit of 2527 MWt. Commonwealth Edison policy concerning licensed full power operations is contained within Nuclear Operations Directive (NOD) NOD-OP.15, which states that minor power excursions to 100.5% power are acceptable provided that the average power level for any eight hour period does not exceed 100%; furthrmore, any excursion above 102% of the licensed power limit or violation of the eight hour average 100% power limit is prohibited. This policy is consistent with current NRC guidance. In this case, the License Condition 2.G reporting requirements were implemented due to core thermal power having exceeded 102%. A 24 hour phone notification was also completed, as required by License Condition 2.G.

The cause of the power increase is attributed to the change in reactor water level which was caused by the change of RFPs. As the pumps were swapped, reactor water level increased by about eight inches while the 2B feedwater regulating valve (FWRV) repositioned. This increase in level was the result of a temporary increase in feedwater flow. The inherently colder feedwater reduced the core inlet enthalpy, effectively reducing the volume of voids in the core, causing power to increase. Another contributing factor was the volume of stagnant (and cooler) water in the suction and discharge piping of the idle reactor feed pump.

The root cause of the failed RFP seal could not be specifically determined. The proximate cause of failure was attributed to cracks developing in the hardface material of the seal. These cracks, in turn, deteriorate the carbon face on which the hardface rides, which leads to seal failure. These cracks could result from overheating the seal or from feedwater pressure oscillations. The feed pumps can overheat if they are operated for long periods with only the minimum flow recirculation to the condenser; a 1987 study of the feedwater system concluded that the existing minimum flow recirculation line capacity is not optimum. This overheating causes stresses to rise in the hardface which can cause cracks. A maintenance history review indicated that this particular seal had been last replaced under Work Request (WR) 98498, on February 27, 1991. The 2A RFP was the first pump started during a recent Unit 2 startup on March 14, 1991. It was started once at 1235 hours on that date and was observed to have a slight cooling water line leak; the pump was then secured and the leak was repaired. The pump was then restarted at 1945 hours and the startup proceeded satisfactorily. The 2A RFP was the second RFP secured at 0428 hours on March 24, 1991 during a Unit 2 shutdown; it was the second pump started in a subsequent Unit 2 startup on April 6, 1991. Over the past six years, 16 seal replacements have been performed on the 2A RFP. The time period between replacements varies between one month and almost three years and has no fixed pattern between failures. The 1987 feedwater system study also resulted in implementation of operating procedure improvements which provide for operation of the RFPs at more optimum points on their performance curves.

D. SAFETY ANALYSIS OF EVENT:

Based on core conditions immediately prior to the event, it was determined that the impact of the power transient on the fuel was minimal. The most limiting fuel thermal limit was the Maximum Fraction of the Limiting Critical Power Ratio (MFLCPR), which initially had greater than 15 percent margin to the Minimum Critical Power Ratio (MCPR) operating limit specified in the Technical Specifications. All other thermal limits had at least a 30% margin to the Technical Specification limits. The maximum thermal power achieved during the event was estimated to be 104% of rated core thermal power. APRM indication peaked at 105%. A feedwater controller failure event has beeen analyzed for Dresden Unit 2 Cycle 13 operation by Advanced Nuclear Fuels (ANF). The analysis assumes a level increase up to the high water level trip setpoint, and therefore bounds this overpower event. Based on this power increase from the initial power level of 99% rated core thermal power as compared to the amount of margin to thermal limits which existed, it is clear that no thermal safety limits were violated. Using the data available, it is not possible to determine conclusively whether or not preconditioning guidelines were violated. However, any violation which may have occurred would have been minor and of very short duration. Furthermore, the lack of change in offgas activity indicates that no fuel failure occurred. For these reasons, the effect of the power increase on the fuel was determined to be negligible.

E. CORRECTIVE ACTIONS:

WR 00884 was written for the Mechanical Maintenance Department (MMD) to replace the inboard seal. These repairs were completed on April 19, 1991. The MMD with the assistance of Engineering and the Technical Staff System Engineer will review the maintenance procedure for completeness and will also evaluate potential enhancements to the RFP minimum flow line capacity (237-200-91-07401). The 2A FWRV trim and actuator are also scheduled for replacement during the upcoming D2R13 refuel outage; further adjustments to the FWRV response characteristics will also be performed at that time (237-200-91-07402). The Reactor Engineer is also evaluating potential operating practice changes concerning mitigation of this type of event (237-200-91-07403).

F. PREVIOUS OCCURENCES:

LER/Docket Numbers	Title
91-007/050237	Violation of Core Thermal Power Limit Due to Unplanned 2B Reactor Recirculation Pump Speed Increase

This event was caused by a problem with an MG set speed control error limiting network connector, which was replaced. Operating procedure improvements were also initiated.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
Byron Jackson	Carbon Seal	N/A	670389RY
Byron Jackson	Rotating Seal	N/A	153765LV

An industry wide NPRDS data base search revealed 98 instances of problems with Byron Jackson Reactor Feed Pumps. A total of 57 of these were due to damaged or leaking seals.