Dominion Energy Kewaunee, Inc.
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MAY 31 2007

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

Serial No. 07-0414
KPS/LIC/RR: RO
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSEE EVENT REPORT 2007-006-00

Dear Sirs:

Pursuant to 10 CFR 50.73, Dominion Energy Kewaunee, Inc., hereby submits the following Licensee Event Report applicable to Kewaunee Power Station.

Report No. 50-305/2007-006-00

This report has been reviewed by the Plant Operating Review Committee and will be forwarded to the Management Safety Review Committee for its review.

If you have any further questions, please contact Mr. Richard Repshas at (920) 388-8217.

Very truly yours,

Leslie N. Hartz
Site Vice President, Kewaunee Power Station

Attachment

Commitments made by this letter: NONE

IE22
NRC/NRR

cc: Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
2443 Warrenville Road
Suite 210
Lisle, IL 60532-4352

Ms. Margaret H. Chernoff
Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O-8G9A
Washington, DC 20555-0001

NRC Senior Resident Inspector
Kewaunee Power Station

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0066), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)

Reactor Coolant System Pressure-Temperature Limits Momentarily Exceeded During Solid Plant Operation

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	**24**	**2004**	**2007**	**-- 006 --**	**00**	**05**	**31**	**2007**	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR : (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard Repshas	**(920) 388-8217**

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT

On 4/5/2007, during an engineering review of past computer data in response to undesirable solid plant cooldown and pressure control situations, the following previously unreported condition was discovered.

On 11/24/2004 at 06:55 CST, with the plant shutdown under solid conditions, a reactor coolant system pressure transient occurred when a reactor coolant pump was started during system dynamic venting. Due to heat contained in the steam generator, the pump start caused a reactor coolant system heatup which resulted in a sudden increase in indicated reactor coolant system pressure from 371.9 psig to 509.8 psig and exceeded the heatup and cooldown curve pressure limit of 493 psig. The pressure transient was transferred to the connected residual heat removal system. The maximum RCS heatup and cooldown rate limits were not exceeded.

The residual heat removal pump discharge pressure increased to 626.5 psig which exceeded the system design pressure of 600 psig, and the 600 psig setpoint of residual heat removal system relief valves. The residual heat removal pump suction pressure increased to 502 psig which exceeded the setpoint of the residual heat removal suction header relief valve (set at 480 psig) and the low temperature overpressure protection relief valve (set at 500 psig). One or more of the relief valves described above lifted resulting in a loss of approximately 8 gallons of primary system inventory.

This event is being reported under § 50.73(a)(2)(i)(B) as an operation which was prohibited by the plant's Technical Specifications.

Event Description:

On 4/5/2007, during an engineering review of past computer data in response to undesirable solid plant cooldown and pressure control situations, the following previously unreported condition was discovered.

During recovery from an outage in November of 2004, Reactor Coolant System (RCS) [AB] filling and venting was in progress, with solid plant pressure control and with the Residual Heat Removal (RHR) [BP] system in the cooldown alignment.

The procedure revision that was in place at the time required running one Reactor Coolant Pump (RxCP) [P] until RCS temperature was greater than 140 degrees F, then starting the second RxCP and running both for 10 minutes. Per the procedure, both pumps were then stopped, the RCS pressure was reduced, and the system was vented. When venting was complete, system pressure was again increased and a RxCP re-started.

When the RCS is heated above 140 degrees F, the Steam Generator (SG) [SG] is also heated. When the RxCPs are secured, the RCS cools down rapidly due to cooling from the RHR system. However, the SG is not cooled down and remains near the maximum temperature that was achieved. Therefore, when the RxCP is re-started, heat transfer takes place in the SG causing a rapid RCS temperature increase, with a resultant pressure increase. Limiting the pressure increase is dependent on the skill of the operator in rapidly opening the letdown control valve [V].

For this event, both RxCPs were run until RCS temperature was 146.2 degrees F. Both pumps were then stopped, the RCS pressure was reduced, and, because of low decay heat following a refueling, RCS temperature decreased to 112.4 degrees F. There is no installed instrumentation to measure SG temperature, but it is conservatively assumed that the SGs remained at the maximum RCS temperature of 146.2 degrees F. At 0655 on 11/24/2004 the RxCP was restarted, causing a RCS heatup and sudden increase in RCS pressure from 371.9 psig to 509.8 psig, exceeding allowable pressure limits. RCS pressure was above the Technical Specification (TS) limit for 12 seconds. Since RHR was in the RCS cooldown alignment, the RCS pressure transient was transferred to the RHR system. The maximum RCS heatup and cooldown rate limits were not exceeded.

RHR pump suction pressure increased to 502 psig. This exceeded the setpoint of the RHR suction header relief valve [RV] (set at 480 psig) and the low temperature overpressure protection (LTOP) relief valve (set at 500 psig). Because the RHR pump was operating, RHR pump discharge pressure increased to 626.5 psig, which exceeded the system design pressure of 600 psig, and the 600 psig setpoint of RHR system relief valves. One or more of the above relief valves lifted momentarily resulting in a loss of approximately 8 gallons of primary system inventory.

The actual maximum pressure limit at 112.4 degrees F is 621 psig (as determined by analysis done in accordance with Appendix G of Section III of the ASME Boiler and Pressure Vessel Code). This maximum pressure has been reduced by 58 psig to account for instrument uncertainty and by 70 psig to account for RxCP differential pressure. Thus, TS Figure 3.1-2, Kewaunee Unit No. 1 Cooldown Limitation Curve, provides for an indicated pressure limit of 493 psig at this temperature (621 psig – 70 psig – 58 psig) . During the transient, the maximum pressure reached was 509.8 psig, as indicated by an instrument having a maximum total loop error of 12.2 psig. Therefore, the actual maximum pressure was 509.8 psig + 12.2 psig + 70 psig = 592.0 psig. This is below the maximum allowed pressure of 621 psig. Therefore, Reactor Vessel [RPV] integrity was not challenged.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

The maximum observed RHR system pressure of 626.5 psig was above the system design pressure of 600 psig. However, the piping design code for the RHR system allows for short variations in pressure above the design pressure, up to 20%, for short periods. 20% of the 600 psig design is 120 psig. Therefore, momentarily exceeding the RHR design pressure by 26.5 psig does not raise any RHR operability or integrity concerns.

Event Analysis:

This event is being reported under § 50.73(a)(2)(i)(B) as an operation which was prohibited by the plant's Technical Specifications (TS).

TS 3.1.b states:

3.1.b. Heatup and Cooldown Limit Curves for Normal Operation

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS 3.1-1 and TS 3.1-2. Figures TS 3.1-1 and TS 3.1-2 are applicable for the service period of up to 33(1) effective full-power years.

    A. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

    B. Figures TS 3.1-1 and TS 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

    C. The isothermal curve in Figure TS 3.1-2 defines limits to assure prevention of non-ductile failure applicable to low temperature overpressurization events only. Application of this curve is limited to evaluation of LTOP events whenever one or more of the RCS cold leg temperatures are less than or equal to the LTOP enabling temperature of 200°F.

(1) The curves are limited to 31.1 EFPY due to changes in vessel fluence associated with operation at uprated power.

The sudden increase in indicated RCS pressure from 371.9 psig to 509.8 psig exceeded the allowable pressure limits of TS Figure 3.1-2, Kewaunee Unit No. 1 Cooldown Limitation Curve. Thus, this event is being reported under § 50.73(a)(2)(i)(B) as an operation which was prohibited by the plant's Technical Specifications.

Even though the limits of TS Figure 3.1-2 were exceeded, the systems functioned as designed and prevented the reactor coolant pressure boundary from exceeding it's 10 CFR 50 Appendix G limit.

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TEXT CONTINUATION

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Safety Significance:

Since the maximum RCS pressure reached during this event was 592 psig (corrected for actual instrument accuracy and RxCP differential pressure), the 10 CFR 50 Appendix G calculated pressure limit of 621 psig was not exceeded and the Reactor Vessel integrity/RCS System Integrity was maintained operable. Since the piping design code for the RHR system allows for short variations in pressure above the design pressure, up to 20%, momentarily exceeding the RHR design pressure does not raise any RHR operability or integrity concerns.

Cause:

The apparent cause is being determined at this time.

Current investigation has identified that procedural guidance at the time of this event was inadequate. The requirement to heat-up the RCS above 140°F to run both RCPs prior to the final venting should have been deleted and the procedure changed to complete all venting prior to starting the RCS heat-up as part of Tech Spec Amendment 144, issued on April 12, 1999.

Corrective Actions:

The required procedural changes have been incorporated. These changes will prevent future occurrences. The cause of the procedure inadequacy is being investigated in the apparent cause evaluation and will be addressed in the corrective action system.

Similar Events:

None