April 1, 2005

NRC-05-037
10 CFR 50.73

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

Kewaunee Nuclear Power Plant
Docket 50-305
License No. DPR-43

Reportable Occurrence 2005-001-00

In accordance with the requirements of 10 CFR 50.73, "Licensee Event Report System," the attached Licensee Event Report (LER) for reportable occurrence 2005-001-00 is being submitted.

This letter contains no new commitments and no revisions to existing commitments.

Craig W. Lambert
Site Vice-President, Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC

Enclosure (1)

cc: Administrator, Region III, USNRC
Senior Resident Inspector, Kewaunee, USNRC
INPO Records Center

IE22

ENCLOSURE 1

LICENSEE EVENT REPORT (LER)
2005-001-00

4 pages follow

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007

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.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Reactor Thermal Power Eight-Hour Average Limit Exceeded

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
01	31	2005	2005	-- 001 --	00	04	01	2005	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	100

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)
Robert Reynnells	(920) 388-8791

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
X	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO		05	13	2005

ABSTRACT

On January 31, 2005, Kewaunee Nuclear Power Plant was in operation at 100 percent power. Instrument and Control (I&C) personnel were performing two surveillance test activities -- Surveillance Procedure (SP) 47-316A (Channel 1 (Red) Instrument Channel Test) and SP 05A-34C-1 (Feedwater Flow Transmitter Channel 1 (Red) Calibration). At approximately 1200 hours, these surveillances caused the reactor thermal power 1-minute, 15-minute, and 8-hour average indications to become inaccurate. The Control Room operators entered procedure A-CP-46 (Abnormal Plant Process Computer System) to determine appropriate actions. The Alternate Reactor Thermal Output (ARTO) system was being used to monitor reactor power. At approximately 1330 hours, the 1-minute thermal power readings returned to approximately 1772 megawatts thermal (MW^{th}), 100 percent power. Procedure A-CP-46 was exited at approximately 1345 hours, and use of ARTO was stopped. At the time the 8-hour Reactor Thermal Output (RTO) average was viewed as not valid. During shift turnover there was no direction given concerning the expected affect on the 8-hour average. At 1952 hours, the shift performed an 11 gallon dilution to compensate for normal core burn up. At both 2047 hours and at 2057 hours, the RTO 15-minute average PPCS alarm came in at 1773 MW^{th}. This computer annunciator cleared within minutes. As the effects of the earlier I&C surveillances were being dropped from the 8-hour average calculation, indicated 8-hour average power started to increase, resulting in a high power alarm at 2131 hrs. Indicated 8-hour average reactor power peaked at 1772.07 MW^{th}. A 3.4 gallon boration was performed at 2141 hours, and power reduced to less than 1772 MW^{th} at 2216 hours. The root cause evaluation of this event is in progress. The final root cause determination will be provided in a supplemental report. Corrective actions have been initiated to revise procedures, to change the alarm setpoint for the 8-hour average RTO, and add the 15-minute average RTO to the alarm response procedure. This event is not identified as a Safety System Functional Failure.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Event Description

On January 31, 2005, the Kewaunee Nuclear Power Plant (KNPP) was in operation at 100 percent rated thermal power of 1772 megawatts thermal (MW^{th}). On day shift (0600 hrs to 1800 hrs), Instrument and Control (I&C) personnel were performing two surveillance test activities. These tests were being performed in accordance with Surveillance Procedure (SP) 47-316A (Channel [CHA] 1 (Red) Instrument Channel Test) and SP 05A-34C-1 (Feedwater Flow Transmitter [FT] Channel 1 (Red) Calibration). At approximately 1200 hours, these surveillance activities caused the indications for reactor thermal power 1-minute, 15-minute, and 8-hour averages to become inaccurate.

The Control Room operators entered abnormal operating procedure A-CP-46 (Abnormal Plant Process Computer System) to determine appropriate actions. The Alternate Reactor Thermal Output (ARTO) system was used to monitor reactor power. At approximately 1330 hours, the 1-minute thermal power readings returned to approximately 1772 MW^{th}. Procedure A-CP-46 was exited at approximately 1345 hours, and the use of ARTO was stopped. At the time, the 8-hour Reactor Thermal Output (RTO) average was viewed as not valid. The 1-minute and 15-minute RTO values had returned to normal readings. Corrective Action Program document CAP 25257 was initiated by the Plant Process Computer System (PPCS) Group at 1453 hours, to document the discrepancy in expected and actual reaction of the PPCS [CPU] to I&C surveillance activities. The day shift operating crew initiated CAP 25258 to document the RTO being questionable as a result of performing procedure SP 05A-34C-1.

Turnover from day shift to night shift occurred at 1800 hours. Control Room Supervisor turnover items included review of questionable values resulting from I&C surveillance activities, the 8-hour average RTO indicating artificially low, and initiation of CAP 25258 that documented problems that occurred during SP 05A-34C-1 & questionable PPCS data. Reactor Operator turnover included discussion of previous dilutions and the thermal power PPCS trends, including the present 15-minute RTO being greater than 1772 MW^{th}. Shift Manager turnover did not discuss actions to ensure the 8-hour average RTO remained below the license limit (for example keeping the 15-minute average RTO less than 1772 MW^{th}). The night shift was aware that the 8-hour average RTO was reading low due to I&C activities on day shift. However, there was no direction given concerning the expected affect on the 8-hour average power indication, when the PPCS calculated 8-hour average value was validated. During the pre-shift brief, the night shift operations crew did not question the 8-hour average RTO indication. Also, the operating crew did not verify the 8-hour average RTO indication to monitor (PPCS or ARTO).

At 1952 hours, the operating shift performed an 11 gallon dilution to compensate for normal core burn up. At both 2047 hours and 2057 hours, the RTO 15-minute average PPCS alarm [JA] came in at 1773 MW^{th}. These computer annunciators [ANN] each cleared within minutes. Shift management was not notified of either of these alarms, as they were expected due to the recent dilution, and there is no specific requirement to announce each computer alarm. As the effects of the I&C surveillance tests on day shift were being dropped from the 8-hour average calculation, the indicated 8-hour average power started to increase rapidly, resulting in the Trouble Light Annunciator (TLA)-11 "Reactor Thermal Power High" alarm (1772 MW^{th}) at 2131 hrs. Indicated 8-hour average reactor power peaked at 1772.07 MW^{th}. Alarm Response Procedure (ARP) 47033-31 for TLA-11 was entered, a 3.4 gallon boration was performed at 2141 hours, and 8-hour average power reduced to less than 1772 MW^{th} at 2216 hours.

Event Analysis and Safety Significance

This event is being reported under 10CFR50.73(a)(2)(i)(B), operation which was prohibited by the Technical Specifications.

The KNPP facility operating license states in Section 2.C.(1) – "The NMC is authorized to operate the facility at steady-state reactor core power levels not in excess of 1772 megawatts (thermal)." NRC guidance relative to "Licensed Power Level" is contained in a letter from Mr. E. L. Jordan (Assistant Director for Technical Programs, Division of Reactor Operations Inspection, Office of Inspection and Enforcement), dated August 22, 1980. This guidance specifies the following:

- The average power level over any 8-hour shift should not exceed the full steady-state licensed power level (and similarly worded terms).
- The exact 8-hour periods are up to the plant. It is permissible to briefly exceed the full steady-state licensed power level by as much as 2 percent for as long as 15 minutes.
- In no case should 102 percent power be exceeded.
- Lesser power excursions for longer periods should be allowed (i.e. 1 percent excess for 30 minutes, 0.5 percent excess for 1 hour, etc). The 8-hour average power limit will prevent excessive multiple excursions.

The 8-hour average reactor thermal power maximum value was 1772.07 MW^{th} for this event. This exceeded the 1772 MW^{th} limit by only 0.07 MW^{th}. The thermal power limitations given above relative to the 15-minute, 30-minute, and 1-hour time periods were not exceeded. The maximum 15-minute reactor thermal power value during this event was 1773 MW^{th}.

Technical Specification 2.1, "Safety-Limits – Reactor Core", requires that the combination of rated power level, coolant pressure, and coolant temperature shall not exceed the limits specified in the Core Operating Limits Report (COLR). The Reactor Core Safety Limits Curve was not exceeded. This event is therefore determined to have a very low safety significance. This event is identified as not being a Safety System Functional Failure.

Cause

The root cause evaluation of this event is in progress. The final root cause determination will be provided in a supplemental report. Any changes to the corrective actions, based on the results of the final root cause evaluation, will also be included in the supplemental report.

Corrective Action

1. Per Work Order 05-002811:
   - Add the reactor thermal power 15-minute average PPCS computer point as an input to Trouble Light Annunciator TLA-11 "Reactor Thermal Power High" alarm.
   - Reduce the TLA-11 alarm setpoint for the 8-hour average thermal power to 1771.7 MW^{th}.
2. Revise ARP 47033-31 (TLA-11) to add the 15-minute thermal power alarm setpoint of 1772 MW^{th}, change the 8-hour thermal power alarm setpoint to 1771.7 MW^{th}, and to provide additional action to use the Valve Position Limiter (VPL) to lower turbine power as recommended actions.

3. Revise procedure A-CP-46 to clarify guidance to monitor reactor power, add steps to ensure 1772 MW^{th} is not exceeded when the PPCS is out-of-service, and to clarify conditions to return to using the PPCS RTO program monitoring.
4. Revise procedure N-O-03 (Plant Operation Greater Than 35% Power) to reference the 1771.7 MW^{th} setpoint, and to add a note to ensure the nominal maximum reactor power 8-hour average is maintained below 1771 MW^{th}.
5. Revise General Nuclear Procedure (GNP) 03.17.10 (Reactivity Management) to discuss the responsibility of the operating crews to maintain the steady state power less than 1772 MW^{th}, as described in the KNPP operating license. This discussion is to define the plant philosophy on maintaining reactor power within the license limit.
6. Perform a review of other alarm setpoints that are set at Technical Specification (TS) values, and potentially would not allow sufficient time for Operations to take action before a TS required value is exceeded.
7. Revise procedures SP-05A-034C-1, 2, 3, & 4. In each SP, Step 6.2.5 identifies which channels are the controlling channels and then states to 'N/A' Steps 6.2.6 through 6.2.9. Step 6.2.6 removes the selected channel from PPCS scan, and this step should not be marked as 'N/A'.

Previous Similar Events

KNPP LER 92-18; Licensed Power Exceeded Due To Inaccurate Feedwater Flow Indication

On September 22, 1992, the plant was returning to 100 percent power after a unit trip. After applying the ultrasonic flow meters (UFMs) correction factor to in-line venturi feedwater (FW) flow measurements and escalating to 100 percent power, it was noted that the electric output was 1 to 2 megawatts higher than before the unit trip. An evaluation determined indicated FW flow, measured by the UFMs, was 0.41 percent low and reactor power was approximately 0.2 percent greater than licensed thermal power. Immediate actions were taken to decrease power to within licensed limits. The change in UFM output was caused by corrosion product build up between the UFM transducers and the FW pipe in conjunction with age related degradation of the transducers. The FW UFMs were calibrated, using the full flow bypass line venturi, to accurately measure FW flow.