Omaha Public Power District

444 South 16th Street Mall
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November 11, 2003
LIC-03-0136

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Subject: Licensee Event Report 2003-003 Revision 0 for the Fort Calhoun Station

Please find attached Licensee Event Report 2003-003, Revision 0, dated November 11, 2003. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(iv). If you should have any questions, please contact me.

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/EPM/epm

Attachment

c: B. S. Mallett, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
INPO Records Center

IE22

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information 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 Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Fort Calhoun Nuclear Station Unit Number 1	05000285	1 OF 3

4. TITLE

Reactor Trip During Plant Shutdown due to Inadequate Preparation

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
09	12	2003	2003 -	003 -	00	11	11	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	1
10. POWER LEVEL	15

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

	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)	X	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
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		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)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Erick Matzke, Licensing Engineer	402-533-2114

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

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

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

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

During a reactor shutdown in preparation for a refueling outage, the power reduction was stopped because the operators were unable to maintain the axial shape index (ASI) within the expected band. In order to minimize the operational challenge, management provided two reactor trip criteria. At 2037, with the power reduction close to a nominal 15 percent power, it was noted that ASI might not be maintained within the required margin if the shutdown continued. At 2055 on September 12, 2003, Operations determined that a reactor trip was required because one of management's reactor trip criteria was about to be met. The reactor operators were directed to trip the reactor using the manual pushbutton. A four (4) hour non-emergency report was made to the NRC Operations Center at 0010 CDT on September 13, 2003, pursuant to 10 CFR 50.72(b)(2)(iv). This report is being made pursuant to 10 CFR 50.73(a)(2)(iv).

No formally approved written guidance was provided to the operator and therefore this event is reportable. Management failed to recognize that a manual trip of the reactor without a change to the shutdown procedure would be reportable.

Appropriate procedural revisions to allow the flexibility in plant procedures to allow a manual reactor trip from power levels greater than 2 percent are being processed.

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

BACKGROUND

Fort Calhoun Station (FCS) is a two loop Combustion Engineering designed pressurized water reactor plant. Cycle 21 continued to utilize the extreme low radial leakage fuel management scheme which minimizes neutron leakage at the reactor vessel critical weld locations. The reactor core design incorporates 49 control element assemblies (CEA's). CEA's are moved in groups to satisfy the requirements of reactor shutdowns, power level changes, and operational maneuvering.

The power range nuclear instrument channels are used to calculate core power used in the Axial Shape Index (ASI). ASI is defined as the ratio of the difference in core power between the upper and lower part of the core divided by the sum of the core power signals. A positive ASI indicates more power is being produced in the lower part of the reactor core. A negative ASI indicates more power is being produced in the upper part of the reactor core. Three separate limits have been established for allowable ASI as a function of reactor power. These limits allow the axial peaks to increase as reactor power decreases.

EVENT DESCRIPTION

On September 11, 2003, FCS was at approximately 67 percent reactor power, while shutting down for the 2003 refueling outage. Control room operators had been adding boric acid solution and using control rod insertion to achieve approximately a 3 percent per hour power reduction rate. The Reactor Engineer (RE) had given the operating crew written shutdown guidance on power reduction rate, ASI control parameters, reactor coolant system temperature, chemistry, and other operational limitations during the planned shutdown.

As the plant shutdown progressed and reactor power continued to be reduced, several attempts were made to drive ASI more positive. These attempts to drive ASI more positive were met with limited success. At 38 percent power the reactor power reduction was stopped because ASI was approaching an Reactor Protection System (RPS) pre-trip set-point. With reactor power stabilized at 38 percent, the operators monitored the trend in ASI.

Plant management met at 1500 on September 12 to discuss the ASI control problems that were creating operational challenges for the crews. Management discussed possible solutions using the operational decision making model. At the conclusion of the meeting, management decided to hold power at 38 percent to allow ASI to trend more positive, such that, when the power reduction was resumed, an RPS pre-trip would not be generated. After the power hold, reactor power would then be reduced by inserting control rods and boration. However, the power reduction would be stopped if ASI could not be maintained in the allowable band to preclude an RPS pre-trip alarm. In addition, management provided the operating crew with two manual trip criteria: 1) If one of four axial power distribution pre-trips were reached, or 2) If the reactor power reduction had stopped because of ASI approaching a pre-trip set-point and the power reduction could not be restarted by 2100. This criteria was developed to reduce the challenge to the operating crew.

Reactor power was reduced from 38 percent to approximately 16 percent over the next two hours (1842 to 2037). At 2037, with the power reduction close to a nominal 15 percent power, ASI was still near the manual trip criteria. At 2055, management's reactor trip criteria was met. The Control Room Supervisor had previously briefed the crew on the manual reactor trip, and the licensed operators were directed to trip the reactor using the manual pushbutton. When the reactor was manually tripped, the operating crew entered EOP-00, "Standard Post Trip Actions." An uncomplicated manual reactor trip was diagnosed, and EOP-01, Reactor Trip Recovery, was entered at 2104. At 2200 on September 12, EOP-01 was exited and OP-3A, "Plant Shutdown," was re-entered to continue refueling outage preparations. A four (4) hour non-emergency report was made to the NRC Operations Center at 0010 CDT on September 13, 2003, pursuant to 10 CFR 50.72(b)(2)(iv). This report is being made pursuant to 10 CFR 50.73(a)(2)(iv).

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

SAFETY SIGNIFICANCE

The manual reactor trip is not significant to nuclear safety because a manual reactor trip is an analyzed action appropriately incorporated into design analyses and FCS design bases. Although the trip challenges plant equipment to respond, all structures, systems and components (SSCs) responded appropriately, and the trip was uncomplicated.

While there is a slight increase in core damage risk due to a manual reactor trip, this risk is largely associated with higher power levels (above 50 percent). The relatively low power level of the reactor prior to the trip (15 percent) contributes to the conclusion that the event was non-safety significant. No reactor safety functions were challenged during the event.

In addition, the magnitude of the ASI transient experienced during the plant down-power was within the reactor's specified design parameters and all design analyses remained valid.

CONCLUSION

Plant management had carefully considered the challenges that the operating crews were experiencing with controlling ASI during the plant shutdown. The option of tripping the reactor, as well as other options, were discussed in detail during the management meetings conducted during the shutdown. Nuclear engineering personnel were consulted to determine if any options were being overlooked. After due consideration to the safety and effectiveness of the various shutdown options to meet the plant shutdown objectives, two shutdown criteria were developed as previously mentioned. Management failed to recognize that a manual trip of the reactor without a change to the shutdown procedure would be reportable in accordance with NUREG 1022, revision 2. No formally approved written guidance was provided to the operators. Therefore, this event is reportable.

CORRECTIVE ACTIONS

Appropriate procedural revisions to allow the flexibility in plant procedures to allow a manual reactor trip from power levels greater than 2 percent is being processed. Other corrective actions to preclude similar situations will be implemented by the plants corrective action system.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event did not result in a safety system functional failure in accordance with NEI 99-02.

PREVIOUS SIMILAR EVENTS

The plant has not experienced any similar problems.