TOLEDO EDISON

EDISON PLAZA
300 MADISON AVENUE
TOLEDO, OHIO 43652-0001

AB-93-0020
NP-33-93-03

Docket Number 50-346

License Number NPF-3

June 17, 1993

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Gentlemen:

LER 93-003
Davis-Besse Nuclear Power Station, Unit Number 1
Date of Occurrence - May 20, 1993

Enclosed please find Licensee Event Report 93-003, which is being submitted to provide 30 days written notification of the subject occurrence. This LER is being submitted in accordance with 10 CFR 50.73(a)(2)(iv).

Very truly yours,

Louis F. Storz

Louis F. Storz
Plant Manager
Davis-Besse Nuclear Power Station

LFS/lkg

enclosure

cc: Mr. J. B. Martin
Regional Administrator
USNRC Region III

Mr. Stan Stasek
DB-1 NRC Senior Resident Inspector

240083

JE22

NRC FORM 366 (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Davis-Besse Unit Number 1	05000-346	1 OF 4

TITLE (4)
Reactor Trip - Loss of ICS Tave Input

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
05	20	93	93	003	00	06	17	93	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			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 (Include Area Code)
Mark A. Turkal, Engineer - Nuclear Licensing	(419) 321-7377

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

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 15 single-spaced typewritten lines) (16)

On May 20, 1993, at 1138 hours, the plant experienced a trip from approximately 102% power. Loss of continuity between a fuse and fuseholder resulted in a loss of power to an auxiliary relay in the Non-Nuclear Instrumentation (NNI) System. This caused a loss of the selected RCS average temperature (Tave) input to the Integrated Control System (ICS). The ICS Tave input failed to zero VDC indicating that the RCS average temperature was approximately 570 degrees F. Normal RCS average temperature is 582 degrees F. To compensate, the ICS Tave integral increased Reactor Demand, withdrawing control rods and increasing reactor power. In an attempt to reduce reactor power, operators took manual control of ICS and decreased the Steam Generator/Reactor Demand signal. The primary reactor operator overreacted and lowered demand too far. Primary to secondary heat flow became unbalanced as feedwater flow followed the lowered demand, but reactor power did not decrease due to the low Tave input. As feedwater flow decreased, RCS pressure increased. The reactor tripped on high RCS pressure.

The faulty fuse cap and fuse were replaced and other NNI fuse holders and fuses were inspected. The plant was returned to service on May 21, 1993 at 1308 hours.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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

Description of Occurrence:

On May 20, 1993, the plant was in Mode 1 at approximately 100% reactor power. The event began at 1135 hours when loss of continuity between a fuse and fuseholder resulted in de-energizing of auxiliary relay 86/RC7A in the Non-Nuclear Instrumentation (NNI) System. Loss of power to the auxiliary relay caused the loss of the selected Integrated Control System (ICS-JA) Reactor Coolant System (RCS-JD) average temperature (Tave) input. The ICS provides automatic coordination of the reactor, steam generator feedwater control, and turbine. The ICS includes four independent subsystems: (1) the Unit Load Demand Control, (2) the Integrated Master Control, (3) the Steam Generator Control, and (4) the Reactor Control. The four ICS subsystems were automatically controlling the plant when Tave failed.

The loss of Tave resulted in an erroneous input to the ICS indicating that the RCS average temperature was approximately 570 degrees F. Normal RCS average temperature is 582 degrees F, which was the approximate RCS average temperature at this time. As a result, the ICS attempted to recover normal RCS average temperature by withdrawing control rods, increasing reactor power to approximately 102%. The ICS includes a Reactor Demand High Power Limiter which prevents the ICS from demanding a power greater than 103% to prevent exceeding an RPS channel high flux trip setpoint.

In a proper attempt to reduce reactor power, operators took manual control of the ICS and decreased the Steam Generator/Reactor Demand signal. The primary reactor operator erred, however, as he overreacted to the conditions and manually lowered demand to approximately 80% of full power without properly informing the balance of the operating crew. Primary to secondary heat flow became unbalanced as feedwater flow decreased without a corresponding decrease in reactor power due to the low Tave input failure. As feedwater flow decreased, RCS pressure increased. The Reactor Protection System (JC) properly tripped the reactor on high RCS pressure (trip setpoint - less than or equal to 2355 psig) at 1138 hours. Maximum indicated RCS pressure was 2350 psig based on plant computer data.

Redundant sensors for major system parameters are available to the ICS. The ICS inputs are monitored by 19 Smart Analog Selector Switch (SASS) channels designed to automatically transfer to a redundant sensor in case of instrumentation problems. The SASS channel monitoring RCS average temperature did not transfer in this event because the failure was downstream of the SASS.

The plant was returned to service on May 21, 1993 at 1308 hours.

Description of Occurrence: (con't)

Initial notification of the reactor trip was made on May 20, 1993, at 1258 hours, in accordance with 10 CFR 50.72(b)(2)(ii). This LER is being submitted in accordance with 10 CFR 50.73(a)(2)(iv).

Apparent Cause of Occurrence

The plant transient was initiated by a loss of continuity between a fuse and fuseholder in the circuit that powers an auxiliary relay for Tave input selection to the ICS. The reactor trip resulted from operator action taken in an attempt to decrease reactor power.

A loss of continuity between a fuse and fuseholder resulted in de-energizing of auxiliary relay 86/RC7A in the NNI system. Loss of power to the auxiliary relay caused the loss of the selected ICS Tave input which provided an erroneous input to the ICS indicating that the RCS average temperature was approximately 570 degrees F. Normal RCS average temperature is 582 degrees F. As a result, the ICS attempted to recover normal RCS average temperature by withdrawing control rods and increasing reactor power to approximately 102%. In an attempt to reduce reactor power, operators took manual control of ICS and decreased the Steam Generator/Reactor Demand signal. Feedwater flow decreased, however, reactor power did not due to the low Tave input. As feedwater flow decreased, RCS pressure increased. The RPS functioned properly by tripping the reactor on high RCS pressure.

Analysis of Occurrence

The event reported in this LER has minimal safety significance.

Plant and operating crew response to the trip was satisfactory. The Control Rod Drive trip breakers opened and all control rods inserted on the reactor trip as designed. Steam generator pressure increased due to the closing of the main turbine stop valves. The Turbine Bypass Valves and the Atmospheric Vent Valves (AVVs) properly opened and the Main Steam Safety Valves (MSSVs) lifted in response to the increasing secondary system pressure. The MSSVs and AVVs closed as steam generator outlet pressure decreased.

Shortly before the reactor trip, there was an undetected lockup of the Safety Parameter Display System (SPDS). As a result, the SPDS was not available to the control room operators post-trip. The failure of the SPDS was not associated with the reactor trip and did not affect the operators' ability to respond to the event. The SPDS was returned to service at approximately 1150 hours and pertinent transient data is available.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

Corrective Action

The faulty fuse cap and fuse were replaced and the remaining 37 NNI fuse holders and fuses associated with SASS instrumentation were inspected. No additional faulty fuse holders or fuses were discovered, however, seven fuses and one fuse cap were replaced as a precautionary measure.

Operations personnel completed required reading regarding the failure of Tave, the plant's response, and general guidance for ICS transients.

Subsequent to the event, simulator training on the transient was initiated. Each operating crew will complete this training by June 25, 1993. Control room team training, in association with INPO, is scheduled for completion by July 30, 1993.

Failure Data

There have been no LERs due to ICS initiated transients in the previous three years. Also, there have been no LERs in the previous three years due to reactor trips resulting from operator action.

NP 33-93-003

PCAQ No. 93-0311