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10CFR50.73

Nuclear Operations

September 11, 1989
NRC-89-01

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

Reference: Fermi 2
NRC Docket No. 50-341
Facility Operating License No. NPF-43

Subject: Licensee Event Report (LER) No. 89-017-00

Please find enclosed LER No. 89-017-00, dated September 11, 1989, for a reportable event that occurred on August 10, 1989. A copy of this LER is also being sent to the Regional Administrator, USNRC Region III.

If you have any questions, please contact Joseph Pendergast at (313) 586-1682.

Sincerely,

WSOrser

Enclosure: NRC Forms 366, 366A

cc: A. B. Davis
J. R. Eckert
R. C. Knop
W. G. Rogers
J. F. Stang

Wayne County Emergency
Management Division

IE22
1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4): Revised Feedwater Line Break Analysis

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)
0 8	1 0	8 9	8 9	0 1 7	0 0	0 9	1 1	8 9	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 9 9

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (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)	X 50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	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)
Joseph Pendergast, Licensing Engineer	3 1 3 5 8 6 - 1 6 8 2

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)

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

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

The analysis of the postulated feedwater line break in the steam tunnel is described in Updated Final Safety Analysis Report (UFSAR) section 3.6.2.2.2. The UFSAR evaluation was completed in the early 1970's. Since documentation for the UFSAR conclusions cannot be located, a new analysis was conducted. The difference between the new analysis and the UFSAR is that the single failure of the fast closing feedwater pump discharge valves as described in the UFSAR is non-conservative. The single failure in the new feedwater line break scenario is a failure in the open position of the feedwater start-up control valve. With this failure of the feedwater start-up control valve, the break will not be isolated until the tripping of the condensate and heater feed pumps on low hotwell level. Therefore the consequences of the revised analysis are additional flooding beyond that previously analyzed.

The feedwater and main steam line break analyses have been reviewed and are in the comment incorporation stage. The final report will be issued by November 1, 1989. After approval of the final report, a UFSAR change will be submitted with the next annual UFSAR update in March 1990.

Initial Plant Conditions:

Operational Condition: 1 (Power Operation)
Reactor Power: 99.5 Percent
Reactor Temperature: 535 degrees Fahrenheit
Reactor Pressure: 1010 psig

Description of Event:

The analysis of the postulated feedwater (SJ) line break in the steam tunnel is described in Updated Final Safety Analysis Report (UFSAR) section 3.6.2.2.2. The UFSAR analysis was completed in the early 1970's. Documentation supporting the conclusions reached has been searched for and cannot be found. Therefore a new analysis was done to support the UFSAR conclusions. A difference was found between the new analysis and the UFSAR conclusions. For the feedwater line break scenario, the single failure assumed in the UFSAR was that of a fast closing (8 seconds) inboard feedwater pump discharge valve (ISV) closes. The break scenario ends when the slow (88 second) outboard feedwater pump discharge valves close (ISV) and High Pressure Coolant Injection (HPCI) (BJ) restores Reactor Pressure Vessel (RPV) level.

The new analysis of the feedwater line break scenario assumes failure of the feedwater startup control valve (TV) in the open position. This single failure was selected to maximize flooding in affected areas of the plant. The feedwater line break scenario with the single failure of the startup control valve is therefore more conservative. The feedwater line break would not isolate until the tripping of the condensate (SG) and heater feed pumps (P) on low hotwell level (SD). No operator action was assumed for the 8 1/2 minutes during which water flows through the postulated pipe break. Supporting documentation is provided in Tables I, II, and III, and the figure attached to this Licensee Event Report (LER).

Cause of Event:

A review of pipe break documentation supporting the original FSAR and existing UFSAR analyses was conducted. Documentation for all pipe break scenarios exists with the exception of the main steam (SB) and feedwater line breaks in the steam tunnel. The UFSAR conclusions resulted from an analysis completed in the early 1970's. Documentation supporting these existing UFSAR conclusions cannot be located, therefore a new UFSAR analysis was developed for the main steam and feedwater line breaks. The UFSAR analysis for feedwater line break was found to be non-conservative.

Analysis of Event:

The new analysis with the feedwater line break in the Steam Tunnel has different consequences. The Steam Tunnel (NM) and Auxiliary Building (NF) first floor, where the Reactor Building Closed Cooling Water (RBCCW) (CC) heat exchanger room is located, reach peak flood depths of 4'6" and 3'10", respectively. The Steam Tunnel and RBCCW heat exchanger room flood elevations then begin to decrease due to flow through the Steam Tunnel floor and equipment drains (DRN). The RBCCW heat exchanger room drains to the northeast and southeast corner rooms of the Auxiliary Building.

The northeast corner room peak flood depth of 7' is reached due to floor drain flow into the sump. Equipment drain flow causes flooding in the southeast corner room to a depth of 8' at which time flooding into the torus room occurs. The torus room flood depth reaches 11". With the RBCCW heat exchanger floor drains open, the evaluation predicts that the control air compressor room will become flooded. When preliminary results of the analysis were received in 1987, to mitigate possible flooding in this room plugs were promptly installed in the RBCCW room floor drains. Plugging of the floor drains eliminates flooding in the control air compressor room and reduces northeast corner room flooding. However, it results in substantially more flooding in the southeast corner room. More flooding of the southeast corner room will have no impact on safe shutdown of the plant since Division 2 of Core Spray is assumed to be lost even without plugging the floor drains. The original UFSAR analysis assumed that the drainage system is sufficient to remove the flood water from the Reactor/Auxiliary Buildings. As a result, the sub-basement corner rooms, air compressor room, and torus room are not flooded.

Reanalysis of the main steam line break in the Steam Tunnel shows that the break flow and flow path are similar to those prescribed in UFSAR 3.6.2.2.1. Flooding levels due to the main steam line break are bounded by the feedwater line break. The resultant peak temperature, pressure and humidity are shown in Table III.

The safe shutdown path considers systems necessary to scram the reactor, depressurize the reactor, and to establish and maintain the shutdown cooling mode of the Residual Heat Removal (RHR) (BO) system. For feedwater line breaks, the availability of offsite power maximizes the consequential effects of flooding. Therefore, offsite power was assumed to be available. If offsite power is not available, the condensate and heater feed pumps will trip, thus ending the break scenario sooner. No water will be lost from the reactor since the feedwater check valves are designed to close immediately. HPCI will restore water level to compensate the loss of feedwater flow. The vessel can be manually depressurized by using the Main Steam Safety Relief Valves. After pressure reduction, the operator places the RHR System (Division 1 or Division 2) in the Low Pressure Coolant Injection (BO) mode. The Residual Heat Removal Service Water (RHRSW) (BI) System is used as the heat sink in the RHR cooling mode.

A preliminary evaluation of the effects of a feedwater and main steam line break has been completed. The feedwater evaluation shows the first floor Auxiliary Building containing the RBCCW, the northeast corner room containing Division 1 Core Spray (BM) and Reactor Core Isolation Cooling (RCIC) (BN), the southeast corner room containing Division 2 Core Spray, and the torus room would be flood to the elevations shown in Table II. Under these conditions the plant can achieve safe shutdown by using the Automatic Depressurization System (ADS) (BF) and both divisions of RHR. The effects of a main steam line break are confined to the Steam Tunnel, first floor Auxiliary Building and the Turbine Building. The Steam Tunnel temperature is bounded by the existing temperature in the UFSAR. The higher Steam Tunnel pressure has been evaluated and found to be acceptable. The higher temperature and pressure in the first floor Auxiliary Building containing RBCCW has no impact to safe shutdown since no safety related equipment is located in this area for both the feedwater and main steam line break in the Steam Tunnel.

In UFSAR Section 3.4.4.4, Internal Flood Protection, under site flooding conditions both divisions of Core Spray, and HPCI, RCIC, the torus room and control air rooms are flooded. With the loss of Divisions 1 and 2 of Core Spray, RCIC and HPCI, safe shutdown capability of the reactor is not affected. ADS and both divisions of RHR are still available for safe shutdown of the plant. Therefore the consequences of the feedwater line break are bounded by this analysis.

Corrective Actions:

When the UFSAR feedwater/main steam line break analyses documentation was determined to be missing, a further review of other pipe break analysis documentation was conducted for other systems. This review concluded that appropriate documentation is available for all other postulated breaks.

The feedwater and main steam line break analyses have been reviewed and are in the comment incorporation stage. The final report will be issued by the vendor by November 1, 1989. After approval of the final report, a UFSAR change will be submitted with the next annual UFSAR update in March 1990.

Previous Similar Events:

This is the first LER describing an inaccurate UFSAR analysis.

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

TABLE I

APPROXIMATE TIME (SEC) UFSAR	APPROXIMATE TIME (SEC) REVISED	FLOW RATE (GPM) UFSAR	FLOW RATE (GPM) REVISED	EVENT UFSAR	EVENT REVISED
0	0	0		BREAK	BREAK
0+	0+	38,000	43,000	FEEDWATER FLOW LIMITED TO 38,000 GPM SO AS NOT TO TRIP RFP AND HEATER FEED PUMP.	o FEED WATER FLOW JUMPS TO 43,000 GPM. o RFP TRIPPED ON LOW SUCTION PRESSURE.
	7		43,000		o REACTOR WATER LEVEL 3 TRIP INITIATES A SCRAM TRIP. o SCRAM TRIP INITIATES CLOSURE OF FEEDWATER PUMP DISCHARGE VALVES. o POST SCRAM FEEDWATER CONTROL AUTOMATICALLY PUT INTO SERVICE.
11		38,000		REACTOR WATER LEVEL 3 TRIP A. SCRAM B. REACTOR FEEDWATER PUMP DISCHARGE VALVES START CLOSING.	
	14		43,000		o STEAM TUNNEL LEAK DETECTION SYSTEM TEMPERATURE EXCEED
19		23,000		ONE FAST CLOSURE RATE (8 SEC). FEEDWATER PUMP DISCHARGE VALVE CLOSES, OTHER CONSIDERED TO HAVE FAILED TO CLOSED.	
20		23,000		PRESSURE RELIEF DOORS OPEN. WATER FLOODS FIRST FLOOR OF AUXILIARY BUIDLING.	
	22		43,000		AFTER AN 8.0 SECOND INSTRUMENT CHANNEL RESPONSE TIME MSIV CLOSURE IS INITIATED.
	25		43,000		VESSEL WATER LEVEL (L2) TRIP INITIATES HPCI AND RCIC OPERATION. RCIC IS NOT TAKEN CREDIT FOR IN THE SCENARIO, SINCE IT IS NOT ENVIRONMENTALLY QUALIFIED AND BECAUSE THE NE CORNER ROOM IS AFFECTED BY FLOODING.
	32		43,000		AFTER A 10 SECOND MSIV CLOSURE TIME THE MSIV'S ARE FULLY CLOSED.
	51		43,000		AFTER THE HPCI INITIATION SIGNAL, HPCI REACHES RATED FLOW WITHIN 25 SECONDS.
	57		21,800		AFTER AN 8 SECOND DELAY (IN ADDITION TO THE 30 SECOND POST DELAY) AND A 12 SECOND CLOSURE TIME, FAST CLOSURE VALVES V12-2531 AND V12-2532 ARE CLOSED AND ALL FLOW IS FORCED THROUGH THE START UP LEVEL CONTROL VALVE.

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APPROXIMATE TIME (SEC) UFSAR	REVISED	FLOW RATE (GPM) UFSAR	REVISED	EVENT UFSAR	REVISED
	64		20,000		THE NUMBER 6 FEEDWATER HEATER LEVEL CONTROL VALVES ARE CLOSED AND FLOW THROUGH THE HEATER DRAIN PUMPS IS ISOLATED FROM THE BREAK.
99		8,000		SLOW CLOSURE RATE (88 SEC). FEEDWATER PUMP DISCHARGE VALVES CLOSE.	
	117		20,000		THE RFP SLOW CLOSURE DISCHARGE VALVES ARE FULLY CLOSED (V12-2503, V12-2504). THIS HAS NO AFFECT ON THE BREAK FLOW BUT IS NOTED TO PROVIDE ASSURANCE THAT FAILURE OF THE FAST CLOSURE VALVES WOULD NOT BE AS SEVERE AS FAILURE OF THE START UP CONTROL VALVE. CLOSURE OF THESE VALVES REQUIRES 80 SECONDS PLUS A 30 SECOND DELAY
150		8,000		REACTOR WATER LEVEL 2 TRIP A. RCIC AND HPCI START B. MSIVS START CLOSING	
190		3,000		REACTOR FEEDWATER PUMPS COAST TO A STOP.	
	274		20,000		REACTOR WATER LEVEL IS RESTORED BY HPCI, WHICH WOULD CLOSE THE STARTUP CONTROL VALVES HAD IT NOT BEEN ASSUMED TO FAIL.
340		0		REACTOR WATER LEVEL RECOVERS. FEEDWATER BYPASS LINE STARTUP CONTROL VALVE CLOSES.	
	447		20,000		CONDENSATE AND HEATER FEED PUMPS TRIP ON LOW HOTWELL LEVEL AND PUMPED FLOW IS ASSUMED TO DECREASE TO ZERO. THE WATER INVENTORY (13275 GAL.) DOWNSTREAM OF THE RFP'S IS ASSUMED TO BE DISCHARGED FROM THE BREAK OVER A 1 MINUTE PERIOD BY GRAVITY FLOW.
	507		0		THE WATER INVENTORY IN THE PIPING IS TOTALLY DISCHARGED.
TOTAL FLOW (GAL)		62,333	185,734		

TABLE II

FLOOD HEIGHT WITH RBCCW
HEAT EXCHANGER ROOM FLOOR
DRAINS PLUGGED

Affected Area	Flood Elevation	Flood Depth
Steam Tunnel	587.98'	53.7"
RBCCW Room	587.31'	45.8"
Control Air Compressor Room	551.00'*	0.0"*
NE Corner Room	548.74'	80.9"
SE Corner Room	548.67'	104.0"
Torus Room	540.89'	10.7"

* This room is unaffected by flooding but is listed because of its significance to the evaluations.

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TABLE III

	TEMPERATURE		PRESSURE		HUMIDITY	
	Revised	UFSAR	Revised	UFSAR	Revised	UFSAR
Steam Tunnel	227°F	260°F	4.9 PSIG	3.4 PSIG	100%	100%
1st Aux. Bldg. (RBCCW Room)g	215°F	160°F	0.8 PSIG	.25 PSIG	100%	100%

*U.S. GPO: 1988-520-589/00070

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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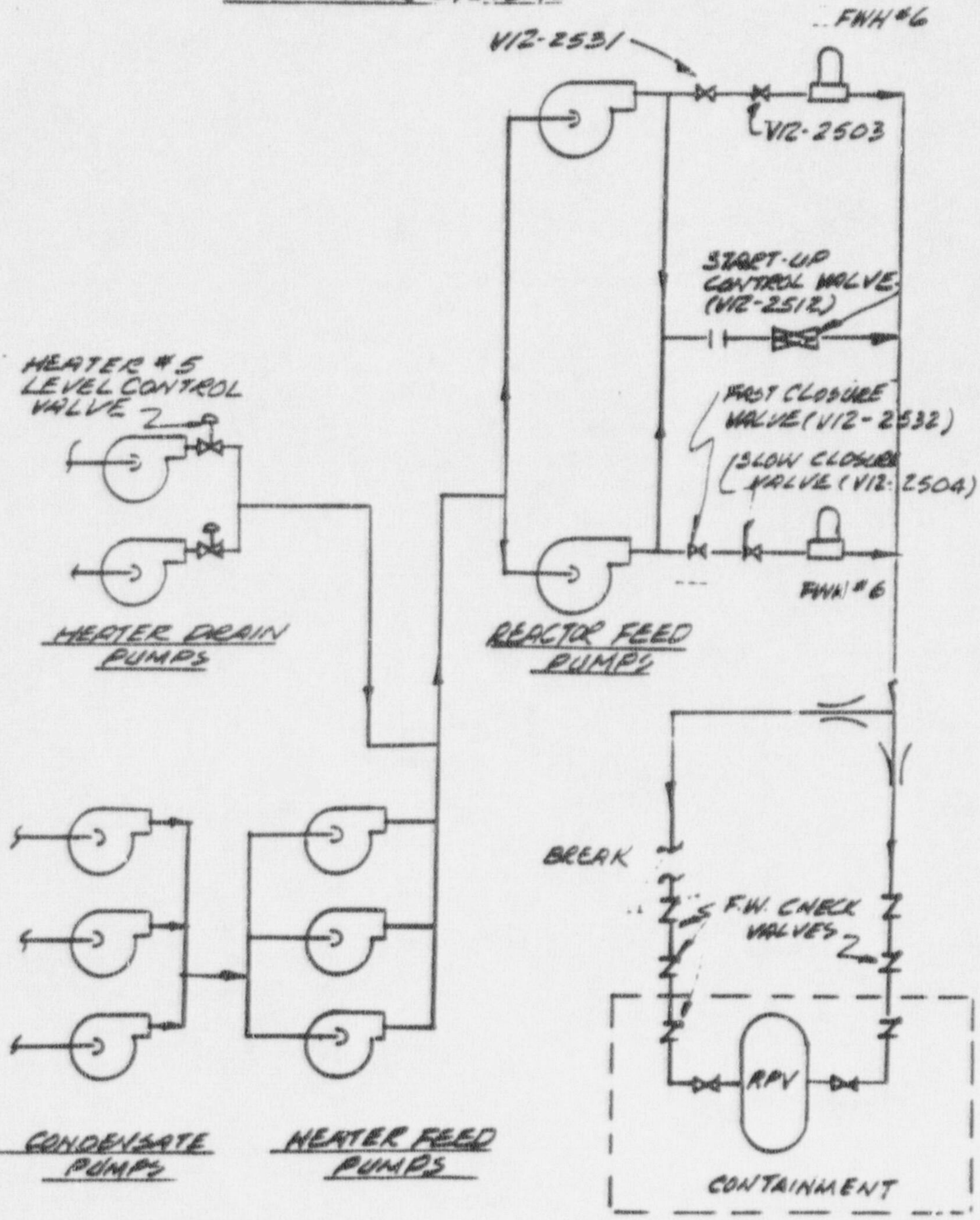
TEXT (If more space is required, use additional NRC Form 366A's) (17)

FIGURE 1
SIMPLIFIED CONDENSATE & FEEDWATER SYSTEMS



*U.S. GPO: 1988-520-589/00070