NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PLANT EDWIN I. HATCH, UNIT 1	0 5 0 0 0 3 2 1	1 OF 7

TITLE (4)
ENGINEERED SAFETY FEATURES ACTUATIONS CAUSED BY PURGE DAM MATERIAL IN INSTRUMENT LINE

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 5	0 1	9 3	9 3	0 0 6	0 0	0 5	2 8	9 3		0 5 0 0 0

OPERATING MODE (9): 4
POWER LEVEL: 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

[ ] 20.402(b)	[ ] 20.405(c)	[X] 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)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below)
[ ] 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	
STEVEN B. TIPPS, MANAGER NUCLEAR SAFETY AND COMPLIANCE, HATCH	912	367-7851

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES(If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15) — MONTH | DAY | YEAR

ABSTRACT (16)

On 5/1/93 at 1215 CDT, Unit 1 was nearing the end of a refueling outage in the Cold Shutdown mode with all control rods fully inserted. Reactor pressure was at approximately 100 psig, and reactor coolant temperature was at approximately 145 degrees Fahrenheit and increasing in preparation for the ASME XI Class 1 System Leakage Test. At that time a full Reactor Protection System actuation signal was received on a low water level signal. The low water level signal also produced a trip signal to several Group 2 Primary Containment Isolation System (PCIS) valves, per design. Licensed Operations personnel verified that actual reactor water level was not low by comparing instrumentation. Two unsuccessful attempts were made to correct the condition by filling the affected instruments' variable sensing line and performing various instrument valve manipulations. When these efforts proved unsuccessful, Instrument and Control technicians filled and vented the reference line of the affected instrumentation. While using a manual hydrostatic test pump to fill the reference line, some resistance to flow was encountered, indicating the line was partially plugged. Further pressurization, however, cleared the obstruction. The root cause of this event was pressurization of a reactor water level instrument reference leg due to blockage in the line coupled with minute leakage through instrument diaphragms or equalizing valves which permitted jet pump pressure to be felt in the reference leg. The blockage in the line is believed to have been soluble weld purge dam material used during an outage-related repair to the sensing line. Corrective actions for this event included clearing the obstruction and verifying level instrument operation. Also, the procedure controlling weld processes on pressure boundaries will be revised and the event will be discussed with personnel who perform welding.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX)

DESCRIPTION OF EVENT

On 5/1/93 at 1215 CDT, Unit 1 was nearing the end of a refueling outage in the Cold Shutdown mode with all control rods fully inserted. Reactor pressure at approximately 100 psig, and reactor coolant temperature at approximately 145 degrees Fahrenheit. At that time, licensed shift personnel were heating the Reactor Pressure Vessel using the Reactor Recirculation System (EIIS Code AD) Pumps in preparation for a Class 1 System Leakage Test in accordance with ASME XI, IWB-2500. During the heat-up, a full Reactor Protection System (RPS, EIIS Code JC) actuation signal was unexpectedly received on a false low reactor water level indication. Several Group 2 Primary Containment Isolation System (PCIS, EIIS Code JM) valves closed as well. The "A" channel reactor water level instruments were fully upscale, indicating that the reactor vessel was still flooded as required for the System Leakage Test. However, when licensed Operations personnel observed the strip chart recorder for the "B" channel reactor water level instruments, they saw that indications were not upscale as expected, but were onscale, erratic, and trending downward. Other level instrumentation was checked for comparison and operators were satisfied that reactor water level was still as high as expected. Therefore, Limiting Conditions for Operation (LCOs) were written against the "B" channel level instruments and they were declared inoperable. Also, the reactor vessel heat-up was suspended, and Recirculation System flow was reduced. At 1350 CDT, operators took the additional precaution of changing reactor water level slightly and confirming that the water level instruments which were still considered operable would actually track the level changes as expected, particularly the high level instruments which were indicating onscale.

As investigation into the event was in progress, operators noted that the indications of the "B" reactor water level instruments began to increase slowly, eventually moving upscale as they should have been. By 1635 CDT, indicated level had increased to the point where the scram signal and the Group 2 PCIS closure signal had cleared. Therefore, the scram and PCIS isolation were reset. Continuing investigation into the event revealed that all level instruments which had previously failed downscale shared a common sensing line. Since this could indicate a failure of the variable leg sensing line, personnel inspected the sensing line, but found no leaks. By 1733 CDT, Instrument and Control technicians (I&C techs) performed a fill and vent procedure on the variable leg sensing line shared by these level instruments and found the piping to be full as expected, and flow into the piping was unrestricted.

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During these activities, the engineer responsible for the System Leakage Test observed that there was a connection between the affected variable leg and the line through which air is injected into the reactor vessel as part of the test. If a leak existed in the boundary valves in this connecting line, it was theorized that air might have been injected into the variable leg resulting in an apparent reduction in reactor level. Therefore, Operations personnel directed that the equalizing valves on instruments connected to the air injection source be checked. It was then discovered that one of the equalizing valves was not fully seated as it should have been. When this valve was tightened on its seat, Operations personnel believed the cause of the problem had been identified and corrected, and they authorized resumption of activities related to the vessel System Leakage Test. On 5/2/93 at 0110 CDT, as Reactor Recirculation System flow was increased to provide heat for pressurization, reactor water level as indicated by the "B" channel instruments again began to decrease in an erratic manner. Although no actuations of Engineered Safety Features occurred, activities related to the Class 1 System Leakage Test were again suspended by 0200 CDT.

The Shift Technical Advisor then suggested that if certain instruments were leaking through their diaphragms, a flow path would be created between the reactor jet pump (EIIS Code AD) discharge and the reference leg sensing line which is also common to the affected instruments. I&C techs then performed testing on these instruments to determine whether signs of leak-by could be detected. They identified two instruments as possibly having minor leakage through their diaphragms or their equalizing valves. When I&C techs had isolated these instruments, Operations personnel again attempted to resume pressurizing the reactor vessel by using the Recirculation Pumps. However, as flow was increased using the Reactor Recirculation Pumps, reactor water level as indicated by the "B" channel instruments once again began to decrease erratically. Therefore, licensed personnel suspended the test activities again and requested I&C techs to repeat the fill and vent procedure on the variable leg and perform the procedure on the reference leg of all affected instrumentation.

I&C techs used the Demineralized Water System (EIIS Code KC) to backfill the variable leg and again found the line to be free of air and unobstructed. However, when the reference leg was being filled, the demineralized water pressure indication was not as expected, reading between reactor pressure and Demineralized Water System pressure. Therefore, I&C techs elected to fill the line, per procedure, using a manual hydrostatic test pump which provides finer control over the fill operation as well as potentially higher source pressure. As they injected water into the line using the hydrostatic test pump, I&C techs observed the pressure in the reference leg increase to approximately 350 psig before suddenly dropping to 100 psig, or approximately equal to reactor pressure. It was apparent at this point that a blockage had been cleared from the reference leg piping. Following this, there were no further anomalies observed in the fill and vent process. When the reference leg had been filled and vented, the affected reactor water level instruments were returned to

service and all indicated normally. There were no further abnormal indications observed on "B" reactor water level instruments, and all affected instruments were declared operable by 1340 CDT on 5/2/93. Subsequently, the reactor vessel Class 1 System Leakage Test was successfully resumed and the reactor vessel was fully pressurized by 0238 CDT on 5/3/93.

CAUSES OF EVENT

The immediate cause of the false low level indications in the "B" channel reactor water level instruments appears to have been pressurization of the reference line connected to these instruments. This reference line is believed to have pressurized because it was obstructed on one end and exposed to reactor jet pump pressure on the other. Reactor water level is sensed in these instruments by measuring differential pressure between the reference and variable legs. If reference leg pressure increases relative to the variable leg pressure, indicated level goes down. Therefore, it is apparent that with blockage present in the reference leg and Recirculation System flow increasing, the reference leg was slightly pressurized by exposure to jet pump pressure. Thus the instruments sensed the change and reacted, per design, by initiating a scram and Group 2 PCIS valve isolation on low reactor water level.

The obstruction which existed in the reference line of the affected instrumentation is believed to have had its origin in an outage-related maintenance activity which required cutting and re-welding the one-inch reference leg sensing line. After the line had been cut, a small amount of water kept trickling out of the pipe, preventing it from being re-welded. Therefore, a soluble weld purge dam material (known as "rice paper") was formed into a plug and placed into the line to stop the water and keep the weld site dry. This is an acceptable welding practice. However, it is apparent that in this case, too much purge dam material was used. It appears that the size of the paper plug coupled with the presence of an air bubble in the line (a result of cutting and partially draining the pipe) prevented the purge dam material from completely dissolving as is normally expected. Therefore, when I&C techs began filling the reference line with demineralized water, the purge dam and air bubble were pushed through the pipe toward a restricting orifice near the condensing chamber. I&C techs then used the manual hydrostatic test pump to pressurize the reference line which resulted moving the obstruction up against the restricting orifice. Once the obstruction encountered the orifice, the increased pressure produced by the test pump resulted in wetting the purge dam material, softening it, and then pushing it through the restricting orifice. After the purge dam was exposed to the larger volume of water in the condensing chamber it would have dissolved completely as designed.

While the purge dam material was still present in the reference leg, it enabled the reference leg to be pressurized. The source of the pressure appears to have been minute leaks through instrument diaphragms or instrument equalizing valves which established a flow path between one or more jet pump pressure sensing lines and the reactor water level instrument reference leg. With the reference leg plugged by the purge dam, therefore, jet pump pressure built up in the

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reference leg as Recirculation System flow was increased. This changed the differential pressure sensed by the level instruments on the affected line, producing an indication of decreasing reactor water level. Then, when System Leakage Test preparations were suspended each time, Reactor Recirculation System flow was reduced, permitting the pressure build-up in the reference line to be relieved as the pressure in the jet pump flow sensing line was reduced. A test apparatus was constructed to test this hypothesis. It was shown that a rice paper obstruction in a one-inch line could hold sufficient pressure to produce the reactor water level indications which were actually observed during the event.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(iv) because several unexpected actuations of Engineered Safety Features occurred. Specifically, a full scram signal was received and several Group 2 PCIS valves closed in response to sensed low reactor water level.

The Reactor Protection System is designed to initiate a reactor scram when any of several instrumentation systems indicate abnormal process conditions exist. Among these conditions are high reactor vessel pressure, closure of the Main Turbine (EIIS Code TA) stop valves when operating above 30% rated thermal power, high neutron flux, low reactor water level, and others. The instrument piping for the reactor water level instrumentation has a fail-safe design feature such that a full scram signal will be produced upon the loss of any one level instrumentation leg.

The Primary Containment Isolation System is designed to isolate the lines penetrating the Primary Containment when process conditions indicate the possible existence of a leak of radioactive material from nuclear process barriers. PCIS valves are divided by design into several groups. Group 2 PCIS valves are generally those valves whose lines penetrate Primary Containment and communicate with the free space inside it without communicating directly with the nuclear steam system. Among the many signals which can produce a Group 2 PCIS valve isolation is low reactor water level, the same level which can produce an RPS actuation.

In this event, an obstruction from weld purge dam material in an instrument line permitted a reference line to be pressurized. This resulted in apparent low water level indications on instruments served by this line. When the indications reached the trip setpoint, the instruments produced a full scram per design and further resulted in isolations of affected Group 2 PCIS valves per design. Soluble purge dam materials are often used in welding applications where it is necessary to create a temporary gas barrier inside a pipe for the inert gas used in certain types of weld processes. They are also commonly used in welding piping systems to prevent water from intruding into a weld area. In this case, it appears that too much purge dam material was used to form a

barrier against water intrusion. In addition, cutting the pipe resulted in the formation of an air bubble which apparently further hindered the material from being thoroughly soaked with water following repair of the pipe. Therefore, the material did not dissolve as expected and obstructed the instrument line.

The effect of this obstruction on the instruments served by this line was not immediately apparent to Operations personnel because of the plant condition which prevailed at the time of the event. The reactor vessel was flooded so that the affected instruments should have been reading upscale at all times during the System Leakage Test. Operations personnel knew that the instruments were malfunctioning only after they began indicating onscale. However, in the event that the plugged reference line had, for some reason, not been discovered during the System Leakage Test, it would have been discovered when vessel level was lowered in preparation for power operation.

If reactor vessel water level were reduced, the affected instruments would have been expected to begin reading onscale rather than upscale. Once the instruments were reading onscale, the discrepancy in instrument performance between the "A" and "B" instruments would have become obvious to Control Room operators and they would have reacted by suspending reactor water level changes until instrumentation indications were reconciled. In the event that the obstruction had not been evident during the reduction of reactor water level to normal operating levels, its presence would have been apparent during nuclear heat-up. The obstruction would have prevented reactor pressure from being fully felt in the reference line. As pressure increased in the variable line, reactor level as sensed by the affected instruments would have indicated upscale, producing high reactor water level trip annunciation. In that event, licensed Operations personnel would have suspended further power increases until the cause of the annunciation was identified and corrected.

Based on this analysis, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all operating conditions.

CORRECTIVE ACTIONS

1. The weld purge dam material was flushed from the line by the operation of the manual hydrostatic test pump.

2. The operation of affected reactor water level instrumentation has been verified by observation. There have been no further malfunctions of instruments which share this sensing line.

3. Procedure 51GM-MNT-025-0S, "GENERAL WELDING REQUIREMENTS FOR PRESSURE BOUNDARY APPLICATIONS," will be revised to require post-maintenance functional testing if weld soluble purge material is used to plug a pipe. This action will be completed by 8/31/93.

4. As an interim action, this event will be discussed in regularly scheduled training sessions with personnel whose duties include welding. This action will be completed by 06/30/93.

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ADDITIONAL INFORMATION

1. Other Affected Systems: No systems other than those mentioned in this report were affected by this event.

2. Previous similar events: No events were reported in the past two years in which a plugged instrument line led to erroneous indications or actuations of Engineered Safety Features.

3. Failed Equipment Information: No failed equipment contributed to or resulted from this event.