Commonwealth Edison
Dresden Nuclear Power Station
R.R. #1
Morris, Illinois 60450
Telephone 815/942-2920

April 05, 1993

CWS PMLTR 93-0141

U. S. Nuclear Regulatory Commission
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Licensee Event Report 93-001-01, Docket 0050237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10 CFR 50.73(a)(2)(ii). This supplement is being submitted to supply additional information on this event.

for 4-5-93

Charles W. Schroeder
Station Manager
Dresden Station

CWS/slb

cc: A. Bert Davis, Regional Administrator, Region III
NRC Resident Inspector's Office
File/NRC
File/Numerical

080015

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Dresden Nuclear Power Station, Unit 2	0 5 0 0 0 2 3 7	1 of 0 6

Title (4) Historical Analysis determined unanalyzed support removed on Isolation Condenser fill line outside FSAR Limits

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)
									N/A	0 5 0 0 0
0 1	0 8	9 3	9 3	0 0 1	0 0	0 2	0 5	9 3	N/A	0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
POWER LEVEL (10)	0 7 3	___ 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)	___ 50.73(a)(2)(vii)	___ Other (Specify
		___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	in Abstract
		___ 20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	below and in
		___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	Text)

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER
	AREA CODE
Dennis Dolecheck, Technical Staff Engineer Ext. 3260	8 1 5 9 4 2 - 2 9 2 0

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 fifteen single-space typewritten lines) (16)

On December 23, 1992, Unit 2 was in the RUN mode at 73% core rated thermal power. At approximately 0730 hours, as part of plant modification M12-2-90-057C "Isolation Condenser Upgrade, Non-Outage Portion", pipe fitters removed sway support M-1170D-58 from Isolation Condenser [BL] Clean Demineralized Water [KC] Fill Line 2-4388-4"-LX. It was determined that the modification design was improperly scoped and that the line was not analyzed with the support removed. Work began immediately to install the new support. On December 23, at 1645 hours, NED determined that with the support removed the line was operable. The new support was installed by December 24, 1992. On January 8, 1993, a historical analysis determined that with the support removed the FSAR limits were exceeded. On January 29th a walkdown of the subject piping revealed that a four way restraint was missing. On February 3, 1993 it was determined that ENS notification of this event should have been made on January 8, 1993. ENS notification was made at 1702 hours on February 3, 1993. The apparent cause of the initial event was procedural deficiency. The designers and installers walkdown checklists did not provide enough direction to prompt consideration of how temporarily removing the support would affect unit operations. The immediate corrective action was to install the new support. Four similar occurrences were identified, but due to root cause differences this is considered an isolated event.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power.

Nuclear Tracking System (NTS) tracking code numbers are identified in the text as (XXX-XXX-XX-XXXXX).

EVENT IDENTIFICATION:

A. CONDITIONS PRIOR TO EVENT:

Unit: 2 Event Date: December 23, 1992 Event Time: 0730 Hours

Reactor Modes: N Mode Name: Run Power Level: 73%

Reactor Coolant System (RCS) Pressure: 978 psig

B. DESCRIPTION OF EVENT:

On December 23, 1992, Unit 2 was in the RUN mode at 73% core rated thermal power. At approximately 0730 hours, as part of plant modification M12-2-90-057C "Isolation Condenser Upgrade, Non-Outage Portion", pipe fitters removed sway support [SPT] M-1170D-58 from Isolation Condenser [BL] Clean Demineralized Water [KC] Fill Line 2-4388-4"-LX. This support was removed to allow the installation of the new support per the design. During a routine walk down the Engineering and Construction (ENC) engineer noticed that the support had been removed, and immediately notified a shift supervisor of the condition. This notification was made at approximately 0800 hours. No systems affecting this portion of the Isolation Condenser System were inoperable during this event. No compensatory actions were deemed necessary.

The cognizant nuclear engineering department (NED) engineer was contacted to verify that the design had really intended to remove this support while the unit was in operation. At approximately 1350 hours it was determined by NED that, while the design did show the support to be removed during the unit operation, the line had not actually been analyzed for this condition.

The immediate corrective actions were to direct ENC to either reinstall the old support or install the new support, whichever would be the most expedient. The old support could not be reinstalled due to the anchor bolt holes being damaged during anchor removal. Work began immediately to install the new support. Problem Identification Form (12-200-92-52000) (PIF) was initiated to track the event. On December 23, at 1645 hours NED determined that the line was operable with the support removed. This information was transmitted to the stations technical staff department and to the shift engineer. Station PIF 12-200-92-52000 was then closed to ENC PIF 925-200-92-01600 to track the historical analysis calculation completion.

On December 24, 1992, at approximately 1400 hours the new support installation was completed. The total time the support was removed was less than 31 hours.

At 1600 hours on January 8, 1993, a historical analysis calculation was completed, showing that for the time the support was removed the line was outside its design basis as described in the FSAR. At this time an Assistant Technical Staff Supervisor (ATSS) began to go through the process of identifying the reportability of this event. He had an initial brief discussion was with the Technical Staff supervisor (TSS)but did not refer to the Commonwealth Edison Reportability manual. They decided that it was thirty day reportable, but was not reportable via ENS. This conclusion was based on the fact that since the new support was installed that the plant was no longer in a degraded condition. At this time

the specific Federal Regulation for reportability was not determined. No discussion with the Shift Engineer as required occurred even though a condition was identified in which a plant component was outside its design basis. On January 13, 1993, written notification was received by the station technical staff from engineering confirming the verbal notification of January 8, 1993, that with the support removed the plant was outside the FSAR piping stress allowables. Station PIF 237-180-93-01 was generated based on this letter to document the event and was carried to the Shift Engineer. Since the event did not appear to fit into any category of the Commonwealth Edison Reportability Manual, a discussion was held between the same ATSS and nuclear licensing to determine the reportability of the event. The ATSS also had discussions with Quad Cities Station nuclear licensing and with the author of the reportability manual. The Dresden nuclear licensing personnel apparently did not have a full understanding of the situation when they advised the ATSS, although the ATSS thought that they did have full understanding. A section of the reportability manual, 10 CFR 50.73(a)(2)(i)(B), was chosen that appeared to make the best fit. This section is 30 day reportable. Also on Janaury 13, 1993, after disucssion with the ATSS, the Shift Engineer also concluded that an ENS notification was not required. On approximately January 15, 1993, while determining which department would write the LER a discussion was held between the ATSS that made the original determination and two other ATSS's. It was agreed that the issue wasn't fully clear, but that the method of reporting the event seemed reasonable. On January 21, the Event Screening Committee concurred with the designation of 10 CFR 50.73(a)(2)(i)(B).

On January 22, 1993 the Technical Staff Engineer (TSE) was continuing the investigation of the event and noted that on the problem screening form, DAP 2-27C, the reportability was marked with 10 CFR 50.73(a)(2)(i)(B). The TSE determined from his interpretation of the event that the proper section was 10 CFR 50.73(a)(2)(ii)(B) and concluded that the problem screening form had a typographical error. the TSE noted on the problem screening form that he believed the code should be changed. On January 25, 1993 the draft report was forwarded to the Station Regulatory Assurance (RA) department for review. During that review it was noted the ENS notification was not made, though it was required if the event was reportable under 10 CFR 50.73(a)(2)(ii)(B). Later that day RA held discussion with the ATSS that made the original determination, nuclear licensing, with the TSE who had prepared the report. No consensus was reached as to whether the reportability should be changed.

While preparing the safety analysis of the event the TSE and RA concluded that the letter issued from engineering was somewhat vague in listing which sections of piping were over stressed. They decided that it would be best if a walkdown of the piping model was performed to assure that the model was correct. On January 29, 1993, at approximately 1600 hours, with the unit shutdown and the Isolation Condenser not required, a walkdown of the associated piping revealed that a penetration that was credited as a four way restrain was filled with foam and not grout. This discovery rendered the piping model invalid. Station PIF 237-201-93-083 was written to document the event and the Shift was notified. Therefore the analysis used which assumed a rigid support to determine that the piping system was operable or outside FSAR allowables was also potentially invalid.

On February 2, 1993 a discussion was held between the TSS and the TSE preparing the report as to the direction that should be taken on reportability since no analysis was now available to make the determination of whether the piping did or did not meet FSAR piping stress allowables. It was agreed that since it was extremely unlikely that another missing restraint would reduce piping stresses, that the LER should be issued with a supplement to follow once the new analysis was completed. The RA department view was that the LER should be reported as voluntary until the new analysis was completed based on the assumption that the final analysis would be outside the FSAR allowables. On February 3, 1993 at 1615 hours, the Station determined that a one hour ENS notification was required for the support removal event. This notification was made at 1702 hours. Due to the complexity of this event, a two page fax transmittal was also sent to assist in understanding this event.

While completing the piping analysis it was determined that the 4 way restraint was not missing. The walkdown markings were misinterpreted, which led to the incorrect belief that the 4 way restraint was missing.

C. APPARENT CAUSE OF EVENT:

This report is being submitted per 10 CFR 50.73(a)(2)(ii)(B). The piping analysis confirmed that the Isolation Condenser drains piping was outside the design basis as described in the FSAR for approximately 31 hours in December, 1992. The apparent cause of this event is procedural deficiency. Both the installers and designers walkdown checklists, procedure ENC QE-62, identified the support needed to be removed and redesigned. The discussion on this support focused only on the need for redesign. No documentation could be identified that indicates that consideration was given to how removal of the support might affect plant operation. The designers and installers walkdown checklists require an answer relative to temporary removal of equipment, but no caution or direction is given as to effects on systems and plant operation. A contributing cause is the method in which each checklist was performed. This modification was broken down into three separate partial modifications. One of which was explicitly meant to accomplish non outage activities. However, when the walkdowns were performed and the checklists filled out for the partial modifications they were not separate. Only one checklist was completed which covered the scope of all the partial modifications. This method of filling out the checklist, especially for such a large modification, would have made it practically impossible to keep the scope of each partial modification separate.

The apparent cause of not properly classifying this event and making the required notification has several causes.

1. Management Deficiency.

   a. On Janaury 8th, 1993 the ATSS should have taken in the information immediately to the Shift Engineer for operations determination of reportability.

   b. The Commonwealth Edison Reportability Manual is unclear regarding the need for reporting a preexisting condition.

   c. In the documentation received from engineering the descriptions of the problem used the phrase "stresses exceeded FSAR allowables". These phrases are not in synch with those used in the CFRs.

2. Personnel Error.

   a. On the initial determination of the event the ATSS did not use the Commonwealth Edison Reportability Manual to determine the classification of the event.

   b. The Shift Engineer did not use the Commonwealth Edison Reportability manual as required by DAP 2-28 to determine the classification of the event.

   c. The communciations between the ATSS and nuclear licensing was not clear. Each person had a different understanding of the problem.

D. SAFETY ANALYSIS OF EVENT:

The safety of the public and the plant was not affected by this event. An analysis determined that the piping was operable throughout the event. Therefore, had the isolation condenser system been required, this event would not have prevented it from meeting its design function.

E. CORRECTIVE ACTIONS:

The immediate corrective actions were to install the new support, and determine that the line was operable.

Corrective actions include:

The designers and installers walkdown procedures, ENC EQ-62, will be revised. If the modification is large or complicated, a separate checklist for each partial modification will be required. Questions concerning removal of equipment will be revised to draw specific attention to how removal could affect plant operation. NTS # 237-180-93-00101

The Downers Grove Nuclear Licensing Department will issue a Lessons Learned notification on this event to all design organizations and this Lessons Learned will be discussed with all station modification personnel. NTS # 237-180-93-00102

The ATSS was counseled that the Shift Engineer is responsible for making reportability decisions not the Technical Staff and that this information should be transmitted to the Shift Engineer promptly.

A recommended revision to the Commonwealth Edision Reportability Manual wil be transmitted to Commonwealth Edison Corporate which clarifies the need for an ENS notification for historical as well as currenlty existing conditions that are outside the design basis as described in the FSAR (NTS 237-200-93-017060).

Site Engineering, Technical Staff, and Operations will develop a methodology for reporting engineering concerns that emphasizes wording addressing "Design Basis" when conditions are identified as outside of that described in the FSAR (NTS 237-200-93-017061).

Operations Manager will have a meeting with the Shift Engineers to reiterate expectations on ENS phone call notifications; specifically, if any doubt exists after using DAP 2-28 and the Commonwealth Edison Reportability Manaual, the Shift Engineer shall make the ENS phone call, or shall contact Regulatory Assurance, Technical Staff, or Engineering if more input is needed (NTS 237-200-93-01708).

On 02/03/93 the Shift Engineers and the Operating Engineers were briefed on this event with emphasis on the need for operations to make reportability determination and notifications.

F. PREVIOUS OCCURRENCES:

A LER search was conducted that revealed the following similar previous occurences. These events have been reviewed and due to the differences in root cause this is considered an isolated event.

LER Number	Title
87-003/0500237	Primary Containment Structural Steel Connection Outside FSAR Design Criteria Due to Apparent Original Construction Oversight.
88-003/0500249	Flued Head Anchor Supports in Excess of Design Criteria Due to Design and Construction Deficiencies.
92-029/0500237	Bellows at Primary Containment Penetration X-125 found Outside FSAR Design Limits Due to Project Instruction Deficiencies.
92-030/0500237	Pipe Supports for the Containment Atmosphere Sampling System not connected to Structural Steel due to Design/Installation Deficiency.

G. COMPONENT FAILURE DATA:

Since there were no component failures during this event, an industry wide NPRDS data base search was not performed.

DAP 02-08
REVISION 10

DAP FORM 2-8D
EVENT SUMMARY AND CAUSE CODES

LER Number
12-2-93-001

[ ]	Lost generation	[ ]	Reactor trip	[ ]	NRC violation, level____
[ ]	Cost > $25,000	[ ]	ESF actuation	[ ]	GSEP event, class_____
[ ]	Hazard or Spill	[ ]	NRC reportable	[ ]	Tech Spec LCO
[ ]	Personnel injury	[X]	LER	[ ]	Potential or future loss
		[ ]	PSE		SALP functional area____

	Component type	Failure mode	Department
X			
X			
X			

	Licensed? L or blank	Level	Department	Type	Detail code
A					
A					
A					

	Type	Detail code	Department
B			
B			
B			

	Type	Detail code
C		

	Type of deficiency	Detail code	Procedure type	
D	I	2	E N C	Deficient Procedure/Checklist
D				
D				

	Type	Detail code	Department
E			
E			
E			

DAP FORM 2-8D