Commonwealth Edison Company
Dresden Generating Station
6500 North Dresden Road
Morris, IL 60450
Tel 815-942-2920

ComEd

May 14, 1996

JSPLTR #96-0074

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

Enclosed is Licensee Event Report 95-021, Revision 1, Docket 50-237 which is being submitted pursuant to 10 CFR 50.73(a)(2)(i), any operation or condition prohibited by the Plant's Technical Specifications.

The supplemental report provides the status of corrective actions and results of the Recirculation Pump Seal Purge modification review.

In addition to those of the original LER this correspondence contains the following commitment:

1. Engineering will re-evaluate the concern identified in IE Notice 90-78 to confirm no other design or analyses issues exist. (2371809502108S1)

If you have any questions, please contact Pete Holland, Dresden Regulatory Assurance Supervisor at (815) 942-2920 extension, 2714.

Sincerely,

J. Stephen Perry
Site Vice President
Dresden Station

Enclosure

cc: H. Miller, Regional Administrator, Region III
NRC Resident Inspector's Office
Illinois Department of Nuclear Safety

210014

IE22
1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Unit 2	05000237	1 OF 6

TITLE (4)
Maximum Thermal Power Exceeded Due to Inadequate Modification Safety Evaluation and Unanalyzed Secondary Containment Bypass Pathway Created

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 NAME	DOCKET NUMBER
									Dresden Unit 3	05000249
12	08	95	95	-- 021 --	01	05	17	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): N

POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(iii)		73.71(b)
	20.2203(a)(1)		20.2203(a)(3)(ii)		50.73(a)(2)(iv)		73.71(c)
	20.2203(a)(2)(i)		20.2203(a)(4)		50.73(a)(2)(v)		OTHER
	20.2203(a)(2)(ii)		50.36(c)(1)		50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(viii)(A)		
	20.2203(a)(2)(iv)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
	20.2203(a)(2)(v)		50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
T. S. Feren, Engineer Ext. 2054 W. A. Poppe, Engineer Ext. 3878	(815) 942-2920

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 December 8, 1995, it was determined that the core thermal power calculation does not correctly account for the Control Rod Drive system flow to the Reactor Recirculation system pump seal purge lines, resulting in a core thermal power calculation that is 0.8 Megawatts Thermal non-conservative. Therefore, any time Unit 2 or Unit 3 reactor was operating at approximately 99.97 percent rated power, the licensed power level may have been exceeded. The event report is provided for Dresden Unit 2 due to DPR-19 license condition 2.G as a specific violation of license condition 2.C(1) reportable per 10CFR50.73(a)(2)(i)(B). There is no Dresden Unit 3, DPR-25, license condition specifically requiring reporting of violations of license conditions. The event is considered as reportable for Dresden Unit 3 (DPR-25) per 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the Technical Specifications. The cause of the event is personnel error resulting in inadequate safety evaluation performed in 1974 for a plant modification. Immediate corrective action was a 1 MWT administrative derate on Dresden Unit 2 and 3.

This supplement is being provided as a result of a corrective action committed to in the original LER. It was discovered that modifications that added the reactor recirculation seal purge system in 1974 created a discrepancy with the FSAR and established an additional potential secondary containment bypass pathway that had not been considered in the offsite dose analyses.

PLANT AND SYSTEM IDENTIFICATION:

General Electric - boiling water reactor - 2527 MWt rated core thermal power

EVENT IDENTIFICATION:

Maximum Thermal Power Exceeded Due to Inadequate Modification Safety Evaluation and Unanalyzed Secondary Containment Bypass Pathway Created.

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2(3) Event Date: December 8, 1995 Event Time: 1600 Hours

Reactor Mode: N(N) Mode Name: N(N) Power Level: 000(100)

Reactor Coolant System Pressure: 000(1006) psig

B. DESCRIPTION OF EVENT:

This supplement is being provided as a result of a corrective action committed to in the original LER. It was discovered that modifications that added the reactor recirculation seal purge system in 1974 created a discrepancy with the FSAR and established an additional potential secondary containment bypass pathway that had not been considered in the offsite dose analyses.

On December 8, 1995, 1600 (Central Standard Time), during a review of industry events and verification of plant configuration, it was determined that Dresden Unit 2 and Dresden Unit 3 have slightly exceeded the licensed power level of 2527 Megawatts Thermal (MWT) at various times in the past. Engineering determined that the plant core thermal power calculation does not account for the Control Rod Drive (CRD) system [AA] flow to the Reactor Recirculation system [AD] pump seal purge lines. Calculations performed by the Nuclear Engineering staff indicated that under bounding conditions, core thermal power would be non-conservatively calculated by approximately 0.8 Megawatts Thermal (0.03 percent rated power). Therefore, any time a reactor was operating at approximately 99.97 percent rated power, the licensed power level may have been exceeded.

The NRC ENS notification center was advised of the event on December 8, 1995 at 1946 (Eastern Time) as a 24 hour report made in accordance with DPR-19 License Condition 2.C(1).

At 1600 on December 8, 1995, Dresden Unit 2 was in a shutdown condition with no fuel in the vessel and Dresden Unit 3 was operating at approximately 100% power. An administrative derate was implemented for Unit 3 to limit maximum thermal power to 2526 MWT. An extensive review of 8 hour averages for core thermal power indicate that a margin of 1 to 5 MWT below the licensed power level is usually maintained. However, it is probable that both units have exceeded their Operating License Condition for maximum power level in the past. Operating Condition 2.C.(1) for Unit 2 (3.A for Unit 3) states that steady state reactor core power level is not to exceed 2527 MWT (100 percent rated power).

The event report is provided for Dresden Unit 2 due to DPR-19 license condition 2.G as a specific violation of license condition 2.C(1) reportable per 10CFR50.73(a)(2)(i)(B). There is no Dresden Unit 3, DPR-25, license condition specifically requiring reporting of violations of license conditions. The event is considered as reportable for Dresden Unit 3 (DPR-25) per 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plants Technical Specifications.

As a result of the committed corrective action (237-180-95-02106), a re-review of the 1974 modifications discovered a discrepancy with UFSAR Section 3.8.2.1.9. These modifications cross connected the Control Rod Drive system with the primary system through two primary system instrument line penetrations. The addition of these non-seismic seal purge lines to the instrument lines outside the drywell was in conflict with the commitments in the FSAR for these types of penetrations. These modifications created an additional potential release path through the Control Rod Drive Hydraulic system.

The evaluation of the Dresden design, performed in 1991 in response to IE Notice 90-78, "Previously Unidentified Release Path from Boiling Water Reactor Control Rod Hydraulic Units", evaluated the potential back leakage to the CST. The evaluation did not identify the potential back leakage through the seal purge lines.

C. CAUSE OF EVENT:

The root cause for the unaccounted CRD flow is a personnel error resulting in an inadequate safety evaluation performed in 1974 for a plant modification. The root cause for the seal purge line pathways not being considered in the plant dose analyses is a personnel error. NRC cause code "A" is applicable for this event. Considering the error did not result in component failure Section 13 of the NRC LER cover sheet is left blank.

A review of plant mechanical drawings M34 and M365 revealed the Recirculation Pump Seal Purge System modification was installed on both units in the 1970's. This changed CRD flow from the original design although core thermal power continued to be calculated based on the original design. A 1974 letter containing a safety evaluation for the plant modifications, M12-2-73-078 and M12-3-73-078, stated that "the function of any piece of equipment or system is not being altered." This statement is incorrect. This is where the error in the safety evaluation occurred. The modification diverts flow prior to the flow element for CRD flow, therefore the CRD flow element no longer functions to measure the CRD system flow assumed in the core thermal power calculation.

D. SAFETY ANALYSIS:

The safety significance of the thermal power limit being exceeded is minimal. The error is much smaller than normal operating margins to restricted power levels as well as operating and design margins for fuel thermal limits.

As a result of this event at several plants, General Electric (GE) issued a letter OG95-858-01 (dated Dec. 13, 1995) to BWR Owners' Group representatives to address the concerns of the heat balance accuracy. It stated that a maximum bias from any plant of 0.1% (Dresden bias is approximately 0.03%) "is very small when compared to the 2% allowance normally associated with safety analyses (i.e. the use of 102% of rated power as the analysis basis). Because of this, these heat balance accuracy concerns are viewed as a potential licensing question, not a safety issue."

The safety significance of the potential back leakage through the Control Rod Drive Hydraulic system is discussed below. If the operating Control Rod Drive (CRD) pump was lost and the standby pump was not started, primary system fluid could potentially leak back through approximately 1000 feet of seal purge line and CRD hydraulic system piping, through the non-leak tested CRD pump discharge check valves, and to the outdoor Contaminated Condensate Storage Tanks and the vented tank level standpipe located within the Unit 2 turbine building. This is assuming no credit is taken for operation of any non-safety related equipment and no credit is taken for operator action. If a LOCA were to occur with severe core damage, coincident with this event, 0.4 gpm of primary system fluid could potentially bypass secondary containment and leak to the environment. In this unlikely event, the potential exists that the control room dose could exceed the GDC-19 limits, and the 0-2 hour Exclusion Area Boundary dose, considering an assumed instantaneous release, would increase but remain below the 10CFR 100 limits.

E. CORRECTIVE ACTIONS:

Engineering will revise procedure DAP 14-15 (Control of Transient and LOCA Analysis Parameters) as necessary to address parameters that could affect the core thermal power calculation (237-180-95-02101) and Engineering will revise DAP 10-02 (10CFR50.59 Review Screening and Safety Evaluations) as necessary to include reactivity questions contained in ComEd corporate procedure NEP 04-03 (10CFR50.59 Safety Evaluations)(237-180-95-02102). The revision to DAP 14-15 is currently in process. DAP 10-02 has been revised (Rev. 12) to include, in the Review Screening and Safety Evaluation forms, the reactivity questions and reference to DAP 14-15.

Engineering will evaluate a representative sample of pre-1986 modifications to determine if system interactions were properly evaluated and design requirements properly implemented (237-180-95-02103). The review of six representative pre-1996 modifications has been completed and the results are undergoing final internal review. (237-180-95-02103A) The reviews identified no outstanding technical issues in any of the modifications reviewed.

Dresden Unit 3 was administratively derated to a maximum power level of 2526 MWT and Dresden Unit 2 will be derated upon startup to 2526 MWT (237-180-95-02107). The derates will be controlled in accordance with station procedures and will remain in effect until corrective action 237-180-95-02105 is completed.

Engineering performed a review of the CRD system flow path on mechanical drawings and interviewed system engineers for a GE heat balance survey for the purpose of identifying other unaccounted flow paths. None were found. The GE heat balance survey was prepared for the BWR Owners Group utilities to assist in gathering and sharing pertinent information on this topic. Dresden is a participating member of the BWROG and Engineering will evaluate final industry recommendations regarding the heat balance survey and heat balance accuracy for implementation at Dresden (237-180-95-02104). This participation, coupled with previous Dresden initiatives related to Feedwater (temperature and flow), give assurance that the core thermal power calculation is being adequately addressed.

Nuclear Engineering will identify and modify computer programs and procedures necessary to permanently correct and document the recirculation pump seal purge flow in the core thermal power calculation programs and procedures (237-180-95-02105). Three of the four computer programs have been updated.

Design Engineering will obtain retrievable documentation for Recirculation Pump Seal Purge modifications M12-3-73-078 and M12-2-73-078 and perform an evaluation of the documentation to determine whether there are any design considerations (other than the change in flow to the CRD flow meter) that have not been addressed (237-180-95-02106). Re-review of these modifications has been completed. A discrepancy was found between the design and the commitments in UFSAR Section 3.8.2.1.9. This discrepancy is discussed in Sections B and D above.

Several actions were taken following this discovery. For Unit 3, an Operability Determination was issued, which included a change to Dresden General Abnormal procedure DGA-12 to require valves in the Control Rod Drive pump discharge in the Turbine Building to be closed post accident, following loss of the operating Control Rod Drive pump. This action would ensure that any back leakage is contained within the Control Rod Drive hydraulic system piping. For Unit 2, the seal purge lines were modified prior to startup to add two safety related check valves in each line in the vicinity of the primary Containment penetration. This plant change will also be made to Unit 3 (237-180-95-02107-S1). A UFSAR change has been generated to clarify the as-modified design of the seal purge lines and their associated Containment penetrations. In addition, Engineering will re-evaluate the concern identified in IE Notice 90-78 to confirm no other design or analyses issues exist. (237-180-95-02108-S1)

F. PREVIOUS OCCURRENCES:

The following is a non-reportable event which occurred at Dresden in 1992.

Event No.	
PIR 2-92-35	Process Computer Feedwater Flow Density Correction Factor
	A computer coding error related to Feedwater density correction was made during the change from the GE computer to the Honeywell 4500 process computer in 1983. The error resulted in a core thermal power calculation error of 0% to 3%. The coding error resulted in non-conservative calculation at non-rated conditions and no violations were discovered.

G. COMPONENT FAILURE DATA:

Not Applicable.