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 8

TITLE (4) Reactor Water Clean Up Pressure Control Valve PCV-1217 Configuration Outside Licensing Basis Requirements Due To Inadequate Modification Design

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
10	08	96	96	-- 016 --	00	11	06	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	100

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)		50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
	20.2203(a)(2)(v)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
J. Fox Ext. 2952	(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)

During engineering design review activities it was identified that the reactor water clean-up (RWCU) pressure control valve (PCV) 2(3)-1217 would not provide the necessary pressure drop as indicated by the Final Safety Analysis Report. The RWCU system was isolated and a Temporary Alteration was installed to mechanically limit the valve stroke such that the pressure drop in the failed open position would prevent overpressurization of downstream piping and components. The cause of the event was the failure to identify licensing basis requirements during the design of plant modifications. A design review will determine long term actions. The safety significance of this event was minimal.

PLANT AND SYSTEM IDENTIFICATION

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

Energy Industry Identification System (EIIS) codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommendation Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Reactor Water Clean Up Pressure Control Valve PCV-1217 configuration outside licensing basis requirements due to inadequate modification design.

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2(3) Event Date: 10/08/96 Event Time: 1215

Reactor Mode: N(N) Mode Name: Run(Run) Power Level: 100(82)%

Reactor Coolant System Pressure: 995(1000)psig

B. DESCRIPTION OF EVENT:

B.1 Sequence of events

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(ii)(B) which requires the reporting of any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant. On October 8, 1996, at 1215, it was determined that pressure reduction design features of the Reactor Water Clean Up (RWCU) [CE] system were inadequate. Design characteristics of Valve PCV 2(3)-1217, "Reactor Water Clean Up Pressure Control Valve", were inadequate for ensuring overpressure protection of the downstream piping with a lower design pressure. Notification of the event was performed pursuant to 10CFR72(b)(1)(ii)(B) at 1259 on October 8, 1996 through Emergency Notification System (ENS) number 31115.

On October 8, 1996 at 1215 the Dresden Shift Manager (Licensed Senior Reactor Operator) was notified that both the Unit 2 and Unit 3 PCV 2(3)-1217 did not satisfy licensing basis requirements. At 1342 the Unit 2 RWCU system was secured in accordance with Dresden Operating Procedure (DOP) 1200-03, "RWCU System Operation With The Reactor At Pressure." The Unit 3 RWCU system was secured in accordance with DOP 1200-3 at 1400. At 1259 through ENS call 31115 the NRC was notified of the event.

On October 8, 1996, at 2324, the Unit 2 RWCU system was returned to service and pressure control was transferred to the PCV bypass valve at 0405. The Unit 2 RWCU PCV was then isolated and taken out-of-service. The decision to return the RWCU system to service was made after a senior management review of a RWCU system transient that occurred on May 20, 1995 (Problem Investigation Report 249-200-95-04800). During this transient, relief valve RV 3-1201-180 lifted and prevented the pressure downstream of PCV 3-1217 from exceeding 150 psig even though the demand on PCV 3-1217 was at 95%.

It was concluded that since RV 3-1201-180 had successfully protected against overpressure during this event, no significant safety risk would be introduced by returning the system to service for a short time. Similarly, at 0207 on October 9, 1996, the Unit 3 RWCU system was returned to service and system pressure control was transferred to the PCV bypass valve at 1000 the same day. The Unit 3 RWCU PCV was then isolated and taken out-of-service.

Activities then commenced to install a temporary alteration on both Unit 2 (Temporary Alteration (TA) II-30-96) and Unit 3's (TA III-51-96) PCV to limit its stroke to less than sixty percent. Temporary alterations II-30-96 and III-51-96 were developed for PCV 2-1217 and PCV 3-1217 to positively limit the stroke of these valves using a mechanical stop. The mechanical stops were sized by calculation DRE96-0195 to ensure that if the valves failed open, a pressure drop of at least 950 psi would result at a flowrate of 1300 gpm. This ensures low pressure piping downstream of PCV 2(3)-1217 will not exceed its design pressure should the limiting failure occur concurrent with a failure of the high pressure isolation instrumentation. This is consistent with the response to Topic V-II.A entitled "Requirements for Isolation of High- and Low-Pressure Systems" of the NRC Systematic Evaluation Program (SEP) Final Report dated February 1983 and the basis for the acceptability of the RWCU high pressure/low pressure interface. At 1554 on October 9, 1996, Temporary Alteration II-30-96 was installed. At 2106 on October 9, 1996, Temporary Alteration III-51-96 was installed.

B.2 Design Requirements

Updated Safety Analysis Report

Dresden's Updated Final Safety Analysis Report (UFSAR) Section 5.4.8.2 "System Description" for the RWCU system contains the following information:

> "Downstream of the heat exchangers, the reactor water enters a pressure reducing valve (PRV) which controls system pressure downstream to a maximum of 150 psig. The PRV is a drag-type valve with a design pressure drop of 950 psi in the full open position. Therefore, if the valve failed open, reactor pressure must be greater than 1100 psig before downstream pressure would exceed the 150 psig design pressure for the piping. The downstream piping is further protected against overpressurization by relief valves as described in Section 5.4.8.3."

The nomenclature used for "pressure reducing valve (PRV)" described in the UFSAR is referring to PCV 2(3)-1217 as discussed in this Licensee Event Report.

Integrated Plant Safety Assessment (IPSA) Systematic Evaluation Program (SEP)

The pressure drop of PCV 2(3)-1217 is discussed in the UFSAR because it was evaluated under Topic V-II.A, "Requirements for Isolation of High- and Low-Pressure Systems", during the IPSA SEP in 1982 (NUREG-0823). The RWCU system did not satisfy the current licensing requirements for systems that had a direct interface with the reactor coolant system. The NRC concern focused on a malfunction of the PCV that causes it to go full open concurrent with a failure of the high pressure RWCU isolation instrumentation downstream of the PCV.

The high pressure isolation did not meet the NRC's criteria (Standard Review Plan Section 7.6 and BTP ICSB-3) since a single pressure switch was used to initiate the isolation of both the inboard and outboard containment isolation valves.

Contrary to the SEP response and description in the UFSAR, the pressure drop through PCV 2(3)-1217 in the full open position would be less than 950 psid at a flowrate of 1300 gpm. This could result in the overpressurization of downstream RWCU piping and components since the pressure relief valves on the low pressure piping (design pressure of 150 psig) have a rated capacity of 1300 gpm when the inlet pressure is 150 psig. Because PCV 2(3)-1217 would provide less restriction to flow if it failed open, the following conditions would change from those previously assumed in the SEP evaluation:

1. Higher flowrate through PCV 2(3)-1217
2. Higher pressure downstream of PCV 2(3)-1217 which is also the inlet pressure to the pressure relief valves
3. Higher discharge flowrate through the relief valves due to a higher inlet pressure

The flowrate through PCV 2(3)-1217 and the relief valves will be in equilibrium. Although the pressure downstream of PCV 2(3)-1217 was not calculated, it is believed to be significantly higher than 150 psig which is the design pressure of the piping and components.

The SEP information supplied to the NRC during the SEP in 1982 errantly indicated that PCV 2-1217 had a design pressure drop of 950 psid in the full open position.

B.3 Configuration Management Activities on PCV 2(3)-1217

According to the original GE design specification (21A5540) for PCV 2(3)-1217, PCV 2(3)-1217 was provided with a restricting orifice (RO) 1218. The specification indicates that during normal operation with upstream pressure at 1000 psig and a flowrate of 1260 gpm, the pressure drop would be 450 psi through PCV 2(3)-1217 and 450 psi through RO-1218. The specification also indicated that during shutdown operation with PCV 2(3)-1217 fully open, the pressure drop through PCV 2(3)-1217 would be only 50 psi at a flowrate of 1260 gpm. Therefore, with PCV 2(3)-1217 fully open, the total pressure drop through PCV 2(3)-1217 and RO-1218 would be 500 psi at a flowrate of 1260 gpm.

PCV 2(3)-1217 1978 Modification

PCV 2(3)-1217 was replaced in 1978 with a self drag type valve (CCI Model No. M2A5-X3-X4P6-X4P6) via plant modifications M12-2-77-029 (Unit 2) and M12-3-77-029 (Unit 3). Restricting orifice RO-1218 was also apparently removed at this time. The valve drawing for the new valve indicated a pressure drop of 950 psi within the "Nameplate Data" section of the drawing.

However, 950 psi appears to be the pressure drop during normal operating conditions and not the pressure drop when the valve has failed wide open. The valve drawing states that the valve flow coefficient at 100% stroke is 60. The relationship between pressure drop, flowrate, and flow coefficient with water as the fluid is given by the following formula:

Pressure drop across the valve (DP)= ((Flowrate (Q)in gpm)**2)/((Valve Flow Coefficient)**2)

At a flowrate of 1300 gpm and a valve flow coefficient of 60, DP = 469 psi. Therefore, it can be calculated that the pressure drop through PCV 2(3)-1217 was less than the 950 psi at 1300 gpm as stated in the SEP response for Topic V.II-A.

PCV 2(3)-1217 1992 Modification

The trim of PCV 2(3)-1217 was again modified by plant changes P12-2-92-694 (Operation Authorized on 6/25/93 for Unit 2) and P12-3-93-613 (Operations Authorized on 7/17/94 for Unit 3). The valve drawing states the valve flow coefficient at 100 percent stroke is 92. At a flowrate of 1300 gpm and a valve flow coefficient of 92, DP = 200 psi. Again, the pressure drop through PCV 2(3)-1217 would be significantly less than the 950 psi at 1300 gpm as stated in the SEP response. The licensing basis requirement to ensure overpressure protection of low pressure piping downstream of the PCV, assuming the PCV fails open concurrent with a failure of the high pressure isolation instrumentation, was not identified during the design of the plant design changes.

Because the pressure drop through PCV 2(3)-1217 would be significantly less than 950 psi in the full open position at a flowrate of 1300 gpm, Engineering personnel concluded that the pressure downstream of PCV 2(3)-1217 could exceed that of the piping design pressure of 150 psig. This could possibly result in a loss of reactor coolant outside of primary containment. This scenario also assumes a failure of the high pressure isolation instrumentation for the RWCU system.

C. CAUSE OF EVENT:

The cause of this event was inadequate configuration management (NRC Cause Code B; Design error) in that there was a failure to ensure design and licensing basis requirements were met during the design of plant changes. When the new trim for PCV 2(3)-1217 was being designed in 1992 and 1993, engineering personnel failed to identify the licensing basis requirement that the pressure drop through PCV 2(3)-1217 must be 950 psid at 1300 gpm to ensure overpressure protection of downstream components.

D. SAFETY ANALYSIS:

The most limiting condition would be if PCV 2(3)-1217 had failed open concurrent with a failure of the high pressure isolation instrumentation. Under this condition, the design pressure of piping and components may have been exceeded. This may have lead to the leakage and/or rupture of downstream piping and components.

Operator action would have promptly limited the consequences of this postulated event. A high temperature alarm exists in the discharge piping of relief valve PRV-1201-180. The annunciator procedure DAN 902(3)-4 H-13 for this alarm instructs the operator to check for a malfunction of PCV 2(3)-1217 and would lead to isolation of the system.

Recently performed calculations performed for breaks in the high energy portion of the RWCU system lasting for up to 10 minutes indicate the offsite dose will be significantly less than 10% of 10 CFR Part 100 limits. The offsite dose consequences from a break in the piping downstream of the PCV will be even less since the temperature of the water will be less than 150 degrees F and fewer radioactive particles will become airborne. Therefore, the safety significance of this event was minimal.

E. CORRECTIVE ACTIONS:

E.1 Temporary Alterations II-30-96 and III-51-96 were installed.

E.2 A comprehensive overpressure protection review will be performed for the RWCU system which will take into account the pressure drop from piping and components upstream of RWCU PCV 2(3)-1217. It is also expected that two-phase flow conditions will develop upstream of RWCU PCV 2(3)-1217 under the postulated scenario where RWCU PCV 2(3)-1217 fails open. This will further increase the pressure drop prior to RWCU PCV 2(3)-1217. Regulatory Assurance will be involved in the review and approval of the final report. This will ensure that all required licensing actions, such as a resubmittal of the SEP response, are identified. The review will be documented in the Nuclear Tracking System. (2371809601601)

E.3 Based on the comprehensive review, appropriate permanent changes will be made to ensure that piping and components downstream of RWCU PCV 2(3)-1217 are protected from overpressure assuming RWCU PCV 2(3)-1217 fails open and the high pressure isolation instrumentation fails. (2371809601602)

E.4 The UFSAR and RWCU Design Basis Document will be updated as appropriate based on the comprehensive review. (2371809601603)

E.5 Appropriate licensing submittals will be performed based on the comprehensive review. (2371809601604)

E.6 Plant management has stressed to Dresden's engineering staff the importance of knowing and complying with design and licensing basis requirements as described in the UFSAR. Dresden's rebaselining of the UFSAR has improved the accessibility of design and licensing basis information, such as that discussed in the SEP, to ensure that these requirements are identified during the design of plant changes.

E.6 Design Basis Documents (DBD) have also been prepared for key plant systems. Validation of DBD content and FSAR information against the actual plant configuration has already been performed for several systems.

E.7 Since the design of the replacement trim for PCV-1217 in 1992 and 1993, Dresden Station has added a Design Engineering group. The primary mission of this group is to ensure the design and licensing basis of the plant is maintained during plant design changes. This group is staffed with experienced design engineers from the nuclear industry from both within ComEd and consulting engineering firms. The experience of the engineers in this group and the mission assigned to them are two additional measures that will ensure design and licensing basis requirements are met during the design of plant changes.

F. PREVIOUS OCCURRENCES:

Previous occurrences of errors associated with design change control include:

LER/Docket Number	Title
96-003/05000237	Main Control Room HVAC Outside of Design Basis Due to Inadequate Implementation of Modification.

Dresden Station emergency procedure DGA-12 did not allow the Control Room Heating Ventilation and Air Conditioning (HVAC) Emergency Air Filtration Unit system to operate and pressurize the Control Room as required in the design bases and Updated Final Safety Analysis Report (UFSAR). DGA-12 failed to provide the Control Room Operator action needed to reestablish pressurization. Immediate corrective actions included revision of the emergency operating procedure to give guidance on restoration of the Control Room HVAC in the pressurization mode. The root cause was personnel error during implementation of a Control Room HVAC modification.

95-19/05000237 — The Control Rod Drive Scram Discharge Volume's Reactor Protection System Control Logic Fails To Meet the Single Failure Criteria Due to Design Deficiency

The Control Rod Drive Scram Discharge Volume's (SDV) control logic did not meet the single failure criterion. The SDV was declared inoperable and an ENS phone call was made. The root cause of the failure was due to an inadequate design review process and inattention to detail during the SDV modification development. Corrective actions included a review of the new modification practices to assure changes made in 1986 are still in place, and a sampling review of other modifications developed and reviewed by the cognizant engineers who developed the SDV modification containing the error.

95-11/05000237 — Unit 2 and Unit 3 Nitrogen Make-up Flow Found Not to Meet Technical Specifications Due to Not Clearly Establishing the Design of the Nitrogen Make-up System.

The maximum normal make-up nitrogen flow for Units 2 and 3 was less than that implied by the Technical Specifications because of material problems with each Unit's Pressure Control Valve (PCV) (2(3)-8527). The cause was due to not clearly establishing the design of the normal nitrogen make-up paths. The modification program has since been upgraded and system walk-downs are being performed, in part, to identify any other modification problems. This event had minimal effect on plant or public safety.

95-6/05000237 — TIP System Isolation Does Not Have 'Seal In' Logic On Group II Isolation

The Primary Containment Isolation System (PCIS) logic associated with the Traversing Incore Probe (TIP)system allowed a TIP ball valve to reopen without operator action after initiation and subsequent reset of a group II PCIS signal.

This condition is contrary to information contained in section 7.3.2.4 of the Updated Final Safety Analysis Report (UFSAR) which states that logic for these valves will not allow valves to automatically reopen after the isolation signal is reset. Thus, the NRC was voluntarily notified of a potential degraded/unanalyzed condition pursuant to 10CFR50.72(b)(1)(ii). Actions were taken to modify the design to preclude automatic reopening of the TIP ball valve following reset of a Group II isolation signal.

95-1/05000237 Inoperable Control Room HVAC Booster Fans, due to improperly sized thermal overload heater devices.

During a Dresden Operating Surveillance (DOS) 5750-01 on the Control Room heating, ventilation, and air conditioning (HVAC) system, the booster fan A tripped. Engineering evaluation was performed and stated that the thermal overload (TOL) devices for the booster fans of both A and B of the Control Room HVAC system were set at a level that would not prevent spurious trips during normal plant conditions (including degraded voltage conditions). The booster fan A TOL device was replaced at 2128 hours on January 7, 1995, and the Control Room HVAC System was declared operable at that time. The booster fan B TOL device was replaced at 2233 hours on January 10, 1995. The safety significance of this event is considered minimal because a means was readily available for the operator to manually restart the booster fans within a reasonable time period, and because only one of the fans would be needed at a time to support the operation of the Control Room HVAC System.

95-022/05000249 CRD SCRAM Discharge Volume Galleries Do Not Meet UFSAR Allowables Due to a Design Deficiency

The Unit 3 East and West Bank Control Rod Drive (CRD) SCRAM Discharge Volume (SDV) gallery platforms did not meet the design allowable stresses specified in the UFSAR. Engineering evaluation determined that discrepancies existed between the design drawings and the as-installed configuration of the galleries and also found modelling and design discrepancies in the design basis calculations of the galleries. An operability evaluation determined that sufficient margin existed in the CRD SDV gallery steel to maintain operability. The root cause of this event is a design configuration and analysis deficiency in that the procedures and controls in place at the time of original plant construction and at the time when the CRD SDV galleries were modified were inadequate. Corrective actions included reanalyzing the steel to remove unnecessary conservatism and adding required reinforcements to return stresses in the Unit 3 CRD SDV galleries to within UFSAR limits and determining if similar discrepancies exist in the Unit 2 CRD SDV galleries.

G. COMPONENT FAILURE DATA:

There were no component failures associated with this event.