Commonwealth Edison
Dresden Nuclear Power Station
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August 22, 1994

RLBLTR 94-0001

U.S. Nuclear Regulatory Commission
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Washington, D. C. 20555

Licensee Event Report 94-023, Docket 50-237 is being submitted as required by Technical Specification 6.6, NUREG 1022 and 10CFR50.73(a)(2)(ii).

Sincerely,

Richard L. Bax
Unit 3 Station Manager
Dresden Station

RLB/LJ:cfq

Enclosure

cc: J. Martin, Regional Administrator, Region III
NRC Resident Inspector's Office
File/NRC
File/Numerical

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

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Units 2 and 3	05000237	1 OF 4

TITLE (4)
ASME Code Allowable Stresses Exceeded on the Control Rod Drive Flange Cap Screws Due to Lack of Engineering Review

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
07	29	94	94	-- 023 --	00	08	19	94	Dresden Unit 3	05000249
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): N
POWER LEVEL (10): 091 (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)		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)
Lance E. Jacobsen, CRD System Engineer Ext. 2363	(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)

At 2130 on July 29, 1994, with Unit 2 at 91% rated power and Unit 3 in a Refuel Outage, Site Engineering personnel determined that ASME code allowable stresses were exceeded on the Control Rod Drive (CRD)[AA] flange cap screws on CRD H-7 on Unit 2 and CRD A-6 on Unit 3. These CRDs were over-torqued to 550 ft-lbs per Dresden Maintenance Procedure (DMP) 300-9 "Control Rod Drive Removal and Replacement" to reduce or eliminate minor o-ring leakage at the CRD flange. This exceeded the vendor specified torque of 350 +\- 25 ft-lbs.

The affected bolts on the Unit 3 CRD were replaced prior to startup and the affected bolts on Unit 2 were determined to be operable and will be replaced during D2F23. The cause of this event was inadequate engineering review in 1981 when calculations were incorrectly performed by maintenance management personnel allowing the CRD cap screws to be torqued up to 550 ft-lbs. The safety significance of this event was minimal because the failure of all eight cap screws on a CRD flange is bounded by Dresden's LOCA analysis and even though code allowables were exceeded, stress values were well below yield.

EVENT IDENTIFICATION:

ASME Code Allowable Stresses Exceeded on the Control Rod Drive Flange Cap Screws Due to Lack of Engineering Review

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 2 (3) Event Date: 07/29/94 Event Time: 2130

Reactor Mode: N (N) Mode Name: Run (Refuel) Power Level: 91% (0%)

Reactor Coolant System Pressure: 1000 psig (0 psig)

B. DESCRIPTION OF EVENT:

At 2130 on July 29, 1994, with Unit 2 at 91% rated power and Unit 3 in a Refuel Outage, Site Engineering personnel determined that ASME code allowable stresses were exceeded on the Control Rod Drive (CRD)[AA] flange cap screws on CRD H-7 on Unit 2 and CRD A-6 on Unit 3. Each CRD is secured to its housing flange by eight 1-8UNC x 5.5 cap screws made of high tensile strength AISI4140 steel. These CRD cap screws were over-torqued to 550 ft-lbs per Dresden Maintenance Procedure (DMP) 300-9 "Control Rod Drive Removal and Replacement" to reduce or eliminate minor o-ring leakage at the CRD flange. CRD H-7 on Unit 2 was over-torqued in December 1990 and CRD A-6 on Unit 3 was over-torqued to 500ft-lbs in June 1986 and up to 550 ft-lbs in August 1994.

Site Engineering determined that a torque of 550 ft-lbs applied to a newly lubricated joint results in an average stress of approximately 75 ksi on each of the eight CRD cap screws that connect the CRD to the housing flange. Although this stress was above the ASME code allowable of 64.4 ksi, the applied stress was well below (approximately 22%) the material yield strength of 96.6 ksi. This margin was sufficient to accommodate a greater than 5% variation in applied torque. The engineering evaluation as documented in ENC-QE-40.1 dated July 30, 1994 (Ref. Chron# 0302880 and amended by Chron# 0302900) determined that the affected CRD flange bolts on CRD H-7 on Unit 2 were operable. In addition, the affected flange bolts on CRD A-6 on Unit 3 were replaced per work request D26830 prior to start-up from D3R13.

Additionally, immediate corrective actions included a maintenance history review to determine if any other CRD's on either unit were affected. Only five CRDs currently installed on Unit 3 were determined to have been affected and were also over-torqued in August of 1994. However, these CRDs (B-9, B-10, L-6, M-10, and N-12) were conservatively determined to be stressed to approximately 37% below yield and approximately 5% below code allowables per the CRD Flange Bolt Over Torque Analysis Technical Audit dated August 5, 1994 (Ref. Chron. 0302900)because of the conditions under which they were over torqued. Licensed operators were also notified of the potential warning signs related to CRD failure.

C. CAUSE OF EVENT:

This LER is submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B), which requires the reporting of an event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being in a condition that was outside the design basis of the plant.

The cause of this event was a lack of engineering review of design calculations performed in 1981. The increased torque value of 550 ft-lbs was derived from an in-house maintenance department calculation. The General Electric recommended torque value for these bolts was 350 +/- 25 ft-lbs. The in-house calculation failed to include differential thermal expansion of the stainless steel flange vs. the carbon steel bolts. In addition, increasing torque is not an appropriate means of correcting leakage due to the nature of the flange/o-ring connection. The individual who performed this calculation was unaware of the effects of differential expansion on the bolts as well as on the mechanics of the o-ring joint used in this application. Apparently, no second review of these calculations was performed nor was the vendor contacted for approval to vary from vendor torque specification.

The Chronology of this change is as follows:

1) May 4, 1981 - Inadequate calculations performed by in-house maintenance personnel allowing torque on CRD Cap Screws to be increased beyond vendor specifications.

2) June 24, 1986 - Temporary procedure change (TPC 86-6-385) approved on DMP 300-09 to allow over torquing of CRD cap screws up to 550 ft-lbs. The 50.59 review stated that bolting stress was inside design code allowances, however, it did not adequately document the basis for this conclusion. The 50.59 process at Dresden Station has been significantly improved since 1986.

3) November 5,1987 - The previously approved TPC was incorporated into a permanent procedure revision.

4) March 18, 1994 - TPC 88-3-169 was approved to provide administrative controls on the bolts that were over-torqued. The change required that all bolts that are over-torqued be tagged and discarded when they are removed for future maintenance. This TPC was later incorporated into a permanent procedure revision.

D. SAFETY ANALYSIS:

Although the ASME Code allowable stress was exceeded, safe reactor operation was not compromised based on the following:

1) The affected bolts were well below the material yield strength.

2) Experience has shown that there has never been any cap screw total failures since the use of 550 ft-lbs was implemented at Dresden Station.

3) The probability of multiple cap screw total failures at any one time is deemed extremely low. The crack growth previously observed in this type of cap screw (Ref. SIL No. 483 "CRD Cap Screw Crack Indications") has been either arrested or very slow. If any such failures were to occur, however, it would be preceded by leakage at the flange joint. The leakage would be detected by leakage monitoring systems.

4) In the unlikely event that failure of all eight cap screws were to occur, the CRD would separate from the housing. The CRD support structure under the reactor vessel would allow the CRD to drop less than one inch. The subsequent control rod movement is substantially limited below one drive "notch" movement (6 inches). Sudden withdrawal of any control rod through a distance of less than one drive notch at any position in the core does not produce a transient that will cause fuel damage.

5) In the worst case scenario (i.e. total flange separation), the total leakage through the flange joint is approximately 840 gpm. This leakage condition is well within the ECCS makeup flow capacity and it is bounded by the LOCA analysis.

6) The flange separation would immobilize the CRD. Failure to scram of a single control rod has been previously analyzed and found acceptable.

Therefore, the safety significance of this event is considered to be minimal.

E. CORRECTIVE ACTIONS:

DMP 300-09 will be revised by Mechanical Maintenance to eliminate the allowance to increase torque beyond 375 ft-lbs. A procedure inquiry was submitted on August 2, 1994 to implement this change. This change will include provisions to replace the bolts on the remaining five CRDs on Unit 3 (B-9, B-10, L-6, M-10, and N-12) which were over-torqued but did not exceed ASME code allowable stresses the next time the CRDs are pulled for rebuild or to replace o-rings. The DMP will also be changed to specify the proper lubricant to be used when installing the bolts.

The bolts on CRD H-7 on U-2 will be replaced prior to start up from D2F23.

This LER will be tailgated with the Maintenance Staff, Site Engineering, and System engineering personnel to express the importance of independently reviewing design calculations as well as the importance of contacting vendors when their recommendations are not going to be followed. In addition, this tailgate will discuss the inappropriateness of overcompressing o-ring joints.

F. PREVIOUS OCCURRENCES:

Not applicable.

G. COMPONENT FAILURE DATA:

Not applicable.