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)

Reactor Head Closure Stud 61-198-047 Outside FSAR allowables for Material Toughness Due to Unknown Cause

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)
0 1	1 6	9 1	9 1	---	0 0 2	---	0 1	0 4	1 6	9 3	Dresden Unit 3	0 5 0 0 0 2 4 9
											N/A	

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIRMENTS OF 10CFR (Check one or more of the following) (11)

POWER LEVEL (10)							
0 0 0		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 in Abstract below and in Text)
		20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii) (A)		
		20.405(a)(1)(iv)	X	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)		

LICENSE CONTACT FOR THIS LER (12)

NAME		TELEPHONE NUMBER	
		AREA CODE	
Gerald Whitman, Technical Staff ISI Coordinator	Ext.2351	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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	B D	R C T	G 0 8 0	Y					

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 January 24, 1989, while Unit 2 was in a scheduled refuel outage, ultrasonic examination revealed cracks in the lower threaded portion of two reactor head closure studs. One stud was sectioned and sent out for chemical and metallurgical evaluation. The evaluation results received in January 1991, while Unit 2 was shutdown for a subsequent refuel outage, indicated that the material toughness of the stud did not meet FSAR requirements.

The cause of the cracking is stress-corrosion cracking (SCC) which initiated in the base of pits located in the thread roots. Exposure of the studs to oxygenated water during refueling outages is thought to be a contributing factor. The cause for the reduction in material toughness has been determined to be a mechanism known as 500° F tempered martensite embrittlement.

Corrective actions for this event included replacement of the two cracked studs and the sectioning of both studs for chemical and metallurgical evaluation. Additionally, a sample of ten studs was examined during the D2R12 Refuel Outage using an enhanced ultrasonic technique. Future corrective actions will include an augmented inspection schedule for closure studs and exploring methods to protect the studs from exposure to water. A structural evaluation of the closure studs has shown that this condition would have minimal effect on proper head closure. This is the first occurrence of cracks being found in reactor head closure studs.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

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: January 16, 1991 Event Time: 1400 Hours

Reactor Mode: N Mode Name: Refuel Power Level: 0%

Reactor Coolant System (RCS) Pressure: 0 psig

B. DESCRIPTION OF EVENT:

At 1400 hours on January 24, 1989, with Unit 2 in its cycle 11 Refuel outage, Ultrasonic Examination (UT) revealed a crack indication in reactor head closure [BD] stud 61-198-47. Because of this indication, the sample was expanded to include 100% of the reactor head closure studs. The expanded sample revealed a crack indication in one additional stud, 61-198-70. Both of the studs were replaced under Work Request 91724.

Subsequent magnetic particle testing on the removed studs verified that the cracks were present. Additional ultrasonic testing using a custom ordered Internal Diameter (ID) probe estimated the maximum crack depths to be 35% through wall (0.88") for stud 61-198-47, and 83% through wall (2.09") for stud 61-198-70. In an effort to determine the cause of the cracking, stud 61-198-47 was sectioned to allow for metallurgical evaluation. Stud 61-198-70 was retained at Dresden Station for use as a calibration standard in future ultrasonic examinations.

The reactor closure studs measure approximately 65 inches long and 5.75 inches in diameter, with a 1 inch diameter bore hole. A 27 inch long section containing the lower threaded region of stud 61-198-47 and 12 inches of the shank was removed for metallurgical examination. A failure analysis was performed at Argonne National Laboratories under the direction of System Material Analysis Department (SMAD) Metallurgy Group personnel. The stud material is specified as ASTM A320-GR.L43, a quench and tempered low alloy steel that is similar to AISI Grade 4340.

The following section contains the results of the chemical and metallurgical evaluations of the sectioned stud. The report outlining these results was transmitted to the Station on January 11, 1991, while Unit 2 was shutdown for its cycle 12 refuel outage.

A visual inspection of the stud section revealed pitting and corrosion of the outside diameter (OD) surface. The attack was most severe in the 12 threads located nearest to the shank. Apparently, these threads were not engaged with the reactor vessel flange bushing during service. Magnetic particle testing revealed cracking in the 7 thread roots located from 14 to 20 threads below the shank transition. The cracking was intermittent and extended around approximately 40% of the stud circumference. In several of the cracked threads, pits appeared to be aligned in the thread root.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

Axial sectioning was performed at two locations that had been identified as sections 1 and 2 by the non-destructive examination testers. The ID bore probe had reported maximum crack depths of 0.40" at section 1 and 0.88" at section 2. Note that section 2 was reported to contain the deepest crack in the stud.

Metallographic examination of sections removed from the identified areas indicated that the cracks initiated at pits in the thread roots. Cracks were observed in adjacent threads and oriented primarily at an angle between 65 and 75 degrees to the stud axis. The maximum crack depth, as measured to the nearest thread root, was 0.40" in section 1 and 0.70" in section 2. Crack propagation occurred in a branching manner, particularly near the crack tips. Etching revealed the microstructure consisted of tempered martensite with a prior austenite grain size of ASTM 8. This is considered normal for the specified Grade L43 material. Because of the small grain size and the fact that most of the cracks were oxidized, it was very difficult to determine if the crack propagation mode was intergranular or transgranular. Metallographic examinations of several of the secondary cracks suggested a mixed propagation mode.

Scanning electron microscope examinations were performed on the section 2 sample after opening up the largest crack. Energy dispersive X-ray analyses of the surface did not identify a corroding species. In addition, because the surface was covered by a thick, black oxide, little information could be gained from the uncleaned surface. After cleaning the surface with an inhibited acid solution, some evidence of intergranular attack could be observed.

Chemical and mechanical tests were performed on material removed from the shank region of the stud. Chemical analyses, tensile, hardness and Charpy V-notch impact testing results are shown in Tables 1, 2 and 3 along with the data reported on the original Certified Material Test Report (CMTR). The chemical analyses correlate well with the original CMTR. However, the tensile test results are 10-20 ksi higher than reported on the CMTR. In addition, the Charpy impact data indicates that a significant reduction in toughness was observed from that reported on the CMTR. Appendix D, Paragraph 10.10 of the FSAR requires closure studs to meet the impact test requirements of the ASME Code, Section III, Paragraph N-330 at a temperature no higher than 10°F. The 1963 Edition through the summer of 1964 Addenda of the ASME Code, Section III requires closure studs to have a minimum impact toughness of 30 foot-pounds for any one individual specimen and 35 foot-pounds for the average of three specimens. As can be seen in Table 3, stud 61-198-47 did not meet these values at the required temperature of 10°F. It was therefore determined that this event met 10 CFR 50.73 reporting criteria.

Table 1 - Chemical Analysis Results (Wt. %)

Element	ASTM A320 Gr. L43	Stud 61-198-47 Near OD	Stud 61-198-47 Near Bore	CMTR For Heat 67-80278
Carbon	0.38-0.43	0.43	0.43	0.43
Manganese	0.60-0.85	0.72	0.67	0.72
Phosphorus	0.035 max.	0.010	0.010	0.010
Sulfur	0.040 max.	0.017	0.012	0.014
Silicon	0.15-0.35	0.30	0.29	0.29
Nickel	1.65-2.00	1.68	1.67	1.75
Chromium	0.70-0.90	0.80	0.74	0.80
Molybdenum	0.20-0.30	0.26	0.23	0.26

Table 2 - Hardness and Tensile Test Results

	Tensile Strength (ksi)	Yield Strength (ksi)	% Elongation	% Reduction in Area	Rockwell C Hardness
A320, Gr.L43	125 min.	105 min.	16 min.	50 min.	No Requirement
Stud 61-198-47					
Near OD	180.2	167.8	17.4	53.5	38/39
1/2 Radius	173.0	155.2	18.0	56.5	34/35
Near Bore	164.0	145.7	17.4	54	32/33
Heat 67-80278 CMTR at 1/2 Radius					
Test 1	156.5	140.0	19.0	59.1	32/36
Test 2	160.0	145.0	18.5	56.9	36/38
Test 3	154.0	137.5	18.5	57.3	31/33

Table 3 - Charpy V-Notch Impact Test Results (ft. - lbs.)

	10°	Room Temperature	80°F	150°F
Heat 67-80278 CMTR at 1/2 Radius	47,52,36	---	---	---
Stud 61-198-47				
Near OD	22,18	31,32	39,31	47,47
1/2 Radius	21,20	22,25	28,27	47,46
Near Bore	20,20	25,23	22,26	44,46

C. **APPARENT CAUSE OF EVENT:**

This report is being submitted in accordance with 10CFR50.73(a)(2)(ii)(B), which requires the reporting of any condition outside the design basis, as the observed toughness of the tested stud did not meet the ASME code requirements specified in the FSAR.

The apparent cause of the cracking was the result of stress-corrosion cracking (SCC) which initiated in the base of the pits located in the thread roots. The fact that the cracks propagated in a branching manner and were located in consecutive threads suggests an SCC mechanism.

For SCC to occur, a tensile stress must be applied to a susceptible material in a corrosive environment. The environment that contributed to the SCC is thought to be exposure to oxygenated water during unit refueling outages. The bushings in which the studs are engaged are slightly lower than the vessel flange sealing surface. This creates a small reservoir where water can accumulate. When the head is re-tensioned, the studs are in tension for up to two weeks prior to the vessel heat-up, when the water would be driven away. This could allow SCC to occur.

Metalurgical evaluation suggests that the cause of the reduction in material toughness and the increase in tensile strength from that reported on the CMTR, is a mechanism known as 500°F tempered martensite embrittlement. This mechanism is caused by the transformation of retained austenite into long thin interlath carbides and untempered martensite, and results in an overall reduction in the toughness of the material. The transformations occur due to the thermal and mechanical instability of retained austenite at temperatures below the transformation temperature. Laboratory research indicates that the reduction in material toughness will be limited to the amount of retained austenite present in the material. Transmission electron microscopy and x-ray diffraction analyses have shown that no retained austenite remains in the two removed unit 2 closure studs. This suggests that no further reduction in material toughness will occur in the remaining closure studs.

D. SAFETY ANALYSIS OF EVENT:

A structural evaluation performed by G.E. Nuclear Energy has shown that there is significant structural margin in the design of the reactor pressure vessel (RPV) closure and that the RPV could still meet ASME code requirements with some cracked studs at the observed toughness level. The required code margins could still be maintained even if 13 studs were completely cracked, provided they were evenly distributed around the flange. Also, up to 43 studs could be cracked to a depth of 1.3", in any distribution around the flange and code margins would still be met.

The most limiting condition for the vessel stud from the fracture mechanics viewpoint is the bolt up condition. The temperature for bolt up can be as low as 100°F and the loading is essentially the maximum value corresponding to stud tensioning. Other conditions such as hydrotest and normal operation are not as severe as bolt up since the temperature (and therefore, the toughness) is significantly higher, but the loading on the stud is essentially the same. Thus the vessel bolt up represents a 'proof test' and after a successful bolt up, the likelihood of a fracture problem in the operating condition is negligible.

The two studs found cracked during the D2R11 Refueling Outage were both bolted up prior to the cycle eleven start-up and de-tensioned during the D2R11 Refuel Outage without failing. Additionally, no other studs were found to be cracked, and the two with crack indications were replaced. Finally, the fact that there was no remaining retained austenite in the samples taken from the two cracked studs suggests that no further mechanism exists for 500°F tempered martensite embrittlement to occur. Consequently, no further reductions in overall material toughness of the remaining studs will occur.

Based on the above, the safety significance of this event is considered to be minimal.

E. CORRECTIVE ACTIONS:

Reactor head closure studs 61-198-47 and 61-198-70 were replaced prior to start-up from the D2R11 refueling outage. Additionally, actions were taken to determine the root cause of the cracking as described in sections 'B' and 'C' of this report.

A sample of ten closure studs was examined using an enhanced ultrasonic technique towards the end of the D2R12 refueling outage. No crack indications were found. The exam was limited to ten studs because the metallurgical and chemical test results of the cracked stud were received by the Station late in the outage. By this time the reactor head was tensioned, the insulation and drywell head were in place and the shield plugs were installed. The examinations were performed from the drywell, through two access ports in the insulation. Only ten studs were accessible.

A sample of 16 non-cattle chute studs were removed from unit 3 during the D3R12 refueling outage and examined by the magnetic particle method (NTS 237-180-91-01403). No evidence of surface cracking was detected. Although the original commitment was to remove a sample of 15 non-cattle chute studs each refueling outage and examine them by the magnetic particle method (NTS 237-180-91-01404), only 14 studs were removed and examined during the D2R13 refueling outage. No evidence of surface cracking was detected on these studs. The decision to limit the sample size to 14 was based upon the difficulties experienced while attempting to remove the studs from the flange, and is justified by the results of the structural margin evaluation described below.

Future actions will include an examination of 100% (92) of the closure studs on both units 2 and 3 each refueling outage, using the enhanced ultrasonic end shot technique. This technique is capable of detecting a 0.3 inch deep crack. A structural margin evaluation performed by G.E. Nuclear Energy conservatively determined the acceptable crack depth for the Dresden closure studs to be 0.57 inches, with a critical crack depth of 1.36 inches. The sensitivity of the end shot UT exam performed each outage will assure that adequate safety margins are maintained. The ISI Coordinator will submit surveillance tracking items for this activity (NTS 237-180-91-01401S1).

Methods to protect the studs from exposure to water are also being explored. These methods include the use of calrods to dry the studs prior to tensioning and the use of water tight stud covers or coatings (NTS 237-180-91-01405).

All other planned corrective actions identified in the original LER and tracked under (NTS 237-180-91-01401, 01402, 01403, 01404) are now considered closed.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of reactor head closure stud cracking at Dresden Station.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	Mfg. Part Number
N/A	Reactor Head Closure Stud	ASTM A320-GR-L43	N/A

An industry wide NPRDS data base search revealed no other reports of this nature.