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ACCESSION NBR:8903210130 DOC.DATE: 89/03/13 NOTARIZED: NO DOCKET #
FACIL:50-261 H.B. Robinson Plant, Unit 2, Carolina Power & Light C 05000261
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 89-002-00:on 890211,failure of fast response RTD thermowells.

W/8 ltr.

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TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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Enclosure to Serial: RNPD/89-0986

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	0 5 0 0 0 2 6 1	1 OF 0 7

TITLE (4)
FAILURE OF FAST RESPONSE RTD THERMOWELLS

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 5 0 0 0
0 2	1 1	8 9	8 9	0 0 2	0 0	0 3	1 3	8 9		0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 0 0

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

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)	X OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	Information Only
20.405(a)(1)(iv)	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)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Freddie L. Legette, Senior Reactor Operator	AREA CODE 8 0 3 — 3 8 3 - 1 2 5 3

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
B	A B	T W	W 1 0 8	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES *(If yes, complete EXPECTED SUBMISSION DATE)* — X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

During the recent refueling outage, H. B. Robinson Unit No. 2 underwent a Plant modification that removed the Reactor Coolant System (RCS) RTD Bypass Loops. This was accomplished by adding three (3) new dual-element fast time response thermowell mounted RTDs in the hot leg of each reactor coolant loop and one (1) in each cold leg. On February 11, 1989, at 1700 hours, while performing low power physics testing, a thermowell in "A" hot leg was discovered to be leaking. The thermowell was subsequently removed from the loop and examined. The results indicated a horizontal crack had propagated through the wall of the thermowell. The two remaining hot leg thermowells were subsequently removed and examined. The investigation revealed both thermowells had failures similar to that found on the original thermowell. The cause of the cracking has been determined through metallurgical analysis to be fatigue failure. The cyclic stresses applied to the thermowell were greater than expected and failure resulted. The thermowells have been modified and replaced.

An independent analysis of the new thermowells design has been performed by the Licensee. In addition the manufacturer has reviewed the event for potential 10CFR21 reportability and concluded that it is not reportable. This LER is submitted for information only.

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S PDC

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. Description of Event

During the recent refueling outage, H. B. Robinson Unit No. 2 underwent a Plant modification that removed the Reactor Coolant System (RCS) RTD Bypass Loops.[1] This was accomplished by adding three (3) new dual-element fast time response thermowell mounted RTDs in the hot leg of each reactor coolant loop and one (1) in each cold leg. The three hot leg thermowells were installed at 0°, 120°, and 240°, with 0° being the vertical direction with the 120° thermowell clockwise when looking toward the reactor vessel.

The preferred installation location for the RCS hot leg RTD Thermowells is in the flow scoops previously utilized by the RCS RTD bypass piping. This location was utilized by loops "B" and "C" hot leg RTD Thermowells. However, for the "A" loop the decision was made not to utilize the flow scoops because the 24 inch clearance necessary to install an RTD was not available at the 240° location due to the close proximity of an adjacent concrete wall. Since all three of the loop RTD's should be in the same plane, it was necessary to move the three "A" loop RTDs to a location down stream in the RCS piping at the entrance to the elbow upstream of "A" Steam Generator. To accommodate the thicker wall at the elbow and to meet the 4.5-inch insertion length into the flow stream, the thread regions were field machined back approximately one (1) inch, thus making the length of the thermowell, from the tip to the threads, one inch longer than the original design, (i.e., the design of the thermowells installed in scoops of loops B and C).

On February 11, 1989, at 1700 hours, while performing low power physics testing, the thermowell at the 240 degree position on the "A" hot leg was discovered to be leaking.[2] The 240° thermowell was subsequently removed from the loop and helium leak tested. The results indicated a horizontal crack that propagated through the wall of the thermowell at the transition region between the low pressure seal thread and its 0.777" diameter body (See attached sketch). The crack spans 200 degrees circumferentially and there are multiple crack initiation sites from the upstream side of the thermowell. Metallurgical examination indicated that the cracking was due to fatigue. Inspection also showed that the fillet radius at the transition region was 0.005 inch.

The 120° and the 0° thermowells were subsequently removed and examined. The 120° thermowell has a similar but partial (not-through-the-wall) crack found at the same region as in the 240° thermowell. The fillet radius at the transition region was approximately 0.015 inch. The crack was induced by fatigue mechanism. The 0° thermowell has a partial crack on the second (from the underside) thread which is just above the transition region. No clear evidence

1/ H. B. Robinson Unit No. 2 is a Westinghouse Pressurized Water Reactor Nuclear Power Plant in commercial operation since March 1971.
2/ EIIS Codes: System - AB; Component - TW; Manufacturer - W108.

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of corrosive elements was confirmed on the fracture face by Edax analysis. However, an abnormal step was observed at the thread where the crack was initiated. The fillet radius at the transition region on this thermowell was 0.030 inch.

II. Cause of Event

The cause of the cracking has been determined through metallurgical analysis to be fatigue failure. The cyclic stresses applied to the thermowell were greater than expected and failure resulted.

A contributor to the failure was the geometry of the shank. Sharp edges intensify the applied stress causing it to be magnified several times. The shank fillet radii vary from .005 inch to .030 inch. The variance in shank fillet radii resulted from a misinterpretation of the design drawing during manufacturing. The design drawing requirements are that the fillet radii are to be 0.030 plus or minus 0.010 inch.

Since fatigue cracking was observed on all three thermowells, flow induced vibration was the primary focus of the investigation. The calculated fundamental frequency of the 4.5" insertion length thermowell is 329 Hz and the corresponding Reynolds's number under nominal flow conditions was approximately 1.3 E6. Based on the calculated frequency, which compares favorably with the tested frequency of the original design, pump induced pulsation was ruled out as a viable mechanism. Also, since the Reynold's number calculated is in the aperiodic region, periodic vortex shedding should not occur.

Therefore, the original fatigue evaluation was carried out considering flow induced vibration due to random turbulent flow. The correlation of the random turbulent forces, based on a flow velocity of 55 ft/sec, acting on the thermowell tip and the exposed 0.777" diameter sections were also derived. The equivalent dynamic forces on the thermowell tip and the exposed 0.777" diameter section were calculated to be 17 and 41 lbf, respectively. The corresponding stresses induced at the transition/thread region were below the estimated actual fatigue endurance limit and thus do not explain the fatigue cracks observed on the thermowells, even with the effects of the undesirable fillet radius and rough surface finish taken into consideration. The corresponding displacement of the thermowell, using a thermowell finite element model, at the elevation of the pipe inside radius was calculated to be only .008 inches. Therefore, a conservative and enveloping approach was taken to assess the thermowell stress levels.

In this approach, the thermowells were assumed to vibrate through the .017 inch radial clearance between the thermowell and the pipe hole. Thus providing an upper bound on the flow loading imparted to the larger length thermowell. This approach is based on the observation that the two partially cracked thermowells

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(0° and 120°) showed no indication of impact on the pipe indicating the thermowells were probably not contacting the pipe. Based on these observations, the engineering analysis was carried out using the loads inferred from the displacement of the longer length thermowell through the total available clearance.

The conclusions of this assessment are as follows.

Comparing the magnitudes of alternating stresses calculated for the cracked thermowells to the endurance limits, it is reasonable to expect the observed cracking to occur, i.e., to have the 240° and 120° thermowells crack at the transition region and to have the 0° thermowell crack at the threads.

Therefore the cracking occurred due to a combination of a flaw in the threaded region of one thermowell, fillet radii on two thermowells that were smaller than that required by the design drawings and higher loads than anticipated.

III. Analysis of Events

For a condition of failure, of up to all 12 thermowells, an evaluation concluded that the Small Break Analysis of record would continue to bound the plant response and the expected transient from such a simultaneous failure would not challenge the acceptance criteria of 10CFR50.46 for the ECCS system as designed. Additionally, the failure of one or two thermowells for the H. B. Robinson plant does not constitute a safety issue because of the capacity of the charging system to maintain RCS inventory.

The mass/energy release associated with the postulated failure of these RTD thermowells would be negligible and would have no adverse effect on the results of the FSAR Analysis.

The loose parts evaluation for the potential of three broken thermowell pieces from "A" hot leg showed that this occurrence would not represent an unreviewed safety question but may in the worst case necessitate repairs to "A" Steam Generator.

IV. Corrective Action

The following features have been incorporated into the replacement thermowells.

a) Reduced thermowell insertion length by 1.0 inch. It was later determined that the 4.5-inch insertion length is not required.

b) Increased fillet radius of 0.07 ± 0.01 inch at the body to seal thread juncture.

c) Shot Peening of the body to seal thread juncture to induce surface compressive stress and polishing of the threads.

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d) Increased fillet radius of 0.055 ± .005 inch at the thread relief above the seal threads.

e) Application of chrome plating of 0.00125 radial thickness on the thermowell body at the zone of the exit into the I.D. of the reactor coolant pipe elbow.

Feature (a) is expected to provide for a lower vibratory response due to flow excitation.

Features (b), (c), and (d) will provide additional margin with respect to fatigue.

Feature (e) is expected to provide lower vibratory response and therefore lower dynamic stresses during plant operation.

Incorporating the above features and the loads from the cracking assessment of the longer length thermowells results in thermowell stress levels well within the ASME Code allowable fatigue endurance limits provided by ASME design curves I.9.2.2. The manufacturer also performed a "worst case" fatigue assessment on the modified thermowells as listed in this section assuming that they would also vibrate through the available clearance. This "worst case" evaluation showed that although the fatigue stress was slightly above the ASME Code allowable endurance value in fatigue, cracking would not occur because of the inherent safety factors in the ASME Code fatigue curves and the material properly enhancements due to peening and subsequent polishing.

An independent analysis of the new thermowells design has been performed by CP&L.

This analysis was based on the following:

1) Analytical values from WCAP 12186, Rev. 1

2) Thermowell fundamental frequency of 446 Hz

3) Displacement limited force was used as a "worst case"

4) Correlation between Reactor Coolant Pump run time and initial failure of the "A" hot leg 240° thermowell

5) Use of ASME design fatigue curves in Figures I-9.2.1 and I-9.2.2 of the Code

The results of this analysis conservatively indicate that the new thermowell design should not crack for a period of time of at least 9 operating months. Therefore, additional corrective action or analyses will be required prior to the end of this time period.

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In addition to the above, two programs have been initiated. The first is design, procurement, and analysis of a significantly more substantial thermowell for use in the "A" hot leg. Material stresses for this design are approximately 25% of those in the presently installed thermowells.

The second program will instrument a thermowell in the "A" hot leg with accelerometers to determine the forcing function acting on the thermowells. This data would then be used in modeling of the "A" hot leg to determine the actual forces acting on the thermowells and thus determine the actual material stresses.

The hot leg thermowells installed in the scoops in "B" and "C" Loops have much higher margins with respect to ASME Code allowable values since the scoop shields the thermowell from direct flow impingement.

The cold leg thermowells are of a different design which results in additional strength in the area of question. The cold leg thermowells experience less than 10% of the stresses of the hot leg thermowells due to significantly thicker construction.

In addition, subsequent dimensional inspections of previously supplied hot leg type thermowells, showed some of these not to be in compliance with the design drawings in the area where cracking occurred. This condition therefore could exist on the other H. B. Robinson hot leg thermowells. These other hot leg thermowells are located in the scoops as described above and analyses performed by the manufacturer considering these nonconformances show that the ASME Code stress requirements are met.

The manufacturer has also reviewed the above dimensional nonconformance with respect to potential 10CFR21 reportability and found that it is not reportable.

V. Additional Information

Failed component identification

A. RTD Thermowell; Weed Dwg. No. 0417-306134-001

B. Previous Similar Events

None

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(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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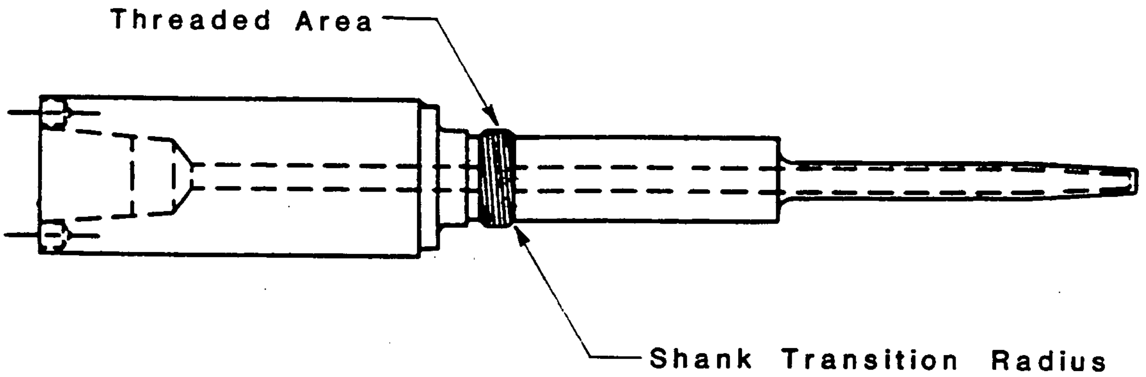

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(9-83)

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CP&L

Carolina Power & Light Company

ROBINSON NUCLEAR PROJECT DEPARTMENT
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Robinson File No: 13510C

Serial: RNPD/89-0986
(10 CFR 50.73)

United States Nuclear Regulatory Commission
Attn: Document Control Desk
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
LICENSEE EVENT REPORT 89-002-00

Gentlemen:

The enclosed Licensee Event Report (LER) is submitted, as an event of potential interest to the industry, in accordance with NUREG-1022 including Supplements No. 1 and 2. The event was evaluated against 10 CFR 50.73 and was determined not to meet the reportability requirements.

Very truly yours,

R. E. Morgan
General Manager
H. B. Robinson S. E. Plant

FLL:lko

Enclosure

cc: Mr. S. D. Ebneter
Mr. L. W. Garner
INPO

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