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REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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RNPD/89-0986 Enclosure to Serial:

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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. Enclosure to Serial: RNPD/89-0986

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II. Cause of Event

0.030 inch.

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

CCLUTY NAME (1) OCCET NUMBER (2) IF A NUMBER (6) PAGE (3) H. B. Robinson, Unit No. 2 0 [5 0 0 2 6 1 8 9 - 0 0 2 - 0 0 0 4 0F 0 COT of 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, 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 10CFS0.46 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.	RC Ferm 366A 83)	LICENSEE EVENT REPOI	RT (LER) TEXT CONTINU	ATION APP	LEAR REGULATORY COMMISS PROVED OMB NO. 3150-0104 PIRES: 8/31/88
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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.	thermowe the engine displace clearand The cond Comparing thermowe cracking transit Therefore region of that red III. Analysis For a conthat the response challeng designed H. B. Ru of the The mas thermow	ells were probably not of neering analysis was ca ement of the longer leng- te. clusions of this assess ng the magnitudes of al- ells to the endurance 1 g to occur, i.e., to have ion region and to have re the cracking occurre- of one thermowell, fill quired by the design dr s of Events ondition of failure, of e Small Break Analysis e and the expected tran ge the acceptance crite d. Additionally, the f obinson plant does not charging system to main s/energy release associ ells would be negligibl	contacting the pipe. arried out using the gth thermowell throug ment are as follows. ternating stresses ca imits, it is reasonab ve the 240° and 120° the 0° thermowell cra d due to a combinatic et radii on two therm awings and higher loa up to all 12 thermow of record would conti sient from such a sin ria of 10CFR50.46 for ailure of one or two constitute a safety tain RCS inventory. ated with the postula	Based on these of loads inferred fr h the total avail alculated for the ole to expect the thermowells crack on of a flaw in the nowells-that were ads than anticipa wells, an evaluat inue to bound the multaneous failur the ECCS system thermowells for issue because of ated failure of t	cracked able cracked observed cat the s. ne threaded smaller_than ted. ion concluded plant e would not as the the capacity , hese RTD

Enclosure to Serial: RNPD/89-0986

NRC Form 366A 9-83)		LICENSEE EVENT	REPORT (LER) TEXT CONT	U.S. NUCLEAR REGULATORY COMMISSI APPROVED OMB NO. 3150-0104 EXPIRES: 8/31/88
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н. в.	Robins	son, Unit No. 2		YEAR SEQUENTIAL REVISION NUMBER NUMBER
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	d)	Increased fillet rac the seal threads.	dius of 0.055 ± .005 in	nch at the thread relief above
	e)	Application of chron thermowell body at t coolant pipe elbow.	the zone of the exit in	radial thickness on the nto the I.D. of the reactor
		are (a) is expected to	o provide for a lower	vibratory response due to flow
	Featu fatig) will provide additio	nal margin with respect to
	Featu Tdynan	ure (e) is expected t nic stresses during p	o provide lower vibrat lant operation.	ory response and therefore lower
	the A curve asses they evalu Code the	ASME Code allowable f es I.9.2.2. The manu ssment on the modifie would also vibrate t uation showed that al allowable endurance inherent safety facto	Eatigue endurance limit ifacturer also performe ed thermowells as liste through the available of though the fatigue str value in fatigue, crac	well stress levels well within is provided by ASME design ed a "worst case" fatigue ed in this section assuming that clearance. This "worst case" ress was slightly above the ASME cking would not occur because of atigue curves and the material quent polishing.
	An in CP&L		of the new thermowells	design has been performed by
	This	analysis was based o	on the following:	·
	1)	Analytical values f	from <u>WCAP</u> 12186, Rev. 2	1
	2)	Thermowell fundamer	ntal frequency of 446 I	Hz
	3)		ed force was used as a	
	4)	Correlation between the "A" hot leg 240	n Reactor Coolant Pump O° thermowell	run time and initial failure of
	5)	Use of ASME design Code	fatigue curves in Fig	ures I-9.2.1 and I-9.2.2 of the
	The desi Ther	results of this anal gn should not crack	vsis conservatively in	dicate that the new thermowell

9-83)		LICENS	SEE EVEN	TREPOR	T (LER) TEXT	CONTINU	JATION	_	APPROVED (DMB NO. 3150-0	
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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION EXPIRES: 8/31/88 CILITY NAME (1) H. B. Robinson, Unit No. 2											
V.	lesign for us 25% of The se accele This d actual stress The ho higher shield The co streng 10% of constr In add type t drawin exist thermo by the stress The mark respect Addit Faile A.	<pre>, procurer e in the those in cond prog rometers ata would forces a es. bt leg the margins ls the the old leg th the the the stre ruction. dition, su thermowell hgs in the on the ot owells are e manufacture to pote ional Info d componer RTD Therr Previous</pre>	ment, and "A" hot 1 the press ram will to determ then be cting on rmowells with resp rmowell area of sses of bsequent s, showe e area wh her H. B e located curer con ments are er has al ential 10 ormation nt identi nowell; W	I analysi leg. Mai sently in instrum nine the used in the the install pect to from dir s are of questio the hot dimensi d some c ere crace. Robins in the sidering met. So revie OCFR21 re lfication	is of a sig terial stre nstalled th ent a therm forcing fu modeling o rmowells an ed in the s ASME Code a ect flow im a differen n. The col leg thermos onal inspe- f these no- king occur son hot leg scoops as these non ewed the ab eportabilit	nificant sses for ermowell in inction ac of the "A" in the "A" in the "A" in thus do scoops in allowable mpingemen in design d leg th wells due ctions of t to be i red. Thi thermowe described conformar ove dimen y and for	this of this of the "A cting of "hot " etermin "B" a: value t. which ermowe to si previ n comp s cond ells. I above nces sh nsional	A" hot 1 on the t leg to c ne the a nd "C" 1 s since result 11s exp gnifica ously s liance ition t These o and an low that	eg with hermowel letermine actual ma Loops hav the scoo s in addi erience 1 ntly thic upplied h with the herefore ther hot alyses po the ASM	ls. the terial e much p tional ess that ker not leg design could leg erformed E Code with	n

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ROBINSON NUCLEAR PROJECT DEPARTMENT POST OFFICE BOX 790 HARTSVILLE, SOUTH CAROLINA 29550

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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 Washington, D. C. 20555

> 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:1ko

Enclosure

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