

Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

February 12, 2010

10 CFR 50.73

U.S. Nuclear Regulatory Commission ATTN: Document Control Desk Washington, D. C. 20555-0001

> Browns Ferry Nuclear Plant, Unit 2 Facility Operating License No. DPR-52 NRC Docket No. 50-260

Subject: Licensee Event Report 50-260/2009-004-01

The enclosed Licensee Event Report (LER) provides details of a Technical Specifications shutdown due to a rise in unidentified drywell leakage and a subsequent automatic scram due to a failure in the reactor protection system. This LER was revised to correct LER Section 11 and to provide an updated Abstract and Narrative description of the event.

The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73(a)(2)(i)(A), as the completion of any nuclear plant shutdown required by Technical Specifications, and in accordance with 10 CFR 50.73(a)(2)(iv)(A), as any event or condition that resulted in manual or automatic actuation of the Reactor Protection System (RPS) including: reactor scram or reactor trip.

There are no new regulatory commitments contained in this letter. Revisions are identified by bars in the right-hand margin. Should you have any questions concerning this submittal, please contact F. R. Godwin, Site Licensing and Industry Affairs Manager, at (256) 729-2636.

Respectfully,

K. J. Polson Vice President

cc: See page 2



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Enclosure cc (w/ Enclosure):

NRC Regional Administrator - Region II

NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

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NRC F	NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION					SION A	PPROVED	BY OMB NO.	3150-0104		E	EXPIRES 08/31/2010												
(9-2007)	(9-2007)						Estimated burden per response to comply with this mandatory collection request. 80 hours						30 hours.											
						industry. Send comments regarding burden estimate to the Records and FOIA/Privac					VPrivacy													
						Service Bra	nch (T-5 F52), U.S	. Nuclear Reg	ulatory	Commission	, Washing e Desk (ton, D0	C 20555- Office of											
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	LICENSEE EVENT REPORT (LER) Budget, Washington, DC 20503. If a means used to impose an inform not display a currently valid OMB control number, the NRC may not								nformatio y not con	n collect duct or	tion does sponsor,													
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4. TIT	4. TITLE: Technical Specifications Shutdown Due to Rise in Unidentified Drywell Leakage																							
5. E	EVENTI	DATE	6.	LER NUM	BER	7. R	EPORT	DATE		8. OT	HER FACI	LITIE	S INVOLV	VED										
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06	11	2009	2009	- 004	- 01	02	12	2010	FACILITY	YNAME				DOCKE	NUME	BER								
9. OPE	RATIN	G MODE	11.	THIS REP	ORT IS S	UBMITT	ED PURS			REQUIREMEN	TS OF 10	CFR §	§: (Check	all that	apply	1)								
			20.	2201(b)			0.2203(a)	(3)(i)	[50.73(a)(2)	(i)(C)		50.7	73(a)(2)	(vii)									
			20.	2201(d)).2203(a)	(3)(ii)	[50.73(a)(2)	(ii)(A)		50.7	73(a)(2)	(viii)(/	4)								
	1		20.2	2203(a)(1)).2203(a)	(4)	C	50.73(a)(2)	(ii)(B)		50.7	73(a)(2)	(viii)(E	з́)								
		:	20.2	2203(a)(2)(i)	50).36(c)(1)	(i)(A)	[50.73(a)(2)	(iii)		50.7	73(a)(2)	(ix)(A))								
10 PO	WEBLI	EVEL	20.2	2203(a)(2)(ii)	50	0.36(c)(1)	(ii)(A)		50.73(a)(2)	(iv)(A)		50.7	73(a)(2)	(x)									
			$\Box 20.2203(a)(2)(iii) \qquad \Box 50.36(c)(2) \qquad \Box 50.73(a)(2)(v)(A) \qquad \Box$				73.7	73.71(a)(4)																
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			20.2	2203(a)(2)(vi)	50).73(a)(2)	(i)(B)	0	50.73(a)(2)	(v)(D)		Spec	ify in Abstra	ict below	or in NRC								
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A	1200	nours Ce	entral D	aylight I i	me (CD	I) on J	une 11,	2009,	Browns		ear Plan		t 2 expe	enenc	ed a	.								
ris	se in ai	rywell lea	ikage di	uring rea	ctor star	tup. Ir	he tour-l	nour u	nidentifi	ed leak rate	e from U8	00 to	5 1200 1	nours	CDI									
or	June	10, 2009), was 0	gallons	per mini	ute (GP	'M), whi	le the	tour-hou	ir unidentifi	ed leak r	ate f	rom 080	JU to '	200									
hc	ours Cl	J on Ju	ne 11, 2	2009 was	3.88 G	PM. Tr	nis incre	ase in	leakage	e exceeded	the Tech	nnica	I Specil	licatio	ns									
(1	S) Lim	iting Col	dition f	or Opera	tion (LC	O) 3.4.	4 limit o	fa2G	SPM inc	rease in un	dentified	lleai	kage in	a 24 r	iour									
pe	eriod. A	At 1555	ours C	DT on Ju	ne 11, 2	2009, U	nit 2 init	tiated a	a reacto	r shutdown	via a ma	inual	I reacto	r SCR	AM									
_	to comply with TS LCO 3.4.4 Condition C to be in Mode 3 in 12 hours and to be in Mode 4 within 36 hours.									and to be in	n Mode 4	with	nin 36 h	ours.										
to	compl	Following verification that the procedure, 2-AOI-100-1, Reactor Scram, actions were completed, the reactor									eactor													
to Fo	compl ollowin	g verifica	tion that	t the pro	cedure,	2-A01-	mode switch was placed in Shutdown. The increase in unidentified leakage was due to failure of a Main Steam																	
to Fo m	compl ollowin ode sv	g verifica vitch was	ition that placed	it the pro in Shute	cedure, own. T	2-AOI-	ease in	uniden	tified le	akage was	due to fa	ilure	of a Ma	ain Ste	ting R Safety Relief Valve (SDV) to fully close. As a result of this steam leakage two main steam SDV/ tailoine									
to Fo m Li	compl ollowin ode sv ne B S	g verifica vitch was afety Re	tion that placed lief Valv	it the pro in Shutc ve (SRV)	cedure, lown. T to fully (2-AOI- he incre close. A	ease in As a res	uniden ult of t	tified le his stea	akage was Im leakage,	due to fa two mai	ilure n ste	of a Ma am SR	ain Ste V tailp	ipe									
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NRC FORM 366A (9-2007)	LICENSEE EVENT R	U.S. NUCLEAR REGULATORY COMMISSION						
FACILITY NAME (1)	DOCKET (2)	L	ER NUMBER (6)		PAGE (3)			
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER				
Browns Ferry Nuclear Plant Unit 2	05000260	2009	004	01	2 of 6			

NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

I. PLANT CONDITION(S)

Prior to the event, Browns Ferry Nuclear Plant (BFN) Units 1 and 3 were operating in Mode 1 at 100 percent thermal power (approximately 3458 megawatts thermal). BFN Units 1 and 3 were unaffected by the event. BFN Unit 2 was in Mode 1 at approximately twelve percent and in power ascension following a refueling outage.

II. DESCRIPTION OF EVENT

A. Event:

During reactor startup from the BFN Unit 2 Spring refueling outage, a failure of a Main Steam (MS) Line B Safety Relief Valve (SRV) [SB] to fully close was revealed. Steam leakage through this SRV stopped when reactor pressure decreased to approximately 850 psig. The Tennessee Valley Authority (TVA) initially thought the steam leakage was due to pilot valve leakage because of observed discharge tailpipe indications and past experiences with pilot leakage. However, following destructive testing, it was determined to be steam leaking by the main valve body. As a result of this steam leakage, two MS SRV tailpipe vacuum breakers, 2.5 inch and 10 inch, were cycling. This SRV failure and vacuum breaker cycling allowed steam to enter the drywell instead of going to the torus.

At approximately 1200 hours Central Daylight Time (CDT) on June 11, 2009, BFN Unit 2 experienced an increase in drywell leakage during reactor startup. The four-hour unidentified leakage from 0800 to 1200 hours CDT on June 10, 2009, was 0 gallons per minute (GPM), the four-hour unidentified leakage from 0800 to 1200 hours CDT on June 11, 2009, increased to 3.88 GPM. This increase in Reactor Coolant System (RCS) operational leakage exceeded the Technical Specifications (TS) 3.4.4, RCS Operational Leakage, limit of a 2 GPM increase in unidentified leakage within the previous 24 hour period.

Therefore, the TS Limiting Condition for Operation (LCO) 3.4.4 was not met and at 1555 hours CDT on June 11, 2009, Unit 2 Operations personnel initiated a manual reactor scram to comply with TS 3.4.4 LCO Condition C, to be in Mode 3 in 12 hours and to be in Mode 4 within 36 hours.

During the reactor shutdown, all automatic functions resulting from the manual scram occurred as expected. All control rods [AA] inserted. No primary containment isolation system (PCIS) [JE] isolations were received.

Subsequently, at 1609 hours CDT on June 11, 2009, a full reactor scram occurred due to Intermediate Range Monitor 'C' spiking high concurrent with the inability to reset Reactor Protection System [JC] (RPS) 'B' scram channel.

Following verification that the 2-AOI-100-1, Reactor Scram, actions were completed, the reactor mode switch was placed in shutdown.

TVA is submitting this report in accordance with 10 CFR 50.73(a)(2)(i)(A), as the completion of any nuclear plant shutdown required by Technical Specifications, and in accordance with 10 CFR 50.73(a)(2)(iv)(A), as any event or condition that resulted in manual or automatic actuation of the RPS including: reactor scram or reactor trip.

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NRC FORM 366A				U.S. NUCLEAF	R REGULATOR	Y COMMISSION		
(9-2007)	LICENSE	E EVENT F	REPORT	(LER)				
	FACILITY NAME (1)	DOCKET (2)				PAGE (3)	AGE (3)	
			YEAR	NUMBER	NUMBER			
Browns Ferry I	Nuclear Plant Unit 2	05000260	2009	004	01	3 of 6		
NARRATIVE	(If more space is required, use additional co	pies of NRC Forr	n 366A) (17)				ا ر منعقب بر رو ی	
B.	Inoperable Structures, Compon	ents, or Syst	ems that	Contributed t	<u>o the Event:</u>	L		
	None.							
C.	Dates and Approximate Times of	of Major Occu	irrences:					
	June 11, 2009, at 1555 hours CD1	Г Un	it 2 reacto	r manually scr	ammed.			
	June 11, 2009, at 1609 hours CDT	۲ Un Int wit	iit 2 full rea ermediate h inability	actor scram oc Range Monito to reset RPS '	curred due to or 'C' spiking B' scram cha	o concurrent annel.		
	June 11, 2009, at 1724 hours CD1	Г Ор Sy 10	erations n stem repo CFR 50.7	nade an Emer rt in accordan 2(b)(2)(i)(B).	gency Notific ce with	ation	-	
· D.	Other Systems or Secondary Fu	unctions Affe	<u>cted</u>		•			
	None.							
E.	Method of Discovery							
	The annunciator for Drywell Floor Main Control Room.	Drain Sump F	Pump Exce	essive Operati	on was recei	ved in the		
F.	Operator Actions							
х.	Operations personnel completed t entered 2-AOI-100-1, Reactor Scr	he shutdown am.	as require	d by Technica	l Specificatio	ns 3.4.4 and		
G.	<u>Safety System Responses</u>							
	The RPS logic responded to the n isolations were received.	nanual reactor	r scram. A	Il control rods	inserted. No	PCIS		
	RPS 'B' scram channel did not res	set after the m	anual scra	ım.				
III. CA	AUSE OF THE EVENT							
А.	Immediate Cause	·		P	<i>.</i> .			
	The immediate cause of the excess SRV to fully close. Also, two MS sources cycling. This cycling allowed stea	ssive RCS op SRV tailpipe v m to enter the	erational le acuum bre drywell in	eakage was th eakers, 2.5 inc istead of going	e failure of a h and 10 inc g to the torus	MS Line B h, were		
B.	Root Cause			• • • • •			1	
	There are two root causes for the	excessive RC	S operatio	onal leakage.				
· · ·	The first root cause of this event we main joint design that develops a second a Target Rock Two or Three Stage identified that mating threads on the point that the shaft appeared to be body from cycling correctly. Generation we applicable to Target Rock Two or	vas identified fretting conditi e SRV. Destri he main valve e cocked appr eral Electric (G hich previous Three Stage S	as an inad ion after ye uctive exa piston-to- oximately SE) Service Iy occurred SRVs, and	equate origina ears of service mination at W main valve ste 1/4 inch, whic e Information I d at Plant Hato BFN has Tan	al manufactur of a main bo yle Laborator em were dam h would prev Letter (SIL) 6 ch. GE SIL 6 get Rock Two	rer-threaded ody valve on ries aged to the ent the main 46 46 is o Stage		
• ,	······································		<u>.</u>					

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			U.S. NUCLEAF	REGULATOR	Y COMMISSION		
LICENSEE EVENT REPORT (LER)							
FACILITY NAME (1)	DOCKET (2)	L	ER NUMBER (6) .	PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
Browns Ferry Nuclear Plant Unit 2	05000260	2009	004	01	4 of 6		
NARRATIVE (If more space is required, use additional cop	ies of NRC Form	366A) (17)					
SRVs. The design detail dimension assembled, permitted the lead thre between the load bearing shoulder this condition, the torque (or applier certification testing of the main valv normal plant operations allows the mechanical binding of the SRV.	ning tolerance ad of the pisto and the final t d preload) bet re body. Subs piston to fret t	for the pa on to prema thread on t ween the j sequent vit he threads	rts, when orig aturely contac the stem of the am nut and the pration of the s of the stem	ginally manuf ct the under- ne main valve he piston was loosened pis which can re	actured and cut area disc. In s lost during ton during sult in		
The SRV steam leakage into the dr breaker was found stuck open and mechanism found to be weak. The cycling due to the leaking SRV. Ste SRV, through the open vacuum bre	ywell occurre the 10 inch va se conditions eam leakage v eakers, and int	d because acuum brea were dete vas flowing to the dryw	the associate aker was four rmined to be g down the ta rell.	ed 2.5 inch vand open with caused by ex ilpipe of the l	acuum the spring xcessive MS Line B		
The second root cause of this even GE SIL 646 at BFN. Organizationa the underlying root cause for failure recommended actions consisted of Target Rock would provide, over th the succeeding 6 to 10 years of ser to be inspected was recommended inspection of other SRVs not install	t deals primar I to Organizat to fully imple inspections a e next three n vice. For inst to be approxi ed or spares v	ily with the ional Inter ment GE S nd, as nee efueling ou alled SRVs mately one was also p	e failure to ful face Deficien SIL 646. The eded, interim itages with fo s and spares, e-third of the rescribed.	ly implement cies were ide GE SIL 646 modifications llow-up inspe the populati SRVs each c	entified as 6, which ections in on of valves outage. The		
Per the cause analysis, previous in parts: generation of work orders fo maintenance (PM) work orders gen implementation revealed that a bre outage scope. The MS System En GE SIL 646 implemented; however replaced during refuel outages to c any given outage. Therefore, refue	nplementation ir valves outsid lerated on the akdown occur gineer genera c, the Valve Er oincide with p el outage work	of GE SIL de of the S remaining red during ted the ap ngineer ide ast practic scope dev	646 at BFN BL requireme valves. Rev initial develo propriate doc ntified only o es of changin velopment an	consisted of t nt and preve riew of both p pment of the cumentation t ne main body ng only one m nd control we	two main ntative parts of the refueling o have y to be nain body in re deficient.		
The root cause for the failure of the scram relay/contactor terminal conr channel did not reset as expected. was not re-energized due to a loose relay/contactor operated correctly. Unit 2 refuel low power startup. Fa (pressure pad) of the power feed co adequate tightening during coil repl	RPS 'B' scrain nection. On J Investigation connection to This relay had ilure analysis onnection bloc lacement two	m channel une 11th a found that block. The d been rec has identif ck was loos weeks price	to not reset w t 1602 CDT, the 5A-K14H connection w ently operate fied that the fir se with the lik or to the even	was found to the RPS 'B' s I scram relay was tightened d in support iction connec ely cause of t.	be a loose scram //contactor d and the of the BFN ction less than		
C. Contributing Factors							
A significant component of the unid Reactor Vessel Drain Valve. The p	entified RCS acking leak w	operationa as caused	I leakage was by ineffective	s a packing le maintenanc	eak on the e. I		
IV. ANALYSIS OF THE EVENT			a sha ka dha		, . I		
During reactor startup from the BFN to fully close was revealed. Steam decreased to approximately 850 ps valve leakage because of observed	N Unit 2 Spring leakage throu ig. TVA initial I discharge tai	g refueling Igh this SR Iy thought Ipipe indic	outage, a fai XV stopped w the steam lea ations and pa	lure of a MS hen reactor p akage was du ast experienc	Line B SRV pressure le to pilot es with pilot		

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NRC FORM 366A (9-2007)			U.S. NUCLEAR	REGULATO	RY COMMISSION			
LICENSEE EVENT REPORT (LER)								
FACILITY NAME (1)	DOCKET (2)	L	ER NUMBER (6))	PAGE (3)			
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Browns Ferry Nuclear Plant Unit 2	05000260	2009	004	01	5 of 6			
NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)								
leakage. However, following destr main valve body.	uctive testing,	it was dete	ermined to be	e steam leak	king by the			
The GE SIL 646 failure mode is de valve stem threaded joint due to de failure mode occurs when the lead cut area between the load bearing In this condition, the torque (or app during certification testing of the m of torque condition is undetectable the main valve body is installed on allows the piston to fret the threads degradation process long enough, between the piston and cylinder ca can result in the mechanical bindin opens is identical for both the mec that resulted in the failure of 2-PCV operating modes of the SRV. The 1999 and had not been inspected With respect to the vacuum breake normal plant conditions. The vacu Normally, during a transient situation each time the SRV opened and clo	escribed as loss eformation of t ing thread edg shoulder and blied preload) I ain valve body without disase the steam line s of the stem. the entire three annot be maint ag of the SRV. hanical and ele J-1-23 is applie valve was inst per the GE SII er cycling, typic um breakers w on, the vacuum osed. Since the of the leaking S	s of torque he leading le of the pi final thread between the vat a limite sembly of t e header, ti lf the main aded joint ained whe The mech ectrical ope cable to bo alled at the cable to bo alled at the cable to bo alled at the cable to bo salled at the cable to bo salled at t	a at the main y edge of the p ston prematu d on the stem le jam nut and d steam supp the certified m he steam flow n valve body i is compromis n the SRV is nanism by whi ening mode. th the mecha e 2-PCV-1-23 mmendations g due to the l could potent nique event, f proximately 2	valve piston piston threa rely contact of the main d the piston ply test facil nain valve b v-induced vi s subjected sed. The ali required to ich the main Therefore, nical and el position sin s. akers do no eaking 2-PC ially cycle o the vacuum 0 hours.	n-to-main ds. The ds. The ts the under- in valve disc. was lost ity. The loss body. When ibration to this ignment open which in valve body the condition lectrical ince April t cycle under CV-1-23. ince and breakers			
V. ASSESSMENT OF SAFETY CONSEC	QUENCES				, -			
The safety consequences of this event complicated. The operational impact v reset the reactor scram at 1602 hours	: were not sign vas manageab CDT.	ificant. Th le during t	e manual scr hese valve fa	am was not ilures. Ope	erations			
Two cases identified in the industry wh that either did not open or only partially Section 14.5, Analysis of Abnormal Op result in a scram of the reactor from lov accidents with only 12 of the 13 SRVs limiting conditions, beginning of core lif condition or event. Further, based on of a history of the GE SIL 646 failure m confidence that the installed SRVs will	ere the SRV fa y opened. BFN erational Trans w power opera available for o fe at rated core the infrequent nechanism fror perform their	ailure mec N Updated sients – Up tion. Inclu pening and flow conc occurrence n past insp safety func	hanism has o Final Safety orated, includ ided are anal d for the inad litions. These e of this type bections of BF tion.	Analysis Re es various a yses for trai vertent oper e analyses l of valve fail FN MS SRV	ulted in valves eport (UFSAR) analyses, which nsients and ning of a MSRV at bound this actual ure and the lack /s, there is a high			
With the exception of the RPS failure to reset, all safety systems operated as required during the manual scram. As expected, there were no PCIS Group 2, 3, 6, or 8 isolations. Although the Emergency Core Cooling Systems were available, none were required. No MS SRVs [SB] actuated. The turbine bypass valves [JI] maintained reactor pressure. The main condenser remained available for heat rejection. Reactor water level was recovered and maintained by the reactor feed water [SJ] and condensate [SG] systems. Therefore, TVA concludes that there was no significant reduction in the protection of the publi by this event.								
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NRC FORM 366A (9-2007)		1 - 1 - 1 - 1 - 1 - 1 - 1 - 1 - 1 - 1 -	1 - 1 - 1 - 1 - 1 - 1 - 1 - 1 - 1 - 1 -		· · · · · · · · · · · · · · · · · · ·			

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

VI. CORRECTIVE ACTIONS

A. <u>Immediate Corrective Actions</u>

Operations performed the immediate actions of operating procedure "Relief Valve Stuck Open." The immediate actions were to identify the stuck open SRV by observing Safety Relief Valve Tailpipe Flow or Main Steam Relief Valve Discharge Tailpipe Temperature. Operations attempted to close the MS SRV, but it still indicated partially open and a work order was initiated.

B. <u>Corrective Actions to Prevent Recurrence</u> - The corrective actions to prevent recurrence are being managed by BFN's corrective action program.

The corrective actions to prevent recurrence are to complete an inspection and refurbishment of all affected main body valves installed on Units 2 and 3 at BFN in accordance with the GE SIL 646 recommended action. TVA will fully implement these recommendations.

The second root cause corrective actions are to revise procedures to ensure appropriate PM work orders are scheduled and to require additional rigor and documentation to initial outage scoping. A training needs analysis will be performed to determine training needs with regards to engineering responsibility for outage scope.

For the extent of condition evaluation, corrective actions also include performance of a review of previous, associated corrective action documents to determine if applicable GE SILs and GE Technical Information Letters (TILs) were appropriately implemented at BFN.

To address the relay/contactor loose connection, periodic verification of coil power termination tightness was added for each relay/contactor being inspected in the procedures.

VII. ADDITIONAL INFORMATION

A. Failed Components

Failed components are a MS Line 'B' SRV, associated vacuum breakers, and Reactor Vessel Drain Valve stem packing.

B. <u>PREVIOUS LERS ON SIMILAR EVENTS</u>

None.

C. Additional Information

Corrective action documents for this report are Problem Evaluation Reports 173480, 174037, and 174044.

D. Safety System Functional Failure Consideration:

This event is a not a safety system functional failure in accordance with NEI 99-02.

E. <u>Scram With Complications Consideration:</u>

This event was not a complicated scram according to NEI 99-02.

VIII. COMMITMENTS

None.