April 3, 2006

U.S. Nuclear Regulatory Commission 10 CFR 50.73
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D. C. 20555-0001

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 2 - DOCKET 50-260 - FACILITY OPERATING LICENSE DPR - 52 - LICENSEE EVENT REPORT (LER) 50-260/2005-004-01

The enclosed revised report provides details of a plant condition which involved a leakage path between the residual heat removal (RHR) system and the RHR Service Water (RHRSW) system inside a Unit 2 RHR heat exchanger. The revision contains additional information regarding the condition identification and corrective actions to prevent recurrence.

In accordance with 10 CFR 50.73(a)(2)(ii)(B), TVA reported this event as an unanalyzed condition that could have significantly degraded plant safety. This event was also reportable under 10 CFR 50.73(a)(2)(v)(C) as a condition affecting the control of the release of radioactive material.

There are no commitments contained in this letter.

Sincerely,

Original Signed by:

Brian O'Grady

cc: See page 2

Enclosure
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(Via NRC Electronic Distribution)

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S:lic/submit/lers/260 2005-04

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-2004)

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Browns Ferry Unit 2	05000260	1 OF 7

4. TITLE
Primary to Secondary Leakage in Residual Heat Removal Heat Exchanger in Excess of Analyzed Limits

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME none	DOCKET NUMBER N/A
04	16	2005	2005-004-01			04	03	2006	FACILITY NAME none	DOCKET NUMBER N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 50

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(i)(C)		50.73(a)(2)(vii)
	20.2201(d)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(A)
	20.2203(a)(1)		20.2203(a)(4)	X	50.73(a)(2)(ii)(B)		50.73(a)(2)(viii)(B)
	20.2203(a)(2)(i)		50.36(c)(1)(i)(A)		50.73(a)(2)(iii)		50.73(a)(2)(ix)(A)
	20.2203(a)(2)(ii)		50.36(c)(1)(ii)(A)		50.73(a)(2)(iv)(A)		50.73(a)(2)(x)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(v)(A)		73.71(a)(4)
	20.2203(a)(2)(iv)		50.46(a)(3)(ii)		50.73(a)(2)(v)(B)		73.71(a)(5)
	20.2203(a)(2)(v)		50.73(a)(2)(i)(A)	X	50.73(a)(2)(v)(C)		OTHER
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(B)		50.73(a)(2)(v)(D)		specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER *(Include Area Code)*
Paul S. Heck, Nuclear Engineer, Licensing and Industry Affairs	256-729-3624

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	BO	HTX	PERFEX	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (*if yes, complete 15. EXPECTED SUBMISSION DATE*)	■ NO				

ABSTRACT (*Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines*)

On 3/21/05, with residual heat removal (RHR) system train 2A in service for shutdown cooling, a radiation alarm on the residual heat removal service water (RHRSW) side of the heat exchanger (HTX) was received. Sampling confirmed the presence of radioactivity. Immediate corrective actions were focused on evaluating the condition of the 2A HTX, and the potential impact of an RHR HTX leak on post-accident design basis assumptions and limitations was not initially recognized. On 4/17/05, after Unit 2 had exited the refueling outage and was in power ascension, indication of a leakage of contaminated water into the RHRSW system was again observed. Activities were undertaken to identify the specific source of the leak, which was localized to RHR HTX 2A. On 4/22/05, engineering evaluations concluded that relatively small leakage rates could result in unacceptable radioactive material release to the environment in post-accident situations if significant fuel damage was assumed to have occurred. The RHR 2A containment cooling functions of suppression pool cooling, suppression pool spray, and drywell spray were declared inoperable, and the appropriate Technical Specification Limiting Condition for Operation actions were invoked. Additionally, on 4/28/05, a manual isolation valve on the RHRSW inlet to the HTX was closed, rendering it unavailable for service. During the interval between reactor startup and closure of the manual valve, had an event occurred which involved significant fuel damage, this HTX could have been utilized for reactor or containment cooling, and its use could have resulted in an unanalyzed release of radioactive material to the environment. The root cause of this condition was raw water corrosion of soft iron gasket and gasket seating surfaces within the HTX. Corrective actions include revising relevant maintenance procedures, evaluating RHR HTX maintenance histories, estimating the rate of gasket material loss, and scheduling and performing preventive maintenance as appropriate.

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

I. PLANT CONDITION(S)

At the time the condition was recognized, Unit 2 was in Mode 1 at approximately 50% power during power ascension following its Cycle 13 refueling outage. Unit 1 was shutdown and defueled and was unaffected by the event. Unit 3 was in Mode 1 at approximately 3458 megawatts thermal (100 percent power) and was also unaffected by this event.

II. DESCRIPTION OF EVENT

A. Event:

On March 21, 2005, residual heat removal (RHR)[BO] system train 2A was placed into service for shutdown cooling at 1540 hours, following shutdown of the reactor for refueling. Subsequently, a radiation alarm on the residual heat removal service water (RHRSW)[BI] side of the heat exchanger was received, and chemistry sampling confirmed the presence of radioactivity soon thereafter. A corrective action document was written at that time, however, the immediate corrective actions were focused on evaluating the condition of the 2A heat exchanger, and the potential impact of an RHR heat exchanger leak on post-accident design basis assumptions and limitations was not initially recognized. Following completion of the refueling outage, after Unit 2 had exited the refueling outage and was in power ascension, on April 17, 2005, indication of leakage of contaminated water into the RHRSW system was again observed. Detailed plant activities were undertaken to identify the specific source of the leak, and on April 18, 2005, the leak was localized to the RHR system heat exchanger 2A. During normal reactor operations, the RHRSW flowpath through the RHR 2A heat exchanger is isolated via a motor operated valve (discharge side) and a check valve (inlet side). There is no RHRSW flow routinely passing through the heat exchanger. At this time, the potential impact on post-accident design basis assumptions was recognized, and Engineering evaluations were commenced to determine the safety impact of such leaks under different scenarios. On April 22, 2005, these evaluations concluded that relatively small leakage rates could result in unacceptable radioactive material release to the environment in post-accident situations if significant fuel damage was assumed to have occurred. The RHR 2A containment cooling functions of suppression pool cooling, suppression pool spray, and drywell spray were declared inoperable, and the appropriate Technical Specification (TS) Limiting Condition for Operation (LCO) actions were invoked. Additionally, on April 28, 2005, a manual isolation valve on the RHRSW inlet to the heat exchanger was closed. Closure of this valve rendered the heat exchanger unavailable for service.

Diagnostic testing was then performed to locate the specific source of the leak within the RHR 2A heat exchanger. Subsequently, following an interval for work planning and the staging of material, work was successfully completed to repair the heat exchanger leak. During the interval from April 16th, when reactor startup had commenced, to April 28th, when the manual inlet isolation valve was tagged closed, nothing prohibited use of this heat exchanger for reactor or containment cooling. Had an event occurred which involved significant fuel damage, use of this heat exchanger could have resulted in a release of radioactive material to the environment beyond that analyzed in the BFN design and licensing bases. Plant safety is deemed to have been potentially significantly degraded during this approximate 12-day interval.

This condition created a radioactive material release path greater than analyzed, resulting in short-term plant operation in an unanalyzed condition that under certain scenarios could have significantly degraded plant safety. The condition is reportable in accordance with 10 CFR 50.73 (a) (2) (ii) (B). Reporting under 10 CFR 50.73(a)(2)(v)(C) also applies as this condition affected systems which impact the control of the release of radioactive material.

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

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None

C. Dates and Approximate Times of Major Occurrences:

March 21, 2005	0900 hours CDT	U2 scrammed to commence the U2C13 refueling outage
	1540 hours CDT	RHR train 2A placed into service for shutdown cooling
	1605 hours CDT	Effluent high radiation alarm received
	1750 hours CDT	Chemistry sampling confirmed the presence of radioactivity. Problem Evaluation Request (PER) [BFN corrective action document] 79143 initiated. Investigation of the leak source and evaluation of heat exchanger conditions were undertaken
April 16, 2005	0249 hours CDT	U2 start-up for Cycle 14 operations commenced
April 17, 2005	1445 hours CDT	Chemistry sampling detected contamination in RHRSW A samples. Off-site Dose Calculation Manual (ODCM) concentration limits were not exceeded.
April 18, 2005	2258 hours CDT	RHR and RHRSW system pressure responses to equipment manipulations indicated a possible leak in RHR heat exchanger 2A.
April 22, 2005	1220 hours CDT	Operations notified by Engineering that evaluations had determined operation with unquantified leakage rates was outside the plant design and licensing bases. A caution tag was placed on the RHRSW flow path through the heat exchanger, and Operations declared RHR 2A inoperable for the functions requiring RHRSW support (suppression pool cooling, suppression pool spray, and drywell spray) and entered the applicable 30-day LCO.
April 28, 2005	1228 hours CDT	A manual isolation valve on the heat exchanger RHRSW inlet piping was closed and tagged to provide another barrier to leakage in addition to the inlet check valve.
April 30, 2005	1800 hours CDT	Testing was completed to specifically identify the leak location as the floating head flange of the RHR 2A heat exchanger.
May 4, 2005	0100 hours CDT	7-day low pressure coolant injection (LPCI) LCO 3.5.1.A entered and the RHR 2A train removed from service to effect repairs to the heat exchanger.
May 9, 2005		Technical Specifications amendment requested by BFN was approved by NRC for one-time use to extend the LPCI LCO completion time to 14 days.

May 15, 2005	1038 hours CDT	Repairs and post-maintenance testing completed satisfactorily. RHR 2A heat exchanger and train declared operable and LPCI and containment cooling LCO's exited.

D. Other Systems or Secondary Functions Affected

None

E. Method of Discovery

Chemical analysis identified a possible leak into the RHRSW system. Operations personnel subsequently manipulated the RHR and RHRSW systems, and the observed RHR and RHRSW system pressures were consistent with a leak internal to the RHR 2A heat exchanger.

F. Operator Actions

Operator action in response to this condition was appropriate. On both March 21 and April 17, Operations directed Chemistry to confirm the alarm. On April 17, when plant conditions called RHR system operability into question, Operations requested evaluation from Engineering after being informed of the possible leak. Operations subsequently properly performed system manipulations to allow specifics of the leak to be diagnosed. Recommendations resulting from the Engineering evaluations were promptly implemented.

G. Safety System Responses

N/A

III. CAUSE OF THE EVENT

A. Immediate Cause

The immediate cause of this condition was the failure of the metallic gasket at the heat exchanger floating head/tube sheet interface.

B. Root Cause

The root cause of this condition was raw water corrosion of the soft iron gasket and gasket seating surfaces at the heat exchanger floating head/tubesheet interface.

C. Contributing Factors

None

IV. ANALYSIS OF THE EVENT

There are four RHR trains on each BFN unit, with each train consisting of a pump and heat exchanger. The pumps provide LPCI flow to the reactor during accident scenarios, and the heat exchangers are used to reject reactor coolant or primary containment heat to the RHRSW system during both normal operations or accident conditions. Under most operating conditions, the operating pressure of the RHR system will be higher than that of the RHRSW system at the heat exchangers, therefore any leakage between the systems will flow from RHR into the RHRSW system. Radiation monitors capable of alarm are installed on the RHRSW piping to detect such leakage, and if the monitors are out of service, compensatory sampling is instituted. These features guard against the existence of sustained leakage from RHR into the RHRSW system through the RHR heat exchangers. For the condition under consideration, both the relevant radiation monitor and compensatory sampling indicated the existence of a small leak.

Following localization of the leak, the floating head gasket and gasket seating surfaces were examined. The soft iron gasket was oxidized with approximately eight inches of significant wastage in the area where leakage was observed during testing. The root cause was determined to be corrosion of the soft iron floating head gasket as a result of prolonged exposure in a wetted environment. In addition, the raw water corrosion resulted in some degradation of the gasket seating surfaces.

During the interval from 3/21 through 4/17 when the leak path alarmed the second time on high radiation, the focus of the work on the PER (all occurring in the midst of the many other activities of the U2C13 refueling outage) was to pursue possible failure modes within the suspected heat exchanger. During this outage period the possible design basis operational issues were not recognized by the plant staff. Systems personnel were aware that active programs were in place which monitor leakage and verify leakage is within specified limits and appropriate actions are taken to correct any deficiencies. In the case of the subject RHR heat exchanger leak, all Containment Leak Rate Program requirements, for example, could be satisfied, including performance of the containment integrated leak rate test which would directly measure all leakage from the primary containment (both air and water) and verify leakage was within limits. If the leakage to the environment was unmonitored (i.e., no radiation monitor in place), heightened scrutiny would have been prompted. Unmonitored paths from primary containment to the atmosphere do exist and program requirements are in place to address them. The hardened wetwell vent (HWWV) is one example; the discharge of the HWWV in the stack is downstream of the radiation monitors, leakage through this path is limited by the FSAR to 10 scfh, and compliance with this limit is verified by program procedures. The RHR heat exchanger leakage was not via an unmonitored release path and procedures were in place to address finding of radioactive material and to take necessary actions.

RHR heat exchangers 2B, 2C, 2D, 3A, 3B, 3C, and 3D were evaluated for the presence of leakage via an enhanced radiological sampling method. The utilization of this enhanced sampling technique made possible the detection of very small leaks in the these heat exchangers. No detectable leakage was found in these heat exchangers. The four RHR heat exchangers on BFN Unit 1 are not currently in service, and, therefore, these heat exchangers were not tested for leakage. The integrity of these heat exchangers will be verified, however, as part of the on-going return to power operation of Unit 1.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The RHR heat exchangers serve as a barrier between water from the reactor coolant system and primary containment and the cooling water which is discharged to the Tennessee River. During the timeframe of the heat exchanger leak, effluent activity was detected in concentrations well below the effluent concentration limits (ECL) and dose limits given in 10 CFR 20.

The BFN Unit 2 baseline core damage frequency (CDF) from all internal event causes is 1.25E-6 events/year. Over the 12-day interval where leakage to the environment could have occurred during a core damage event, the estimated event frequency is: 12/365 x 1.25E-6 = 4.1E-8 events. Stated as a probability, this corresponds to 1 chance in approximately 24 million of a core damaging event occurring within this 12-day span of time. Regarding large early release frequency (LERF), even if the conditional probability of LERF given a core damaging event is considered to be 1.0, the LERF value can be no

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

greater than 4.1E-8. Since the above frequency estimates include incidences of relatively small amounts of core damage, the probability of an accident involving significant core damage is therefore more unlikely still. However, if an event involving significant core damage is postulated to have occurred during this small timeframe, releases could have increased above the limits of 10 CFR 50.67 resulting in an unanalyzed condition. Considering the extremely low occurrence probability of any core damage event during this limited period of time, it is concluded that this condition had a negligible impact on the health and safety of the public.

There were no releases above regulatory limits, and, based on the above discussion, it is apparent there was no adverse safety impact of this event. There was no effect on the health and safety of the public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

The service water flow path through the 2A RHR Heat Exchanger was isolated. Planning was commenced for diagnosing the leak source and repairing the leak. During the repair activity the RHR 2A heat exchanger floating head gasket was replaced using a modified split ring and seal weld.

B. Corrective Actions to Prevent Recurrence[1]

- Revised maintenance procedures to include instructions for replacing the RHR heat exchanger floating head. (completed)
- Review RHR heat exchanger repair history on all three BFN units to determine dates of floating head gasket replacements and to identify heat exchangers with the original design soft iron floating head gasket installed. (completed)
- Estimate the rate of material loss for the RHR heat exchanger floating head gasket seating surface for predictive purposes. (completed)
- Develop a schedule for gasket and/or head replacement on the remaining BFN RHR heat exchangers. (completed)
- Implement the heat exchanger repair plan (in progress)

VII. ADDITIONAL INFORMATION

A. Failed Components

PERFEX (now Thermal Engineering, Inc.) CES 51-6-240 heat exchanger floating head gasket

B. Previous LERs on Similar Events

None. It is noted that BFN has previously incurred RHR heat exchanger leakage in events reported to NRC in the late 1970's and early 1980's, however, the leakage in these events resulted from issues of maintenance of bolting preload and not gasket material failure.

(1) TVA does not consider these corrective actions regulatory commitments. The completion of these actions will be tracked in TVA's Corrective Action Program.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Browns Ferry Nuclear Plant Unit 2	**05000260**	**2005**	**-- 004**	**-- 01**	7 OF 7

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

C. Additional Information

Browns Ferry corrective action documents PER 79143, PER 81236, and PER 83123

D. Safety System Functional Failure Consideration:

The condition being reported involves a safety system functional failure which was reported as a performance indicator data element in accordance with NEI 99-02. One safety function of the RHR system is to act as an extension of primary containment, and the condition identified herein represents a failure of this function.

E. Loss of Normal Heat Removal Consideration:

N/A. This condition being reported did not involve a reactor scram.

VIII. COMMITMENTS

None