UPDATE REPORT

PREVIOUS REPORT DATE 3/4/83

NRC FORM 366 (12-81) 10 CFR 50

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

APPROVED BY OMB 3150-0011

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1	LICENSEE CODE: A L B R F 1 (2)	LICENSE NUMBER: 0 0 - 0 0 0 0 0 0 0 0 (3)	LICENSE TYPE: 4 1 1 1 1 (4)	CAT (5)

CON'T

0 1	REPORT SOURCE: L (6)	DOCKET NUMBER: 0 5 0 0 0 2 5 9 (7)	EVENT DATE: 0 2 0 5 8 3 (8)	REPORT DATE: 0 6 3 0 8 3 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 Following reactor scram MSRV 1-1-22 apparently failed to close and drywell
0 3 unidentified leakage exceeded 5 gpm (T.S. 3.6.C.1). Unit 1 was placed in
0 4 cold shutdown; the vacuum relief valve for MSRV 1-1-22 was replaced. During
0 5 reactor heatup on 2/8/83, drywell unidentified leakage exceeded 5 gpm. Unit
0 6 1 was placed in cold shutdown. The vacuum relief valve for MSRV 1-1-22 was
0 7 found damaged. There was no effect on public health or safety. The remaining
0 8 vacuum relief valves were operable.

0 9	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
	C C (11)	B (12)	A (13)	V A L V E X (14)	X (15)	C (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 3	—	0 0 7	/	0 3	X	—	1

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER (26)
C (18)	F (19)	C (20)	A (21)	0 1 2 2	Y (23)	Y (24)	N (25)	G 2 0 2

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The 10 inch vacuum relief valve (GPE Controls Model #LF240-435) failed to
1 1 reseat. With 1-1-22 open, this resulted in leakage to the drywell. The
1 2 vacuum relief valve was replaced. The vacuum relief valves have been
1 3 modified on unit 2 and are to be modified during the present unit 1 and the
1 4 next unit 3 refueling outages. (See additional details on supplemental sheet).

1 5	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
	G (28)	0 0 0 (29)	NA	(31)	Surveillance Instruction

1 6	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

1 7	NUMBER	TYPE	DESCRIPTION (39)
	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

1 8	NUMBER	DESCRIPTION (41)
	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

1 9	TYPE	DESCRIPTION
	Z (42)	NA

8307120501 830630
PDR ADOCK 05000259
S PDR

IE 22 1/1

PUBLICITY

2 0	ISSUED	DESCRIPTION (45)
	Y (44)	Press release to media.

NRC USE ONLY

NAME OF PREPARER E. T. Holder

PHONE (205)-729-0885

Tennessee Valley Authority
Browns Ferry Nuclear Plant

Form BF 17
BF 15.2
2/12/82

LER SUPPLEMENTAL INFORMATION

BFRO-50- 259 / 83007 Technical Specification Involved 3.6.C.1

Reported Under Technical Specification 6.7.2.b.(2)* Date Due NRC 7/01/83

Event Narrative:

On 2/5/83, unit 2 was in a refueling outage, unit 3 was operating at 100 percent power, and unit 1 was being shutdown after a scram and subsequent failure to reseat of MSRV 1-1-22, when drywell unidentified leakage exceeded 5 gpm (Technical Specification 3.6.C.1). Reactor was placed in cold shutdown. Repairs were effected on MSRV 1-1-22 (see BFRO 50-259/83006). The vacuum relief valve for MSRV 1-1-22 was found opened and was replaced. Four other vacuum relief valves were found damaged and replaced.

On 2/8/83, during reactor heatup, drywell unidentified leakage again exceeded 5 gpm. Unit 1 was placed in cold shutdown. The vacuum relief valve for MSRV 1-1-22 was found damaged and was replaced. MSRV 1-1-22 was found stuck open (see BFRO 50-259/83006). In addition, the degree indicating arm on the vacuum relief valve was found to be bent and has caused the vacuum relief valve to be stuck open. The degree indicating arms were removed from the vacuum relief valves in unit 1 until these valves are modified during the next outage.

The vacuum relief valves on unit 2 were modified during the unit 2 cycle 4 refueling outage, unit 1 valves are to be modified this refueling outage, and the unit 3 vacuum relief valves are to be modified during the next refueling outage on the unit. This modification includes the installation of a new modified hinge arm, a new modified hinge shaft, a new modified hinge bearing and associated components, a spacer, and also the removal of the valve position indicators. The modification is required to prevent the damage that occurs to the vacuum relief valve when abnormal cycling of MSRVs occurs.

The failure analysis revealed that the suspected cause for damage to the vacuum relief valves are opening conditions that were not anticipated in the design process. Thus, the valves were procurred to operate under conditions less severe than are actually experienced (see attached NRC and failure evaluation).

The attached safety evaluation by Engineering Design provides justification for continued operation of unit 3 until the cycle 5 refueling outage. Modifications of these valves will be completed on all three units by the end of the unite 3 cycle 5 refueling outage.

Previous Similar Events :

None

Retention: Period - Lifetime; Responsibility - Document Control Supervisor

*Revision: JRP

DIVISION OF ENGINEERING DESIGN
NONCONFORMANCE REPORT

MEDS Accession No. '830310 8-

1 REPORT NO. BFNNEB8303

2 PLANT Browns Ferry	3 UNITS 1, 2, and 3
4 PREPARER/ORGANIZATION/DATE T. W. Barkalow/EN DES-NEB/March 3, 1983 3/10/83	

5 DESCRIPTION OF CONDITION

GPE vacuum breaker valves installed on the main steam safety/relief valve tailpipes have been unexpectedly damaged in service. The suspected cause is that the valves are being subjected to opening conditions that were not anticipated in the design process. Thus, the valves were procured to operate under conditions less severe than are actually experienced.

6 DATE OF OCCURRENCE EST ( ) ACT ( )	9 SIGNIFICANT CONDITION ADVERSE TO QUALITY 3/10/83 YES ☒ NO ☐
7 METHOD OF DISCOVERY Inspection	10 *BRANCH CHIEF/DATE /NEB 3-10-83
8 UNID CODE (EN DES-EP 8.01)	

11 CORRECTIVE ACTION

12 CORRECTIVE ACTION DEVIATES FROM A DESIGN CRITERIA REQUIREMENT YES ☐ NO ☐

13 DESIGN CRITERIA DOCUMENT NO EXCEPTION REQUEST NO.

14 ECN REQUIRED ☐ YES NO ☐ ECN NO

15 SCHEDULE IMPACT ☐ P ☐ A ☐

FAILURE EVALUATION   Date

By Tom Barkalar   3/16/83

Checked By [signature]   3/16/83

Concurrence: M-5 Dan W. Wilson   3/17/83

Group Head RCWeir   3/17/83

NCR BFNNEB8303 (copy attached)

The deficiency identified by the NCR renders the affected system(s):

_____ Still qualified for all design loading combinations and design conditions.

_____ Not qualified for some design loading combination or design condition. However, a functional impairment of the system(s) is not likely.

__X__ Not qualified for some design loading combination or design condition. A system(s) failure or functional impairment is likely. If this item is checked, complete the following two items of additional information.

Identify the critical loading combination or condition: Repeated SRV actuations. Damage accumulates with each SRV actuation.

Identify the type and location of the failure or functional impairment: Vacuum breaker valve disc fails to seat or separates from hinge. Tailpipe integrity inside drywell lost. SRV steam discharge can enter drywell rather than being condensed in torus. This was experienced on Unit 1.

Justification for the above judgment:
Unit 1 vacuum breaker valve on SRV tailpipes showed damage on 6 of 13 valves. Valve showing worst damage was near total failure with disc unscrewing from hinge arm. The associated SRV had been actuated about 6 times during the cycle 4 operating period.

Comments and/or recommendations (optional):
Vacuum breaker valves were replaced on Unit 1. Unit 2 valves were modified prior to end of refueling outage. Unit 3 experienced damage to at least one valve that was repaired. Units 1 & 3 presently operating with unmodified valves.

See attached safety evaluation (NEB 830318221)

R 0

BROWN'S FERRY NUCLEAR PLANT

SAFETY EVALUATION REPORT

TITLE: NCR BFNNEB 8303
SRV Tailpipe Vacuum Breaker Damage

SYSTEM: Main Steam

APPLICABLE UNITS: 1 and 3

SYSTEM NUMBER: 1

REV NO.	TOT PP	PREPARED	REVIEWED	APPROVED	DATE APPD
0*	13	Tom Barbalato	*Alan G. Gilson	RCWeir	3/18/83
1					
2					
3					
4					
5					

*RO=INITIAL ISSUE

*JKR

MEDS ACCESSION NO.	
R0	NEB '830318 221
R1	
R2	
R3	
R4	
R5	

Statement of Problem:

To determine the safety impact of loss of pressure boundary in a main steam safety/relief valve (SRV) tailpipe due to structural failure of a vacuum breaker valve.

References:

1) Attached "Management Alert" from N.R. Beasley to D.W Wilson dated 2/15/83
2) Conversations with Elvis Hollins (NUC PR-Mech. Br.), W.R. Brock (NEB-NAL), J.K. Rochelle (CEB), and S.R. Lawson (BWP)
3) NCR BFNNEB8303 (NEB 830316 85)

Evaluation:

Each main steam safety/relief valve (SRV) has a tailpipe that routes the SRV discharge to beneath the torus water. The tailpipes contain a 2½" and a 10" vacuum breaker. The 2½" vacuum breaker is a Powell swing check valve and is part of the original design. The 10" vacuum breaker is a GPE vacuum breaker valve with a 7" disc and is a portion of the long term torus structural modifications. The function of these two vacuum breakers is to admit drywell air into the tailpipes after SRV discharge steam has been condensed. This prevents a permanent partial vacuum from forming in the tailpipe and minimizes the

water leg that develops when ~~[illegible]~~ a temporary partial vacuum is formed. The GPE vacuum breaker valve was added to provide increased relief capacity so as to minimize the water leg that can develop. Unfortunately, the pressure drop in the tailpipe due to condensation of the steam once the SRV closes is apparently so rapid that the GPE vacuum breaker valve flies open and impinges upon the inner pipe wall. The resultant impact loads damage the GPE valve. Each time ~~the~~ a SRV actuates and recloses, its associated GPE vacuum breaker will open and suffer impact loads. Damage appears to accumulate with repeated impacts. Damaged GPE vacuum breaker valves have been found in units 1 and 3. Per the attached "Management Alert" dated 2/15/83 and conversation with Elvis Hollins (NUC PR - Mech. Br.), the most damaged GPE valve on unit 1 was near total failure (disc was unscrewing from hinge arm) and its associated SRU had been actuated about 6 times.

The failure modes identified are two:

1) Valve retains integrity, but fails to seal and permits steam leakage into drywell whenever ~~the~~ its SRV discharges; and,

2) Valve loses integrity with disc separating

from hinge arm.

Failure During Normal Operation or Transients

For either identified failure mode, vacuum relief will be provided. Either failure mode could result in reduced vacuum relief capacity if the disc lodged in such a manner as to reduce the effective flow area of the valve. In this case the water leg could rise to a higher level and exist longer than assumed in analyses. If the SRV associated with the damaged vacuum breaker valve reactuated when the water leg was at its highest point, the subsequent loads imposed on the tailpipe could be higher than previously analysed. Lodging of the valve disc such as to provide significant reduction of the vacuum relief capacity is considered highly unlikely as the disc is smaller in diameter than the tailpipe and no credible mechanism for lodging it in the closed position is foreseen. For either identified failure mode, steam leakage into the drywell when the SRV actuates will be increased. The event that ~~lead~~ led to the discovery of the damaged GPE vacuum break valves involved sufficient leakage to the drywell to be detected and to warrant investigation. Assuming the GPE valve failed in such a way that the disc did not impede flow in either direction through the valve, then the discharge from the associated SRV would enter the drywell

rather than be condensed in the torus. This would occur due to the path resistance being less for flow to the drywell than for flow through the T-quenchers. Under such conditions everytime the associated SRV actuated, significant quantities of steam will enter the drywell. For very brief actuations (typical of proper actuation of an SRV due to temporary overpressure conditions following anticipated transients such as turbine trip), the steam released into the drywell will condense without deleterious safety consequences. For prolonged actuations such as may result from a deliberate blowdown or a stuck open SRV, the steam released into the drywell will pressurize the drywell and raise the drywell atmospheric temperature. The conditions created will approximate those caused by a $0.15\ ft^2$ steam break. This size steam break belongs to the intermediate break accident (IBA) category. An IBA results in high drywell temperature conditions which require actuation of containment spray for mitigation. An IBA also results in the highest analytical dynamic loads on the torus structure due to drywell pressure relief via the downcomers at gradually increasing flows (which are conducive to chugging). IBA's have been extensively analyzed and torus

structural modifications made specifically for the IBA analytical loads. Even with the torus structural modifications in a partially implemented state (as presently exists on all three units), enough modifications have been achieved to withstand IBA analytical loads. Actions required to mitigate an IBA have been well defined. Thus BFN is designed for and the operators are trained to mitigate IBA's. There are two key differences between analyzed IBA's and steam release to the drywell due to a failed vacuum breaker valve:

1) IBA's are assumed to be permanent while an SRV can be ~~redos~~ reclosed (it's a sealable break), and,

2) IBA steam is released directly to the drywell atmosphere where as SRV discharge steam expands into the tailpipe and then into the drywell via the failed vacuum breaker valve. This slightly reduces the total energy imparted to the drywell atmosphere. These differences tend to make the failed vacuum breaker case less severe than an IBA. However, if another vacuum breaker valves or valves failed during the event, the consequences could be made as severe as ~~the~~ an IBA. In any case the failed vacuum breaker with prolonged SRV actuation is bounded

by existing analyses and is within the capability of the plant to mitigate.

It is also possible for the SRV to simmer at a given pressure. A constant or near constant volume of steam will be released in amounts much less than from a fully open SRV. If this steam or a portion of it is released to the drywell via a leaking or failed vacuum breaker valve, the consequences are much less severe than for a fully open SRV. The steam will be condensed without deleterious safety consequences. However, continued plant operation may be hindered due to gradually increasing drywell pressure and/or Tech. Spec. limits on leakage inside primary containment. This type event did occur on unit 1 in early February 1983.

For the failure mode where the disc separates from the hinge arm, it is conceivable that the disc could enter the tailpipe. If the loose disc settled in such a manner as to come loose upon a subsequent discharge, it could be propelled by the steam and become a missile within the tailpipe. Such an internal missile could produce impact loads due to impingement at pipe elbows, especially if an elbow is at the end of a lengthy near horizontal run. A loose disc could eventually enter the

T-quencher where it would then remain. Blockage of T-quencher holes by a loose disc would not present a problem as the vast majority of the steam would be entering the drywell. The impact loads due to impingement could damage the tailpipe and tailpipe supports if the loads were very high. The worst case damage would be rupture of the tailpipe inside the torus airspace. Such a rupture could create a severe loading condition on the torus structure if the SRV was not closed and torus containment spray was not actuated to condense the steam entering the torus airspace. Less severe than this case would be damage to the tailpipe supports that would lead to tailpipe deformation without loss of integrity. While the tailpipe would no longer be qualified under code allowables, preservation of pressure boundary integrity would assure the torus structure would not be damaged even if further SRV actuations occurred.

Impact loads were discussed with J.K. Rochelle (CEB). According to him the disc weighs approximately 10.6 pounds and must be accelerated ~~to~~ to a velocity of about 250 ft/sec prior to impact in order to produce loads that would yield a tailpipe support. The longest horizontal tailpipe run is roughly 50 feet. The disc must be accelerated by 506 ft/sec^2 to achieve 250 ft/sec within 50 feet. Considerable force would be required to accomplish this. The only credible mechanism available is the force of the SRV discharge steam. Only steam flowing through

NEB '830318 221

the tailpipe to the T-quencher will affect a loose disc. As stated earlier the majority of the SRV discharge steam will enter the drywell. This reduces the steam flow that will affect a loose disc. Since the disc has a diameter of 7.65" and the tailpipe inner diameter is 9.564", a considerable gap will exist between the disc edge and the tailpipe wall even if the disc is perfectly perpendicular to the tailpipe wall (in this position it would expose the greatest surface area to the steam flow). This gap permits steam to escape around the disc and reduces the force exerted on the disc. After considering these factors, consulting the Crane Company pipe flow manual, and discussing the situation with W.R. Brock (NEB), conservative engineering judgment is that the disc will not have enough force exerted on it to achieve 250 ft/sec. Thus it is concluded that a loose disc will not result in tailpipe rupture or damaged tailpipe supports.

The last concern to address is tailpipe response to reaction forces caused by steam exiting into the drywell during an SRV discharge. If the vacuum breaker valve disc separates from the hinge arm, the resulant opening forms a crude nozzle (square edge orifice at the end of a short pipe). Steam flow through the opening during an SRV discharge forms a stream that creates a thrust and impingement loads. Per discussions with J.K. Rochelle (CEB), the resultant loads will not damage the tailpipe, tailpipe supports, or impinged equipment. Thus the exiting steam

will not present a safety problem.

In summary all failure consequences during normal and transient plant conditions are bounded by analyzed events. However, the consequences of a stuck open SRV are increased if the vacuum breaker valve fails open. Also, the probability the drywell will experience high temperature conditions and IBA type loadings is increased as the likelihood of a stuck open SRV with a failed vacuum breaker valve is higher than a pipebreak or critical crack.

Failure During Accident Conditions

The previous discussions pertaining to vacuum relief capability, separation of the disc leading to a loose disc in the tailpipe, and steam jet loadings are applicable to failure during ~~design basis~~ loss of coolant accident conditions. However, the effect of steam leakage into the drywell is different.

From the viewpoint of reactor water level and vessel pressure responses, SRV actuation during an SBA (small break accident) or IBA (intermediate break accident) is the same regardless of whether the vacuum breaker ~~x~~ remains open. However, when considering the torus structural loadings, it does ~~t~~ make a difference whether steam enters the torus via a downcomer or a T-quencher. All structural loading conditions have been analyzed

assuming a pipebreak inside the drywell that is of a fixed size. Entrance of steam into the drywell via a failed vacuum breaker valve when the associated SRV is open produces an effect as though the pipebreak size changes. For example, an SBA can become an IBA and then return to an SBA when an SRV opens and closes. The effect of this has not been included in previous analyzes.

The situation can be made more complicated if multiple vacuum breaker valves fail during the accident (since the forces opening the vacuum breaker valves increases during SBA/IBA conditions, the valves become more likely to fail and may fail with fewer SRV actuations). For example, an SBA occurs. Upon containment isolation SRV's life at their pressure setpoints and reset. A vacuum breaker valve fails during this. Subsequent SRV actuations created IBA type conditions due to the failed vacuum breaker valve and ~~fails~~ further vacuum breaker valve failures occur. A blowdown occurs (either manual or ADS) and steam enters drywell from several failed vacuum breaker valves creating ~~LOCA~~ DBA (main steam break) type conditions inside primary containment.

Needless to say such a situation has never been investigated nor analyzed. According to J.K. Rochelle (CEB) the highest torus structural loadings are associated with an IBA. The torus response to an SBA is made more severe if an actuating SRV creates IBA or DBA type conditions. The torus response to an DBA bounds this situation.

NEB '830318 221 Sheet 11

The torus response to an IBA is not made more severe if an actuating SRV adds more steam to the drywell. The worst case analysis still bounds this event. No other analyzed events are made more severe than the worst case analysis due to vacuum breaker valve failure. The drywell pressurization, reduction in steam flow to the T-quencher of the failed vacuum breaker valve tailpipe, and altered thermal load were all considered. Since the situations considered are unanalyzed, the ~~possibility of exceeding~~ resultant loading situations ~~beyond~~ as compared to analyzed bounds cannot be ~~completely ruled out~~ strictly defined. However, conservative engineering judgment is that the unanalyzed events are bounded by the worst case analysis.

In summary all failure consequences during accident conditions are judged to be bounded by worst case analysis. However, the consequences of an SBA or IBA would be made worse due to vacuum breaker valve failure.

Conclusions

1) The unmodified GPE vacuum breaker valves will experience damage whenever SRV's actuated;
2) Damage accumulates with additional SRV actuations, with about 6 actuations resulting in severe damage under normal operations and probably fewer actuations under LOCA conditions;
3) Separation of the valve disc from the hinge arm on a severely damaged valve could occur (possibility cannot be precluded;
4) The consequences of GPE vacuum breaker valve

NEB '830318 221

failure are judged to be bounded by analyzed events and worst case analyzes; and,

5) The probability and consequences of particular events/conditions are increased due to use of the unmodified GPE vacuum breaker valves. However, as stated above, all consequences are judged to be bounded by analyzed events (i.e. the units can successfully withstand and mitigate the valve failure consequences).

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
1750 Chestnut Street Tower II

83 JUL 5 A9:47

June 30, 1983

Mr. James P. O'Reilly, Director
U.S. Nuclear Regulatory Commission
Suite 2900
101 Marietta Street, NW
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-259 - FACILITY OPERATING LICENSE DPR-33 - REPORTABLE OCCURRENCE REPORT BFRO-50-259/83007 - REVISION 1

The enclosed report is a supplement to my letter dated March 4, 1983, concerning unidentified leakage exceeding 5 gpm following a reactor scram. This report is submitted in accordance with Browns Ferry unit 1 Technical Specification 6.7.2.b(2).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

Hubert Abercrombie

for H. J. Green
Director of Nuclear Power

Enclosure
cc (Enclosure):
Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Records Center
Institute of Nuclear Power Operations
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

NRC Inspector, Browns Ferry

OFFICIAL COPY

IE22

1/1