CP&L

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MAR 31 1993

FILE: B09-13510C 10CFR50.73
Serial No: BSEP-93-0049

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

BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2
DOCKET NO. 50-325 AND 50-324
LICENSE NO. DPR-71 AND DPR-62

LICENSEE EVENT REPORT 1-92-023 SUPPLEMENT TWO

AND

NOTIFICATION OF A 10CFR21 REPORTABLE OCCURRENCE

Gentlemen:

In accordance with Title 10 of the Code of Federal Regulations, the enclosed Licensee Event Report supplement is submitted. This report is submitted in accordance with the format set forth in NUREG-1022, September 1983. A supplemental information section has been added to satisfy the 10CFR21 notification requirements.

Very truly yours,

J M Brown for

C. C. Warren, Plant Manager - Unit 2
Brunswick Nuclear Plant

SFT/

Enclosure

cc: Mr. S. D. Ebneter
Mr. P. D. Milano
BSEP NRC Resident Office

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NRC FORM 366 (5/92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Unit 1	05000325	1 of 16

TITLE (4)
Local Leak Rate Test Failure of Both RCIC Steam Line Isolation Valves

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 NAME	DOCKET NUMBER
09	27	92	92	- 23 -	002	03	31	93	BSEP Unit 2	05000324
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 04
POWER LEVEL (10): 00

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)		OTHER (Specify in Abstract and Text)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)	X	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
Steve F. Tabor, Regulatory Compliance Specialist	(919) 457-2178

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	JM	ISV	A391	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE)		NO		12	31	93

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

On September 26, 1992, with Unit 1 in Cold Shutdown, Local Leak Rate Testing (LLRT) of the Reactor Core Isolation Cooling (RCIC) outboard steam isolation valve, 1-E51-F008, identified valve seat leakage in excess of acceptable limits. An LLRT of the RCIC inboard steam isolation valve, 1-E51-F007, performed on August 3, 1992, similarly identified excessive leakage past the valve seat. Consequently, with both Primary Containment Isolation (PCIS) valves failing to seat, the potential existed for loss of the Primary Containment Isolation capability for the affected penetration had the unit been operating. An inspection of the 1-E51-F007 was performed following the LLRT. The inspection did not reveal the cause for the excessive leakage. The 1-E51-F007 valve discs were replaced due to a previously identified concern with binding of this type of valve in the open position. A post maintenance LLRT was performed which verified 0 SCFH leakage. Although initial inspection of the 1-E51-F007 did not reveal the cause of the excessive leakage, a root cause evaluation of this failure is continuing. Additionally, an investigation into the cause of the 1-E51-F008 leakage is in progress. Failure of both PCIS valves to seat on the same 3 inch line is significant and represents a loss of the PCIS capability for that penetration. A previous similar event is addressed within LER 1-91-016. Subsequent valve testing has identified additional failures of 3", 4", 6" and 10" Anchor/Darling double-disc gate valves. Cumulatively, these failures are considered 10CFR21 reportable. A supplemental information section is included to address the 10CFR21 notification requirements.

Supplement 2 provides additional information as contained in the Safety Assessment and the Summary of Root Cause Investigation sections.

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

TITLE

Local Leak Rate Test Failure of Both RCIC Steam Line Isolation Valves

INITIAL CONDITIONS

On September 26, 1992, Unit 1 was in Cold Shutdown in day 158 of a maintenance outage. To satisfy Technical Specification (TS) containment leakage rate surveillance requirements, Technical Support engineering personnel had commenced leak testing of the RCIC outboard steam isolation valve, 1-E51-F008. The reference testing is accomplished by closing the valve, pressurizing the disc/seat area to 50 psig, and measuring the makeup flow to the valve. The normal acceptable leakage rate for the associated penetration is 3 scfh.

EVENT NARRATIVE

On August 3, 1992, a LLRT of the RCIC Steam Inboard Isolation Valve, 1-E51-F007, was conducted. The test results indicated that the valve would not pressurize to the desired 50 psig test pressure. On August 13, 1992, troubleshooting was initiated. Following valve disassembly, the valve internals were inspected and the valve disc blue checked. The inspection did not reveal a condition to which the excessive leakage could be attributed. The blue check verified 98% disc to seat contact. Due to a previously identified generic concern with the Anchor Darling double-disc gate valve (i.e., valve disc sharp edges resulting in jamming in the partially-open position when opened against differential pressure), the valve disc was replaced. On September 25, 1992, a post maintenance LLRT verified the 1-E51-F007 leakage to be 0 scfh.

On September 26, 1992, a LLRT of the RCIC Steam Outboard Isolation Valve, 1-E51-F008, was conducted. The test results indicate that valve leakage was 252.07 scfh.

Based on the results of the 1-E51-F008 LLRT and having previously identified an excessive leakage rate while testing the 1-E51-F007, a ENS Notification was made at 1235 on September 27, 1992.

Investigation into the cause of the 1-E51-F008 LLRT failure is currently in progress. The vendor is assisting in this investigation. Preliminary results of the investigation indicate that disc wedge irregularities may be contributing to this problem. Technical Support and Nuclear Engineering Department personnel are pursuing completion of the root cause for failure of both of the RCIC steam isolation valves. Corrective actions to prevent recurrence will be established as part of the root cause investigation.

CAUSE OF EVENT

An investigation into the cause of the RCIC steam isolation valve leakage is in progress. An action item has been assigned to Technical Support to complete the investigation and develop/implement corrective actions to prevent recurrence.

CORRECTIVE ACTIONS

The RCIC Steam Inboard Isolation Valve, 1-E51-F007 has been repaired based on the current understanding of the cause of the problem. LLRT results verified that the valve leakage following maintenance is within the acceptable TS limit. However, the on-going investigation of the 1-E51-F008 may identify additional failure modes not recognized at

the time of the 1-E51-F007 repair. Consequently, additional 1-E51-F007 corrective maintenance may be required.

Additional corrective actions resulting from the completion of the root cause investigation will be documented in a supplement to this report.

SAFETY ASSESSMENT

The LLRT failure of both the 1-E51-F007 and 1-E51-F008 valves resulted in a leakage of 252 SCFH. An analysis was performed to determine the safety significance of this condition during the High Energy Line Break (HELB) and the Loss of Coolant Accident (LOCA) scenarios. The following provides the results of this analysis.

HELB:

The RCIC steamline isolation valves are designed to close in the event of a RCIC steamline break in secondary containment to prevent an uncontrolled loss of reactor coolant. Since a source term is not created by a RCIC steam line pipe break, a leak tight isolation of these valves is not critical. CP&L has addressed this position in a response to Supplement 3 to Generic Letter 89-10.

During a RCIC steam line break or leak outside of primary containment, the 1-E51-F007 and 1-E51-F008 will automatically close due to high temperature in the RCIC steam line area and/or 300% steam flow. These valves will immediately isolate when 300% steam flow is detected; however, isolation due to high RCIC area temperature is delayed for 30 minutes in accordance with system design. As addressed by the Reactor Building Environmental Report (RBER), the most limiting case for a RCIC steam line break or leak occurs during a RCIC small break condition. For the RBER analysis, the RCIC steam line is assumed to leak at 299% of normal steam flow for 30 minutes prior to isolation. A steam flow of 25 lb/sec (1500 lb/min.) into the reactor building is assumed for the full 30 minutes.

Comparison of the HELB leakage and the expected leakage from the subject valves at a rate proportional to that experienced during the LLRT revealed that the amount of leakage from the valves is not significant. The 252 SCFH at 49 psid equates to a water leakage of 3.654 gpm at 1010 psid. This amount of leakage is lower than the Technical Specification limit of 5 gpm unidentified drywell leakage. Additionally, this leakage is approximately two orders of magnitude lower than the 30 minute RCIC leak assumed in the RBER and is expected to have an insignificant impact on the post-HELB reactor building environment.

The preceding leakage comparison analysis was performed assuming a leak rate of 252 SCFH following a HELB. Although the subject valves exhibited significant leakage during the LLRT, significant leakage is not expected following a HELB. The conditions contributing to the LLRT failure of the subject valves is expected to occur only when the valves are closed against low differential pressure. During a HELB these valves would be expected to close against a differential pressure of at least 1000 psid. Under these conditions the differential pressure assists the valve discs in achieving a good seal. This characteristic was demonstrated in June of 1991 when the subject valves were found leaking against a low differential pressure, and yet were capable of isolation under operational conditions.

Based on the above, the LLRT leakage from the 1-E51-F007 and 1-E51-F008 valves is not considered safety significant during a RCIC steamline HELB.

LOCA:

During a LOCA, RCIC will initially inject to the vessel in an attempt to raise vessel water level. When vessel pressure drops to approximately 62 psig, RCIC will trip and the subject valves will close to isolate the RCIC steam line. Since the drywell will be pressurized and a source term present, the subject valves will be closing for PCIS purposes. During a LOCA the subject valves will close against low differential pressure resulting in a potential leak rate of 252 SCFH. The off-site dose calculations in section 15.6.4 of the Update Final Safety Analysis Report (UFSAR) assume that primary containment leaks 0.5 percent of the contained free volume per day to secondary containment. This equates to a primary containment leakage of approximately 266 SCFH into secondary containment. When the cumulative LLRT leakage from the other containment penetrations is added to the 252 SCFH leakage resulting from the subject valves, the total leakage would exceed that assumed in the UFSAR analysis.

A review of the potential leak paths out of the RCIC system was performed to understand the safety significance of this event. Leakage through the subject valves could be contained within the RCIC piping, released into the reactor building, or released to the condenser through the E51-F025 and E51-F026 valves. The most safety significant case is that of a release to the condenser. Assuming no line losses, the 252 SCFH could be released to the condenser and then to the environment. The General Electric Company performed an evaluation of the environmental impact of a 400 SCFH total Main Steam Line Valve leak to the Condenser. The results of this analysis indicated that an additional 400 SCFH leak would not increase the off-site doses above the 10CFR100 limits. Since the 252 SCFH is less than the 400 SCFH assumed in the analysis, off-site effects are not considered safety significant. Although this analysis did indicate off-site dose within 10CFR100 limits, the control room allowable thyroid dose limits as defined by the General Design Criteria (GDC) 19 could potentially be exceeded. Based on the potential for exceeding the GDC-19 limits, this event is considered potentially safety significant.

PREVIOUS SIMILAR EVENTS

A previous similar event involving failure of both RCIC steam isolation valves to seat is documented in LER 1-91-016. The cause of that event was attributed to atypical accumulation of corrosion products.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
PCIS	JM
RCIC	BN
1-E51-F007	BN/ISV
1-E51-F008	BN/ISV

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

SUPPLEMENTAL INFORMATION

Both Units have been in cold shutdown since notification of the failure of the RCIC Steam Isolation Valves 1-E51-F007 and 1-E51-F008 to meet local leak rate testing requirements. Subsequent LLRTs of Anchor/Darling double-disc gate valves have identified four Unit 1 and three Unit 2 containment isolation valves which failed to meet acceptable leakage limits. To date, the defects described herein have been identified in six double-disc gate valves. Initial NRC notification of these defects was made on November 13, 1992. Attachment One provides a list of the types of deficiencies identified.

The basic components identified as containing the defects contributing to the LLRT failures are:

3-inch, 900 lb Class, Double-Disc Gate Valves
4-inch, 900 lb Class, Double-Disc Gate Valves
6-inch, 900 lb Class, Double-Disc Gate Valves
10-inch, 600 lb Class, Double-Disc Gate Valves

Although deficiencies have been identified primarily in the 3", 4", 6", and 10" valves, these deficiencies are considered potentially present in all Anchor/Darling double-disc gate valve sizes.

The basic components identified as containing the defects were supplied by:

Anchor/Darling Valve Company
701 First Street
PO Box 3428
Williamsport, PA 17701

The valves listed below identify the Anchor/Darling double-disc gate valves installed at the Brunswick Nuclear Plant (BNP). With the exception of the 1/2-E41-F001 valves, these valves provide a primary containment isolation function.

B21-F016/F019	Main Steam Drain Valve (3-inch)
E41-F002/F003	HPCI Steam Isolation Valves (10-inch)
E41-F006	HPCI Injection Valves (14-inch)
E41-F001	HPCI Steam Admission Valves (10-inch)
E51-F007/F008	RCIC Steam Isolation Valves (3-inch)
E51-F013	RCIC Injection Valves (4-inch)
G31-F001/F004	RWCU Inlet Isolation Valves (6-inch)

The following is a discussion of the double-disc gate valve deficiencies which have been identified as contributing to the LLRT failures. Some of the deficiencies were identified by inspections occurring in September of 1991 and at that time were considered isolated occurrences. However, based on the number and types of deficiencies identified during the recent outage and the similarity of these deficiencies to those identified in 1991 as addressed in the Brunswick Information Report dated December 27, 1991, BNP has recognized the need to report these deficiencies in accordance with the requirements of 10CFR21.

Uneven Stanchion Length On The Lower Wedge

The bottom of the lower wedge has two stanchions as shown on Figure 1. The purpose of these stanchions is to stop lower wedge motion during valve closure, so that the advance of the upper wedge into the lower wedge will result in normal load against the in-body seats. To provide even wedge

expansion and consequently an even seating load distribution, the stanchions should be the same length.

During an inspection of the 1-G31-F001 in September of 1991 and the recent inspections of the 1-E51-F008 and the 2-G31-F001 valves, the associated lower wedge stanchions were identified to be shorter than the other stanchion. If not accounted for by adjusting the lower wedge geometry, uneven stanchion lengths cause the lower wedge to cock as the valve seats. Any cocking of the lower wedge results in uneven contact with the upper wedge during seating. This uneven contact causes the wedge pieces to load predominately on only one incline which results in uneven transmission of the stem wedging force to the discs. This contributed to the failure of the valves to seal when subjected to the low LLRT pressure.

Casting Flaws In The Wedge Surfaces

Inspections of the 1-E51-F008, 2-G31-F001, and 2-G31-F004 identified discontinuities in the cast surface of the upper wedges. The 2-G31-F001 and 2-G31-F004 valves were found to have a casting flaw (raised metal) on the upper portion of the upper wedge (see Figure 2). These casting flaws prevented the upstream disc from sitting flush against the machined surface of the upper wedge. Although the upstream disc does not provide a seal against differential pressure, these flaws may have prevented proper wedging and uniform loading of both the upstream and downstream discs. Additionally, these casting flaws caused gouges on the inside edge of the upstream discs.

The 1-E51-F008 valve was found to have a sharp "lip" on the upper wedge where the wedge begins to taper (See Figure 3). During wedging, this lip would contact a slight protrusion on the lower wedge which would cause binding of the two wedges. This binding could cause an uneven distribution of the seating force to the valve discs and subsequent seat leakage.

Non-uniform Contact Of The Upper And Lower Wedges

Non-uniform contact between the upper and lower wedges has been identified on the 1-B21-F016, 1-E51-F008, 2-E51-F013, 1-G31-F001 (in September 1991), 2-G31-F001 and 2-G31-F004 valves (See Figure 4). This non-uniform contact results in an uneven distribution of the seat force and potential seat leakage.

The non-uniform contact between the upper and lower wedges has resulted from several different causes. The mating surfaces between the upper and lower wedges are hand-ground surfaces. This process may result in low spots or slightly different wedge angles which cause uneven contact and wedging. The uneven contact for the 1-E51-F008 valve may have been caused by a poor lower wedge casting.

Improper Centering of the Valve Stem

During recent inspection of the 1-E41-F002 valve, the valve stem was found not to be centered in the valve when the discs were seated. This condition was caused by the stem connection in the upper wedge not being properly centered. On 4-inch and larger double-disc gate valves, the stem is rigidly threaded into the upper wedge. With a misaligned stem, the upper wedge and stem must deflect when the valve is seated. This condition may cause seat leakage and galling of the valve stem on the bonnet bore. Improper valve stem centering was also observed to exist on the 1-E41-F001 valve.

Improper Wedge Orientation

During an inspection of the 2-G31-F001 valve in September of 1991, it was determined that the valve was supplied with the upper wedge installed on the downstream side of the valve. Although the Anchor/Darling Valve Company maintains that these valves should provide bi-directional sealing, there is a preferred wedge orientation. Since the upper wedge is rigidly connected to the stem, the wedge cannot self-align to the seat. Therefore, any non-parallelism between the line of stem action and the seat face or misalignment between the upper wedge face and the seat face cannot be accommodated. Consequently, this results in a non-uniform distribution of contact pressure and potential leakage. Since the lower wedge is free to "float", this wedge should always be installed on the downstream or sealing side. The installation of the upper wedge on the downstream side of the 2-G31-F001 valve also resulted in the stem not being centered as described in the preceding discussion of improper valve stem centering.

In addition to the aforementioned defects, BNP has identified a concern with Anchor/Darling double-disc gate valves larger than 6-inches. These valves have a stellite hardfacing on the upper and lower wedge mating surfaces. Non-uniform contact between the upper and lower wedges has resulted in localized loading and subsequent cracking of the stellite. Although this stellite cracking has not resulted in LLRT failures, it is of concern since pieces of stellite could have broken off during valve operation. Cracking of the stellite hardfacing with the associated cracks propagating into the base metal has been observed on the wedges of the 1-E41-F001 and 1-E41-F006 valves.

These manufacturing defects in the Anchor/Darling double-disc gate valves may result in significant seat leakage when subjected to low differential pressures. Excessive seat leakage on both isolation valves for a given containment penetration or excessive leakage on one valve coincident with a single failure of the redundant valve could result in a loss of the PCIS safety function.

Corrective Actions

Each double-disc gate valve installed in a PCIS application has been or will be local leak rate tested at least twice during the current outages. Since the noted defects can result in inconsistent seating and sealing of the valves under low-pressure conditions, the second leakage test is performed to demonstrate repeatability. Valves which fail one of these local leak rate tests have been or will be disassembled and repaired as necessary. For the defects noted in this 10CFR21 report, the repair typically involves custom grinding of the valve wedges or replacement of the wedges to achieve proper contact and uniform distribution of the seating load. Once the necessary corrective actions are implemented, each repaired valve will receive at least two post-maintenance local leak rate tests. The second leakage test is performed to demonstrate repeatability and the adequacy of the repair. BNP believes that repeat testing, whether performed pre-or-post repair, provides adequate assurance that the defects noted to date would be identified.

The corrective actions to eliminate the noted defects will be completed prior to startup of Units 1 and 2. Startup of Unit 2 is currently scheduled for March 28, 1993. Startup of Unit 1 is expected to occur in the second quarter of 1993.

Technical Support will establish the appropriate controls for ensuring that future inspection/repair of Anchor/Darling double-disc gate valves will include an inspection for the deficiencies noted herein.

The final root cause for the failure of these LLRT failures including the 1-E51-F007 and 1-E51-F008 will be completed by February 26, 1993. A supplement to this LER reflecting the final root cause is expected to be submitted by 3/31/93.

Industry Recommendation

Based on BNP's experience to date with Anchor/Darling double-disc gate valves, BNP believes the noted defects can result in unpredictable and unrepeatable LLRT results. With the noted defects present, it may be possible to successfully perform one leakage test and fail a subsequent test after the valve has been stroked. Consequently, to ensure repeatability, it may be necessary to perform redundant leak rate tests.

Summary of Root Cause Investigation

During the current unit outages, six of ten Unit 1 and three of ten Unit 2 Anchor Darling double disc gate valves failed to meet LLRT acceptable leakage limits. Additionally, one Unit 2 valve demonstrated inconsistent LLRT results. The root cause of these LLRT failures is primarily attributed to manufacturing deficiencies in the valve wedge pieces. Additionally, low spots in the seats were identified as contributing to some of the failures.

The following provides the problems identified, the corrective actions implemented, and the root cause of the failure for each of the Anchor Darling double-disc gate valve LLRT failures which occurred during the current Unit 1 and Unit 2 outages:

Unit 1 Failures:

1-B21-F016:

The problems identified with the 1-B21-F016 valve included minor seat pitting, low spots on the downstream in-body seat, non-uniform point of contact between the upper and lower wedges due to rough wedge finishes, and stem galling 2 to 3 inches above the backseat. The LLRT failure of the 1-B21-F016 valve is attributed to the low spots in the downstream seat and the uneven contact between the upper and lower wedges. The uneven contact between the upper and lower wedges resulted from poor vendor manufacturing practices. The low spots in the in-body seats are believed to have resulted from thermally induced stresses incurred during reactor heatup and cooldown. Temperature changes from reactor heatup and cooldown may have provided some stress relief resulting in minor changes to the valve seat geometry, i.e., low spots. The stress relief phenomenon represents a hypothesis of how the low spots were formed. This hypothesis has not been confirmed by actual insitu or prototype testing. Corrective actions taken include replacement of the galled stem and both the upstream and downstream discs. The in-body seats were lightly lapped and narrowed. The wedges were custom ground to achieve even contact. Two post maintenance local leak rate tests were successfully performed prior to returning the valve to service.

1-E41-F002:

The problems identified with the 1-E41-F002 valve included low spots in the downstream in-body seat, wedge centerline fit-up problems resulting in a non-centered stem in the bonnet bore, and spider web cracks in the upstream disc stellite overlay. The LLRT failure of the 1-E41-F002 is attributed to the low spots in downstream in-body seat. The cause of the low spots is the same as that addressed in the preceding paragraph. The improper centering of the valve stem in the bonnet bore is a contributing factor in the LLRT failure of this valve. The improper centering of the valve stem is due to a manufacturing

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deficiency resulting in misalignment of the valve body seats and the valve stem with the valve center line. The cracking of the upstream disc stellite overlay is not considered a contributor to the LLRT failure of this valve. Although a definitive reason for the cracking has not been established, recent examination supports the hypothesis that the cracking could have been caused by overstressing of the discs into the seats or thermal shock. Corrective actions taken include lapping of the upstream and downstream seats to remove the low spots. The cracked disc on the upstream side of the valve was replaced. To correctly center the valve stem the valve discs were switched. Two post maintenance local leak rate tests were successfully performed prior to returning the valve to service.

1-E41-F003:

The problems identified with the 1-E41-F003 valve included uneven contact between the upper and lower wedges and cracking of the upstream valve disc stellite overlay. The LLRT failure of the 1-E41-F003 valve is attributed to the uneven contact of the upper and lower wedges. This uneven contact resulted in the tilting of the upper wedge and valve stem and galling of the valve stem. Tilting of the upper wedge is believed to have caused the lower wedge to tilt during valve seating resulting in a loading of the wedge pieces on one incline. This resulted in a non-uniform transmission of the seating force to the discs and, consequently, seat leakage. The cause of the stellite overlay cracking is the same as that addressed in the preceding paragraph. Corrective actions for this valve are not complete. The procurement of new machined wedges to eliminate the uneven wedge contact is in progress. The cracked upstream disc will be replaced. The valve will be local leak rate tested at least twice prior to returning the valve to service.

1-E51-F008:

The problems identified with the 1-E51-F008 valve included a bent stem, the existence of a sharp lip on the upper wedge, a skewed and slightly twisted lower wedge, and uneven lower wedge stanchions. The LLRT failure of the 1-E51-F008 is attributed to the deficiencies with the upper and lower wedges. The sharp lip on the upper wedge is believed to have caused the upper wedge to bind with the lower wedge during seating. This binding resulted in an uneven distribution of the valve disc seating force and the subsequent leakage. The skewed and twisted lower wedge and the lower wedge uneven stanchion lengths is believed to have resulted in an uneven distribution of the seating force and subsequent seat leakage. Corrective actions included replacement of the valve stem and wedges. The discs were machined and polished to eliminate the sharp edges. Two post maintenance local leak rate tests were successfully performed prior to returning the valve to service.

1-E51-F007/1-B21-F019:

Disassembly and investigation into the cause of the LLRT failures of the 1-E51-F007 and 1-B21-F019 valves have not been performed. These valves will be disassembled and inspected to support a root cause determination prior to the startup of Unit 1. The results of this determination will be reported in a supplement to this LER.

Unit 2 Failures:

2-E51-F013:

The LLRT failure of the 2-E51-F013 valve is attributed to the uneven contact between the upper and lower wedges resulting in tilting of the lower wedge during seating. The uneven contact was caused by a shallower wedge angle on one side of the upper wedge. The tilting resulted in the non-uniform transmission of the seating force to the discs and the

inconsistent seating of the valve. Corrective actions to achieve uniform contact included replacement of the original wedges with new custom fit wedges. Two post maintenance local leak rate tests were successfully performed prior to returning the valve to service.

2-G31-F001:

The problems identified with the 2-G31-F001 valve included downstream in-body seat low spots, upper wedge casting flaws, and uneven stanchions on the bottom of the lower wedges. The 2-G31-F001 valve LLRT failure is attributed to the low spots in the downstream in-body seat. The cause of the low spots is addressed in the preceding discussion of the 1-B21-F016. Additional factors contributing to the 2-G31-F001 valve LLRT failure include the upper wedge casting flaws and the uneven lower wedge stanchion lengths. The effects of the uneven lower wedge stanchion lengths is addressed in the preceding 1-E51-F008 valve discussion. The upper wedge casting flaws prevented the upstream disc from sitting flush against the machined surface of the upper wedge. Although the upstream disc does not provide a seal against differential pressure, these flaws may have prevented proper wedging and uniform loading of the upstream and downstream discs. The casting flaws and uneven lower wedge stanchion lengths would result in inconsistent seating of the valve. Corrective actions included polishing of the discs and lapping of the downstream in-body seat to remove the low spots, replacing the upstream disc damaged by the casting flaws on the upstream wedge, and replacing and custom fitting the upper and lower wedges. Two post maintenance local leak rate tests of the 2-G31-F001 were successfully performed prior to returning the valve to service.

2-G31-F004:

The problems identified with the 2-G31-F004 valve included a galled stem above the backseat. upper wedge casting flaws, and uneven contact between the upper and lower wedges. The LLRT failure of the 2-G31-F004 valve is attributed to the upper wedge casting flaws and the uneven contact between the upper and lower wedges. The effects of the upper wedge casting flaws is discussed in the preceding paragraph. The effects of the uneven contact are addressed in the discussion of the 1-E51-F003. Corrective actions included replacement of the upstream disc, blending of the stem galling, and the custom grinding of the upper and lower wedges to achieve uniform contact. Two post maintenance local leak rate tests were successfully performed prior to returning the valves to service.

2-B21-F019:

During initial local leak rate testing of the 2-B21-F019 valve, the valve passed two consecutive local leak rate tests. However, the results of these tests were not consistent and consequently the valve received an internal inspection and root cause investigation. The problems identified with the 2-B21-F019 valve included poor mating of the wedges resulting in the rocking of the lower wedge. The rocking is attributed to differences in the thicknesses of the wedge sliding surfaces. This condition resulted in the uneven distribution of forces at the seats and can resulted in inconsistent local leak rate test results. The corrective actions included widening of the seats on the downstream mating surface and the dressing of the upper wedge sliding surface until the top-to-foot dimensions were equal.

Additional Corrective Actions:

During the investigation to determine the cause of the deficiencies contributing to the LLRT failures of the Anchor-Darling double disc gate valves installed at BNP, the vendor has assisted in the repair and trouble shooting of the defective valves. Based on the problems witnessed by the vendor, the vendor recognizes that their manufacturing techniques

warrant improvement. The vendor is currently assessing changes to the manufacturing process which could prevent recurrence of those problems experienced at BNP. An example of this is evidenced by the recent change in the manufacture of the upper and lower wedges. In the past, Anchor-Darling has custom fit each disc by hand grinding the mating surfaces of the upper and lower wedges. This process has resulted in inconsistent seating when the valves are closed. Anchor/Darling has recently changed its manufacturing process such that the wedge mating surface are machined instead of hand ground. This enhancement should result in more reliable valve operation. To ensure that future wedge replacements utilize the enhanced wedges, BNP is currently procuring the new wedges.

To ensure continued reliability of the Anchor/Darling double disc gate valves, each double disc gate valve used to satisfy a PCIS function will receive two local leak rate tests during the next Unit 1 and Unit 2 refuel outages.

The repair of the 1-E41-F003 and the disassembly, inspection, and repair of the 1-E51-F007 and 1-B21-F019 will be completed prior to Unit 1 startup.

The final supplement to this LER will be submitted following the Unit 1 startup to address the results of the inspection and repair of the 1-E51-F007 and 1-B21-F019 valves.

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

ATTACHMENT ONE

Summary of Deficiencies Identified Per Valve

(This table identifies the double disc gate valve deficiencies identified to date either during the current outage or during previous refuel outages.)

Valve	Deficiencies Identified
1-B21-F016	3,7,8
1-E41-F001	4,6,9
1-E41-F002	4, 8, 10
1-E41-F003	3, 10
1-E41-F006	6
1-E51-F008	1,2,3,7
1-G31-F001	1,3
2-B21-F016	7
2-B21-F019	7
2-E51-F007	7
2-E51-F008	7
2-E51-F013	3
2-G31-F001	1,2,5,8
2-G31-F004	2,3

Deficiencies

1 - Uneven Stanchion Length on Lower Wedge
2 - Casting Flaws in Wedge Surfaces
3 - Non-uniform Contact of the Upper and Lower Wedges
4 - Improper Centering of the Valve Stem
5 - Improper Wedge Orientation
6 - Cracked Stellite on Wedges
7 - Sharp Edges on Discs
8 - Low Spots in Seats
9 - Disc Scraping Body Casting
10 - Stellite seat face cracking

Lower Wedge

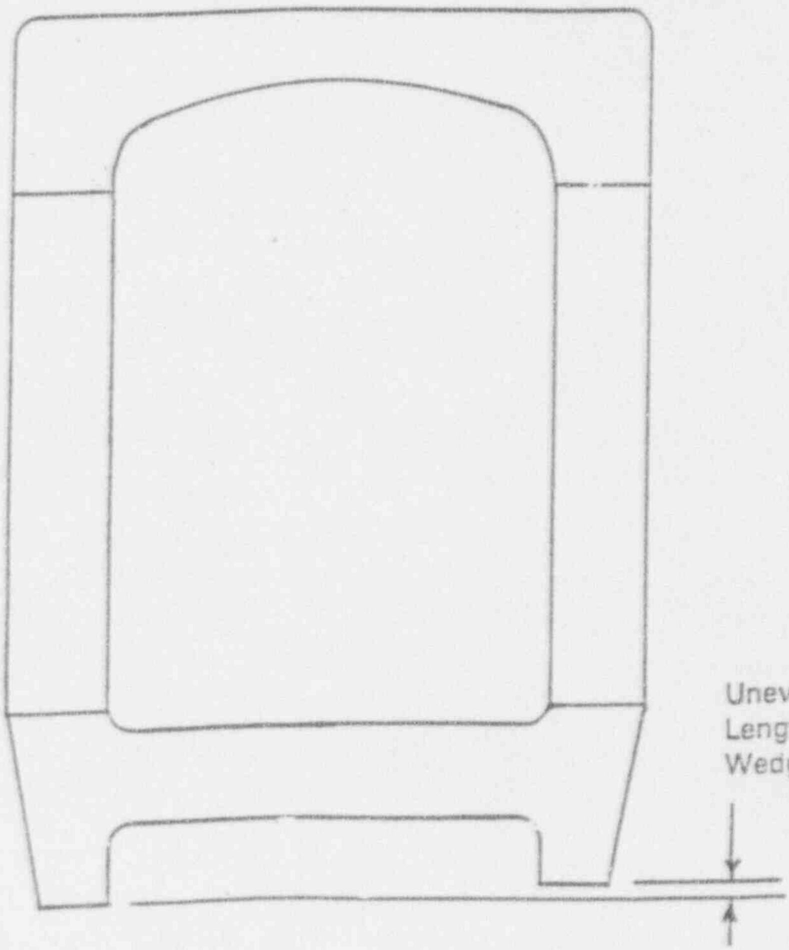

Figure 1

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

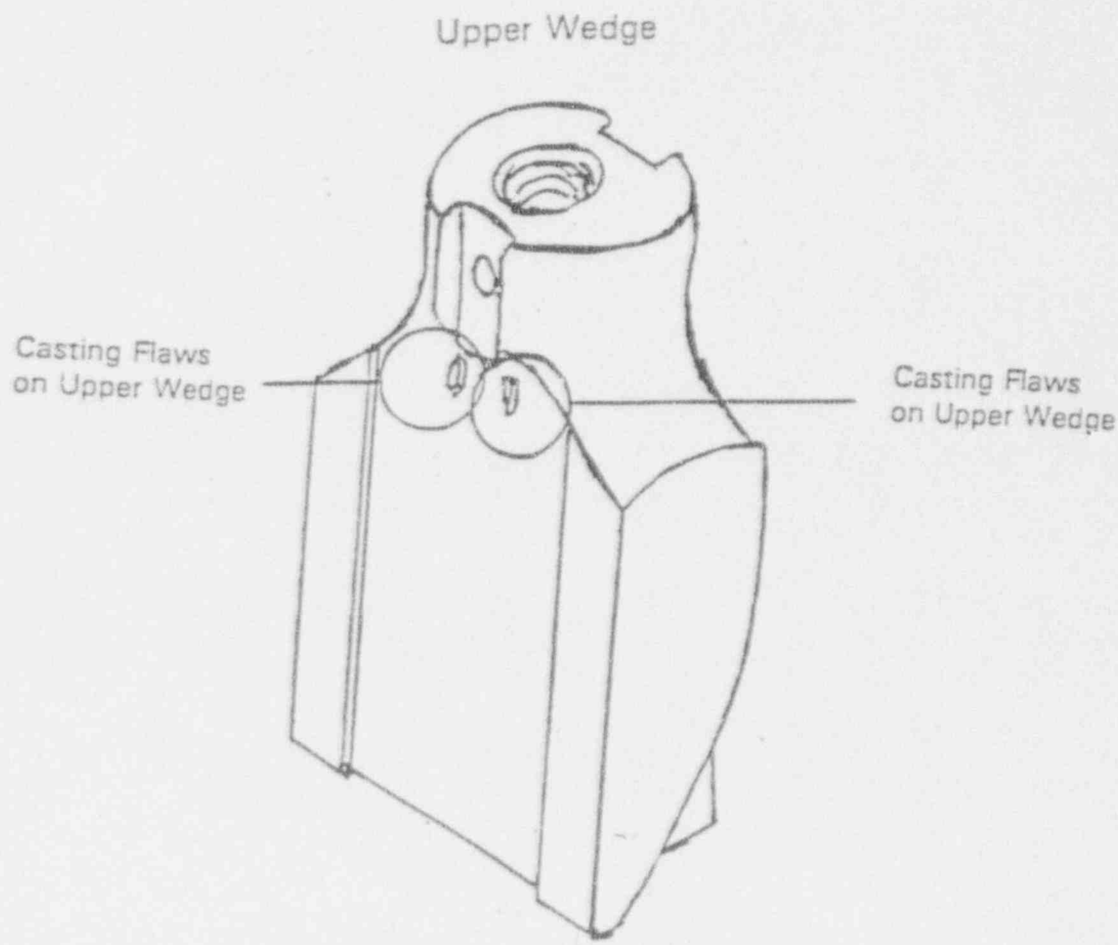

Figure 2

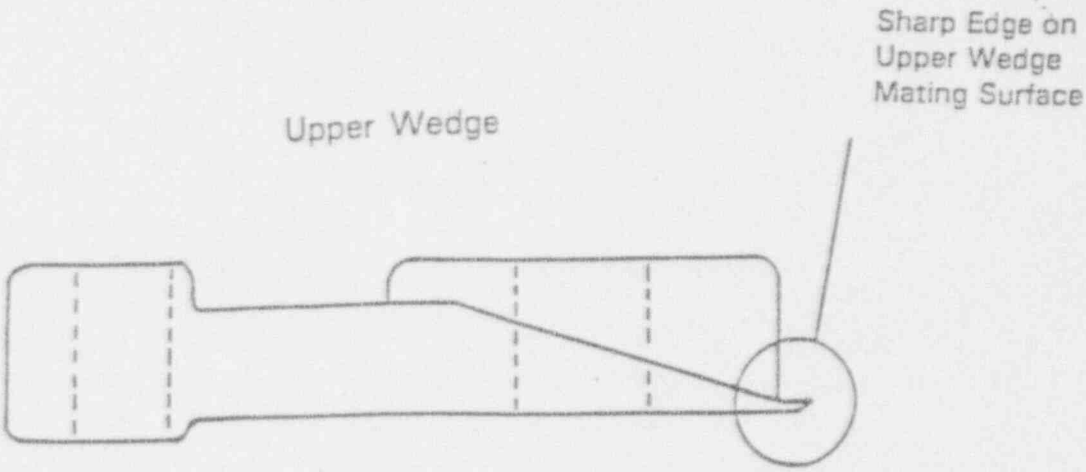

Figure 3

NRC FORM 366A (5/92)	U. S. NUCLEAR REGULATORY COMMISSION	APPROVED OMB NO. 3150-0104 EXPIRES: 5/31/95
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant Unit 1	05000325	92	- 23 -	002	16 of 16

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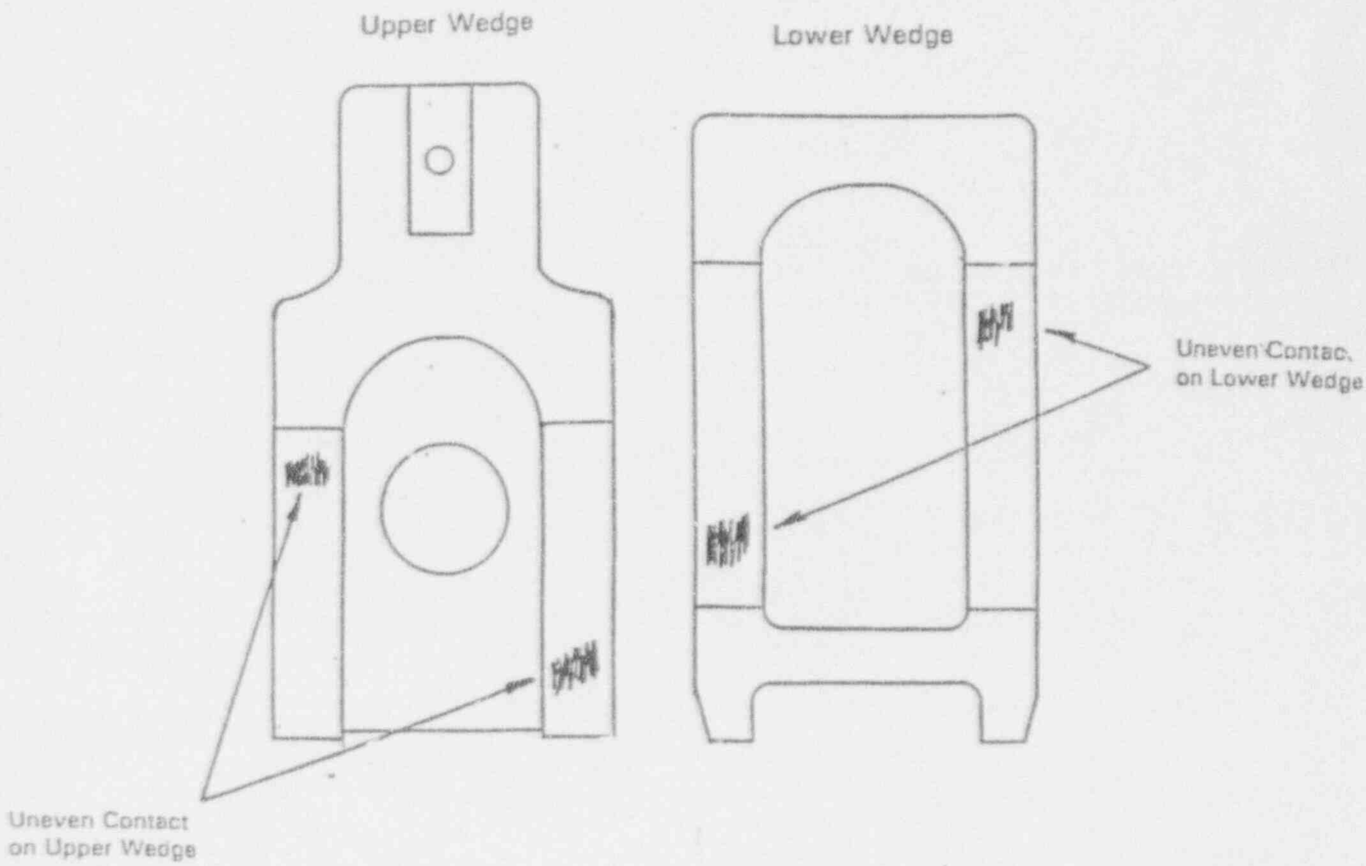

Figure 4