NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

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ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), 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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Nuclear Generating Plant Unit 2	05000 306	1 OF 8

TITLE (4)
Defect in Primary System Pressure Boundary Observed on the Motor Tube Base of Part Length CRDM Housing

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
02	01	98	98	-- 02 --	02	5	24	99	Prairie Island Unit 1	05000 282
									FACILITY NAME	05000

OPERATING MODE (9): 5
POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	√ 50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jeff Kivi	651-388-1121

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

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE).	√ NO	EXPECTED SUBMISSION	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

Prairie Island Unit 2 was shut down on January 24, 1998 to repair a small leak on the reactor coolant system. The unit was taken to the cold shutdown condition for the repair. The source of that leakage appeared initially to be the intermediate canopy seal weld for the part length control rod drive mechanism (CRDM) at location G9 on the reactor vessel head. However, the canopy seal was covered and couldn't be directly observed. Filling and venting the reactor coolant system was initiated while the repair of the intermediate canopy seal weld was still in progress. During that fill and vent operation it was noted that water was leaking from the G9 part length CRDM approximately one inch above the intermediate canopy seal that was being repaired.

Both Unit 1 and Unit 2 part length CRDM's have been removed and replaced with Head Adapter Plugs which were screwed on and seal welded. Root cause analysis indicates the flaw on the G9 part length CRDM originated during fabrication and is an isolated case.

EVENT DESCRIPTION

Prairie Island Unit 2 was shut down on January 24, 1998 to repair a small leak on the reactor coolant system. The unit was taken to the cold shutdown condition for the repair. The source of that leakage appeared initially to be the intermediate canopy seal weld for the part length control rod drive mechanism (CRDM) at location G9 on the reactor vessel head. However, the canopy seal was covered and couldn't be directly observed. Filling and venting the reactor coolant system[1] was initiated while the repair of the intermediate canopy seal weld was still in progress. During that fill and vent operation it was noted that water was leaking from the G9 part length CRDM approximately one inch above the intermediate canopy seal that was being repaired.

Because of the location of the affected CRDM, accessibility to the CRDM is limited. A video camera was utilized to investigate the leakage. Examination of the leak location using the video camera identified water dripping from an apparent flaw in the wall of the CRDM Motor Tube Base above the intermediate canopy seal. Filling and venting was secured and the reactor coolant system was drained to a level below the leak elevation. Subsequent investigation revealed that the leak was from a through-wall crack in a weld between the CRDM Tube Base and the CRDM Motor Tube.

Because the part length CRDMs are not used and are abandoned in place, it was decided that the part length CRDM at location G9, would be permanently removed. This would eliminate the need for repair of the flaw and would facilitate the metallurgical evaluation of the flaw. Removal of the G9 part length CRDM was completed on February 8, 1998. The remaining three part length CRDMs on Unit 2 and all four part length CRDMs on Unit 1 were removed and capped. Caps have been screwed onto the threaded end of the penetration and seal welded with a 5/8 inch fillet weld.

CAUSE OF THE EVENT

A defect in the primary system pressure boundary observed on the Motor Tube Base of Unit 2 part length CRDM location G9 caused the RCS leak. Westinghouse investigated the cause of the defect and prepared WCAP - 15054, "Metallurgical Investigation and Root Cause Assessment of Part Length CRDM Housing Motor Tube Cracking at Prairie Island Nuclear Generating Plant Unit 2." The report has been submitted to the NRC by letter dated May 15, 1998. In summary, the report concludes that:

(1) The defect in Unit 2 part length CRDM from location G9 originated from weld fabrication.

(2) There is no evidence of additional service-related growth of the defect.

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[1] (EIIS System Identifier: AB)

(3) No other defects were found (by metallographic examination) in five other part length CRDM motor tube welds from Prairie Island (nor did UT examination identify any other similar defects in some 60 part length motor tube welds in the industry).

(4) Introduction of contaminants (such as, sulfur, boron, copper, and zinc from an unknown external source during fabrication) most likely contributed to the defect in the Unit 2 part length CRDM from location G9.

(5) The defect in the Unit 2 part length CRDM from location G9 is an isolated event.

ANALYSIS OF THE EVENT

UNIT 2:
The defect in the Unit 2 primary system pressure boundary was detected as leakage, based on increased RCS leak rate and containment radiation. The leak rate increased 0.2 gpm and indicated radiation doubled on Unit 2 containment radiation monitors[2].

The potential extent of this indication was reviewed considering other Unit 2 locations and also Unit 1. The part length CRDM design is different than the full length CRDM housings. The part length CRDM Motor Tube Bases were fabricated from multiple pieces, welded together, and machined to final geometry. The full length CRDM housings do not contain any welds in the latch housings or rod travel housings, other than the canopy seal welds. Thus, the part length CRDMs have welds relied on to maintain structural integrity of the primary system pressure boundary, while the full length CRDMs have only seal welds.

Original receipt inspection documentation for the part length CRDM motor tubes was reviewed. This documentation indicates that the finished weld was radiographed and liquid penetrant examined. The original radiographs were not available at Prairie Island. The part length CRDMs were hydrostatically tested by the fabricator and again by NSP as part of original construction. The base material, weld wire and weld butter layer material heat information was reviewed. The part length CRDM motor tube base with the defect, G9, was welded with a unique heat number (ladle batch) end clad buttering. Other heat numbers for base metal are similar to Unit 1 and others on Unit 2, except G5 on Unit 2. Unit 2 location G5 used a unique heat number for the motor tube base transition piece and end clad buttering wire. Unit 2 locations I7 and E7 used all the same material as on Unit 1 (One part length CRDM on Unit 1 does use a unique weld butter heat number). Therefore, a decision was made to remove the I7 and E7 part length CRDMs sequentially, so NDE results could be more definitively correlated to Unit 1.

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[2] (EIIS Component Identifier: RI)

Additional information from the defect part length CRDM motor tube base from location G9 was obtained on February 9. An unqualified ultrasonic (UT) examination was performed after the part length CRDM was removed from the reactor head. The preliminary UT results showed two inside diameter indications, one approximately five inches long and a second three inches long. The UT method used could not locate the indication in relation to the thickness of G9, and the UT method could not measure the indication depth. The leak location was centered in the five inch long indication.

On February 10, several radiographs (RT) were performed to gain additional information on the preliminary UT indications. The RT results supported the UT results with respect to the length of the five inch indication. No depth information was available from RT data. The three inch indication identified via the preliminary UT was not identifiable via RT. Boric acid seepage on the exterior of the motor tube base at the leak location indicated the exterior crack was approximately 3/4 inch long.

A second weld exists on each part length CRDM tube. This weld is above the motor tube center section, approximately two feet above the defect area. This weld on the G9 CRDM was also UT examined to help determine if the defect in the motor tube base was of a generic nature. The UT on this upper weld showed no apparent discontinuities.

Additional information was obtained after removal of the second part length CRDM motor tube from location I7 on February 12. The motor tube base and the upper weld were UT examined. These UT examinations on I7 showed no apparent discontinuities. The I7 CRDM was cut into shorter sections to reduce radiation exposure and remove internal components from the area of interest at the motor tube base. Internal diameter visual inspections with 10X magnification and internal fluorescent dye PT inspections were conducted on I7. No relevant linear indications were found on the internal diameter of part length CRDM I7.

Additional information was obtained February 14 after removal of the last two part length CRDMs from locations E7 and G5. The motor tube bases and the upper welds of E7 and G5 were UT examined. Both motor tube bases and upper welds for E7 and G5 showed no apparent discontinuities. The CRDMs were cut into shorter sections to remove internal components from the area of interest at the motor tube base. Internal diameter visual inspections were conducted. No relevant linear indications were found on the internal diameter of part length CRDMs E7 or G5.

Westinghouse labs performed a 20-25X magnification visual examination on the inside diameter surface of part length CRDM motor tube base from location G9. The visual examination found a continuous circumferential indication for 120 degrees around the leak location. A UT inspection on the defect G9 piece was performed at Westinghouse labs. The UT results indicate circumferential cracking 360 degrees around the inside diameter surface.

The final root cause analysis (Westinghouse WCAP - 15054) was submitted to the NRC by letter dated May 15, 1998. The conclusions of this report are summarized in the section above labeled "Cause of the Event."

UNIT 1 (operating unit):
Considering a potentially generic extent, the RCS leak rates, containment sump pump run times, and containment radiation monitor trends were reviewed for Unit 1 immediately upon determining that the leak in the Unit 2 part length CRDM at location G9 was not in the canopy seal weld. Stable trends were found on all parameters.

Unit 1 reactor head was inspected in December of 1997 for leaks upon startup from 1997 refueling outage. No leaks were noted. The inspection procedure specifically requires a signoff for satisfactory inspections at the intermediate canopy seal weld locations.

Parameters and trends were monitored daily as part of normal operating practice. This monitoring included: daily RCS leak rate and trends, daily review of containment radiation monitor trends, monitoring of containment humidity, monitoring of containment sump run times, monitoring of the containment fan coil units condensate drains, monitoring of radiation monitor alarms, and computer radiation monitor alarms.

The RCS leak rate trend is reviewed by obtaining a printout of the results of the last forty RCS leak rate calculations, and sump run times. The radiation monitor trends are reviewed by obtaining a printout of the last week's monitor count rates. Minor trend changes can be observed by reviewing these results. The monitoring described above was successful in identifying the leak on Unit 2. Specifically, a increase in daily leak rate in combination with increasing radiation monitor levels keyed operators into notification of plant management. This level of monitoring was deemed sufficient on February 1.

Enhanced monitoring of the RCS leak rate and radiation monitors started February 9 (formal instructions were effective February 10). Operations personnel were instructed to increase the frequency of the RCS leak rate calculation to once each shift. This also included review of the leak rate and sump run time trends. Containment radiation monitors 1R11 and 1R12, and containment humidity were also put on continuous trend display in the control room. Containment activity trend increases would be indicative of RCS leakage.

Surveillance was also increased on the Unit 1 Containment Fan Coil Units condensate collection tank levels. These levels indicate leakage into the containment atmosphere that is being condensed by the cooling coils. Operations personnel were instructed to inform the Operations General Superintendent if the RCS leak rate should increase above 0.1 gpm, or any increasing trends on the monitored parameters.

A containment inspection was conducted February 11. The inspection included a visual inspection of Unit 1 reactor head. No leaks were noted from this inspection.

Documentation was reviewed verifying operations personnel were trained recently on LOCA accidents. Requalification training completed on June 27, 1997 included review of the LOCA emergency response procedure and related simulator scenarios. Additionally, recent requalification examinations included LOCA simulator scenarios on 9 of 24 tests. No weaknesses were noted with respect to operator responses to LOCAs during these examinations.

A review was conducted of the potential consequences of a postulated failure of a Unit 1 part length CRDM housing. A catastrophic failure of the housing could theoretically lead to a small break Loss of Coolant Accident (SBLOCA) as large as 4 inches. The more likely scenario would be that the internal components of the drive assembly (i.e. the thrust bearing retainer assembly) would restrict the flow to something much smaller than a 4 inch break. However, the assessment conservatively assumes that the break is 4 inches.

A review of the licensing basis SBLOCA confirms that a break of 4 inches is bounded by the limiting cold leg break of 6 inches. The results in WCAP-13920 Small Break Loss-of-Coolant Accident Engineering Report for the Prairie Island ZIRLO Fuel Upgrade show that the peak cladding temperature for a 4 inch cold leg break is approximately 400°F lower then the limiting 6 inch cold leg break, and that for Prairie Island the limiting 6 inch cold leg break has approximately 1000°F margin to the limit of 2200°F. Since Westinghouse's approved LOCA methodology has determined that a SBLOCA in the cold leg is more limiting than a break in the hot leg, and since a failure of a rod drive housing would be more similar to a hot leg break, it is also reasonable to conclude that a catastrophic failure of the rod travel housing is bounded by our licensing basis analysis. Additional information related to accident analysis has previously been provided in NSP Response to Request for Additional Information Concerning Prairie Island Unit 2 Control Rod Drive Mechanism Leak, dated February 23, 1998.

Another issue considered is the effect of a missile caused by the postulated failure of a CRDM housing. This issue is addressed in the Updated Safety Analysis Report (USAR) section 3.5.4.1.3, which states that the kinetic energy of the broken off piece would be dissipated in the missile shield. The USAR goes on to conclude that the circumferential failure would not cause damage to adjacent housings that would increase the severity of the initial accident. All of the above information leads to the conclusion that the postulated failure of a rod travel housing on Unit 1 would be bounded by the safety analyses in the USAR. Additional missile effects information have been previously submitted in NSP Response to Request for Additional Information Concerning Prairie Island Unit 2 Control Rod Drive Mechanism Leak, dated February 23, 1998.

The root cause analysis (documented in Westinghouse WCAP - 15054 submitted to the NRC by letter dated May 15, 1998) concludes that the defect in the Unit 2 part length CRDM from location G9 is an isolated occurrence. Thus, we believe that there are no similar defects in the Unit 1 part length CRDM housings. However, as it was deemed prudent to do so, the four part length CRDMs on Unit 1 have been replaced with Head Adapter Plugs which are screwed on and seal welded.

Because the circumferential defect in the G9 Motor Tube Base was (for a short distance) 100% through wall and resulted in a breach of the primary system pressure boundary, this event is reportable per 10CFR50.73 (a)(2)(ii).

CORRECTIVE ACTION

1. Remove and inspect all four Unit 2 part length CRDMs. This action is complete.

2. Implement a design change to cap the penetrations for all four Unit 2 part length CRDMs. This action is complete.

3. A notice has been put out to other nuclear plants through Nuclear Network.

Four related commitments were made in NSP Response to Request for Additional Information Concerning Prairie Island Unit 2 Control Rod Drive Mechanism Leak, dated February 23, 1998. One of these commitments was, "Northern States Power, at a minimum, will perform additional inspections of the Prairie Island Unit 1 part length CRDM housing welds at the next scheduled refueling outage, which is currently scheduled to begin in April 1999."

The four Unit 1 part length CRDMs have been replaced with Head Adapter Plugs which are screwed on and seal welded. This closes the last remaining open commitment related to this event.

Based on the conclusions of the root cause analysis (documented in Westinghouse WCAP 15054, submitted to the NRC by letter dated May 15, 1998), no additional NRC commitments, beyond those already made, are deemed necessary at this time.

FAILED COMPONENT IDENTIFICATION

Leakage through the primary system pressure boundary was caused by a through-wall defect in the Unit 2 part length CRDM Motor Tube Base transition weld at location G9.

The Unit 2 G9 Motor Tube Base contains the only defect identified to date. A summary of the UT inspections on the part length CRDMs for Unit 2 shows only G9 Motor Tube Base with apparent defects. G9 upper weld, a warehouse spare, and all other Unit 2 part length CRDMs show no apparent discontinuities. A total of ten areas were examined between the upper welds and motor tube bases for the four Unit 2 part length CRDMs and the spare part length CRDM in the warehouse.

As documented in Westinghouse WCAP - 15054, submitted to the NRC by letter dated May 15, 1998, some 60 other part length motor tube welds inspected in the industry identified no other instances of cracking similar to that identified on the Unit 2 part length CRDM from location G9.

PREVIOUS SIMILAR EVENTS

None.