UPDATE REPORT - PREVIOUS REPORT DATE 5/19/80

NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

EXHIBIT A

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

01 ARAN01 (2) 00-000 00-00 (3) 4 1111 (4) (5)

LICENSEE CODE — LICENSE NUMBER — LICENSE TYPE — CAT

CON'T 01 REPORT SOURCE L (6) 0500 0313 (7) 0510 80 (8) 0711 7180 (9)

DOCKET NUMBER — EVENT DATE — REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

During steady state power operation, the operations staff observed a step increase in RCS leakage, as indicated by an increased rate of change of Makeup Tank level. Reactor Coolant Pump 'C' seal instrumentation confirmed that a problem existed with the seal or associated piping. A controlled unit shutdown per T.S.3.1.6. was initiated with operator actions governed by Small Break procedure, O.P.1202.06, Sect. II. No similar occurrences. Reportable per T.S.6.12.3.1.c.

09 SYSTEM CODE CB (11) CAUSE CODE E (12) CAUSE SUBCODE B (13) COMPONENT CODE PUMPXX (14) COMP. SUBCODE B (15) VALVE SUBCODE Z (16)

(17) LER/RO REPORT NUMBER: EVENT YEAR 80 — SEQUENTIAL REPORT NO. 015 — OCCURRENCE CODE 01 — REPORT TYPE X — REVISION NO 1

ACTION TAKEN A (18) FUTURE ACTION X (19) EFFECT ON PLANT A (20) SHUTDOWN METHOD A (21) HOURS (22) 0715 ATTACHMENT SUBMITTED Y (23) NPRD-4 FORM SUB Y (24) PRIME COMP SUPPLIER N (25) COMPONENT MANUFACTURER B580 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

Following an orderly shutdown of the unit, a failure investigation was initiated which included: examination of failed seal, good seals, and spare seal, compilation of seal failure history, assessment of probable failure mechanisms; and ultimately, provide recommendations to reduce probability of recurrence. Seal supplier (B&W) and vendor (Byron Jackson) is assisting in investigation.

FACILITY STATUS E (28) % POWER 086 (29) OTHER STATUS (30) NA METHOD OF DISCOVERY A (31) DISCOVERY DESCRIPTION (32) Operator Observation

ACTIVITY RELEASED G (33) CONTENT OF RELEASE M (34) AMOUNT OF ACTIVITY (35) 2844.7 Ci of Xe-133 LOCATION OF RELEASE (36) Reactor Building vent to Atmosphere (Controlle

PERSONNEL EXPOSURES NUMBER L00 (37) TYPE E (38) DESCRIPTION (39) 25 Man-Rem (Estimated)

PERSONNEL INJURIES NUMBER 000 (40) DESCRIPTION (41) NA

LOSS OF OR DAMAGE TO FACILITY (43) TYPE B (42) DESCRIPTION Damage to RCP 'C' Seal

PUBLICITY ISSUED Y (44) DESCRIPTION (45) News Release, 5/12/80

NRC USE ONLY

NAME OF PREPARER Chris N. Shively PHONE: 501/968-2519

8007240150

1. Reportable Occurrence Report No. 50-313/80-015/01-1

2. Report Date: 7/17/80 3. Occurrence Date: 5/10/80

4. Facility: Arkansas Nuclear One - Unit 1
Russellville, Arkansas

5. Identification of Occurrence:

Abnormal degradation discovered in reactor coolant pressure boundary, due to total reactor coolant leakage rate exceeding 10GPM, defined by T.S.3.1.6.1.

6. Conditions Prior to Occurrence:

Steady-State Power	X	Reactor Power	2208 MWth
Hot Standby		Net Output	672 MWe
Cold Shutdown		Percent of Full Power	86 %
Refueling Shutdown		Load Changes During Routine Power Operation	
Routine Startup Operation			
Routine Shutdown Operation			

Other (specify)

Operations personnel conducting RCS leak rate calculation at time of occurrence.

7. Description of Occurrence:

At 0145 on May 10, 1980, while at approximately 86% full power, the "C" Reactor Coolant Pump seal at ANO-Unit I failed resulting in excessive RCS leakage to the Containment sump. Operations personnel were taking leak rate data when a step decrease in makeup tank level occurred. This was the first indication that an RCS inventory loss had occurred. Reactor Coolant Pump "C" seal instrumentation confirmed that a problem existed with the seal or associated piping. Based on this information, it was decided to initiate a power reduction in preparation for shutdown as required by Technical Specification 3.1.6.1. The system dispatcher was notified of the problem and associated power reduction. At 0220, RCS letdown was secured to reduce RCS inventory loss. Due to the observed symptoms, Operations personnel actions were governed by the small break procedure (O.P. 1202.06, Sect. II). Extra Operations staff were called in at approximately 0225 because it was decided to go to Cold Shutdown operation. The decision to go to Cold Shutdown operation was made during

the power reduction. The resident NRC Inspector was notified at approximately 0227. NRC headquarters in Maryland was also notified of a controlled shutdown at approximately 0227. The Main Turbine Generator was off line at 0247 (approximately 62 minutes after power reduction was initiated). The power reduction initially started at a rate of approximately 5% per minute decrease with an estimated RCS leak rate of 10 to 20 gpm. During the power reduction, the RCS leak rate was observed to be increasing and the load reduction rate was increased to approximately 20 to 30% per minute. The "C" RCP was tripped one minute after the turbine was taken off line with the Reactor still critical. RCS leak rate took a step increase following the "C" RCP being shut off. The "C" RCP lift pumps were started and stopped four times in succession in an attempt to reduce the leak rate. After the fourth lift pump start, a decrease in RCS leak rate was observed. The Reactor was manually tripped from approximately 10% full power three minutes after the Main Generator was off line. In order to maintain pressurizer level and RCS pressure following the Reactor trip, "B" and "C" High Pressure Injection pumps were started and all High Pressure Injection valves (CV-1219, CV-1220, CV-1227, and CV-1228) were manually opened. The "C" Reactor Coolant Pump seal return was isolated at 0254 to prevent RCS inventory loss through the seal return line. The Reactor Coolant Pump seal flow was increased to quench the steam/water that was leaking by the failed seal. At this time, it was noted that the Reactor Building pressure had increased from 14.7 psia to 15.2 psia. The increase of building pressure and the associated increase in building radiation levels confirmed that the RCS leakage was inside the containment. At 0256 the Reactor Building Emergency Coolers were put in service to reduce the containment building pressure increase. "A" Reactor Coolant pump was secured at 0301. "C" High Pressure Injection pump was stopped and the High Pressure Injection valves were closed at 0305. This action terminated High Pressure Injection and normal RCS Makeup was established with "A" and "C" Makeup pumps taking suction from the Borated Water Storage Tank (BWST). Due to the relatively high Reactor Coolant System cooldown rate ($\sim$75°F/Hr), the operators did not reach the remote controls to bypass the SLBIC System actuation prior to reaching the 600 psi setpoint on the "B" Loop. When SLBIC actuated, the steam driven Emergency Feedwater pump (P-7A) started automatically. The "A" Loop did not reach the SLBIC actuation setpoint at this time. Steam header pressure was controlled by cracking open the Main Steam Isolation valve on the "B" Loop and then closing again. After raising header pressure above 600 psi the SLBIC function was bypassed; however, the header pressure was increased to approximately 650 psi which reset SLBIC and removed the bypass

Consequently, "A" Loop had SLBIC actuation when steam header pressure was decreased to the 600 psi setpoint. The steam header pressure was again increased and SLBIC reset. This time the SLBIC function was successfully bypassed and the header pressure was dropped below the 600 psi setpoint without SLBIC actuation. At approximately 0320, the Emergency Feedwater pump (P-7A) was stopped and the Auxiliary Feedwater pump (P-75) was lined up to feed the steam generators. CV-1220 (HPI Valve to RCS Loop A) was throttled as a makeup path in addition to the normal makeup through CV-1234 and CV-1235. This was performed as flow control was needed until about 1600 psi primary pressure.

As RCS pressure was decreased, a containment building entry was made to isolate the Core Flood Tank discharge valves. The entry was required to prevent the Core Flood Tanks from discharging to the RCS as the RCS pressure decreased below the Core Flood Tank discharge pressure of 600 psi. Following the Core Flood Tank isolation, the total decrease in Core Flood tank levels were 18 inches on "A" Core Flood Tank and 12 inches on "B" Core Flood Tank. The decrease occurred during the time Operations took to isolate the tanks. The RCS cooldown was essentially complete at 0900 on May 10, 1980 with the Decay Heat Removal System in service and all four Reactor Coolant pumps off. Throughout the incident, RCS conditions were maintained such that > 100°F margin to saturation was maintained. As a result of the RCP seal failure and subsequent cooldown, it is estimated that approximately 60,000 gallons was collected in the Reactor Building basement. Stack activities and Reactor Auxiliary Building areas remained at background levels. A normal Reactor Building purge was performed prior to maintenance activities. There were no personnel injuries or high radiation exposures.

10. Corrective Action:

Replaced/Rebuilt all four Reactor Coolant Pump Seal Cartridges.

All RCS leakage was reprocessed for use in the RCS, thus requiring no liquid releases as a result of the seal failure.

8. Designation of Apparent Cause of Occurrence:

Design	_____	Procedure	_____
Manufacture	_____	Unusual Service Condition Including Environmental	_____
Installation/ Construction	_____	Component Failure	X
Operator	_____		

Other (specify)

Reactor Coolant Pump 'C' Seal Failure.

9. Analysis of Occurrence:

A failure investigation was initiated, which included:

1) Examination of failed seal, the three remaining seals, and one spare seal. The findings of this examination were as follows:

   The RCP 'C' Seal 3rd stage damage was very severe. The stationary carbon ring was destroyed, the titanium carbide rotating face ring was broken, and the rotating face lock ring was distorted. There was additional related damage. The damage to the 3rd stage seal was too extensive to identify with certainty the initiating cause or failure made. The failure to the lower two seal stages most probably resulted from damage sustained during shaft excursions produced by the breakup of the upper (3rd) seal stage.

   The three remaining seals indicated that high temperature conditions had been experienced.

2) Seal failure history at ANO and other utilities of similar design.

3) Assess possible failure mechanisms:
   a) some type of transient, such as operating with the controlled bleed-off valve shut, can initiate a damaging wear pattern leading to a latent failure.
   b) Inaccurrate setting of the seal, specifically axial seal location, could produce abnormal wear, resulting in an increase in total closing force, sealing area, and total horsepower of the seal stage, all leading to failure.

4) Provide recommendations to prevent recurrence:
   Recommendations and guidelines are being considered at this time.