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U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Licensee Event Report 50-313/93-005-01

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i)(B), 50.73(a)(2)(ii)(B) and 10CFR50.73(a)(2)(v), enclosed is a supplemental report concerning the Reactor Building sump.

Very truly yours,

Jerry W Yelverton

JWY/mmg

enclosure

cc: Regional Administrator
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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

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)	PAGE (3)
Arkansas Nuclear One, Unit One	05000313	1 OF 8

TITLE (4) Breaches In The Reactor Building Sump Integrity Which Resulted From An Inadequate Design Review Created The Potential For Degraded Low Pressure Injection, High Pressure Injection And Reactor Building Spray System Flows During The Recirculation Mode of Operation Following a LOCA

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
10	14	93	93	-- 005 --	01	12	16	93	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	000

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

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		70.71(b)
	20.405(a)(1)(i)		50.36(c)(1)	X	50.73(a)(2)(v)		70.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		Specify in Abstract Below and in Text
	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 (Include Area Code)
R. H. Scheide, Nuclear Safety and Licensing Specialist	501-964-5000

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

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

On October 14, 1993, with the plant in cold shutdown, it was determined that the High Pressure Injection, Low Pressure Injection and Reactor Building Spray systems could have been rendered incapable of performing their design basis functions during the recirculation mode of operation following a Loss of Coolant Accident as a result of identified breaches in the integrity of the Reactor Building (RB) sump. The breaches included twenty two unscreened 6 inch long by 3 inch high split pipe openings at the base of the sump curb, four unsealed conduit penetrations in the sump screen, and two tears in the screen which could allow foreign material to enter the sump. The breaches in sump integrity occurred as a result of a modification made to the sump prior to initial power operation by the plant designer. The root cause of the condition was an inadequate review prior to modifying the sump design. Corrective actions were completed prior to startup following 1R11 to correct the deficiencies and restore sump integrity. The design change process in place at the present time is considered adequate to prevent the occurrence of similar events.

A. Plant Status

At the time this condition was identified, Arkansas Nuclear One, Unit 1 (ANO-1) was in Cold Shutdown with Reactor Coolant System (RCS) [AB] temperature at approximately 90 degrees and the RCS at atmospheric pressure. Refueling outage 1R11 was in progress.

B. Event Description

On October 14, 1993, at approximately 1315 CDT, it was determined that the High Pressure Injection (HPI) [BG], Low Pressure Injection (LPI)[BP] and Reactor Building Spray (RBS) [BE] systems could have been rendered incapable of performing their design basis functions during the recirculation mode of operation following a Loss of Coolant Accident (LOCA) as a result of identified breaches in the integrity of the Reactor Building (RB) sump.

The RB sump collects reactor coolant lost from the RCS as a result of a LOCA and provides a reservoir for the alternate suction of the Decay Heat Removal/Low Pressure Injection (DHR/LPI) pumps and the RBS pumps for long term core cooling and building spray when the Borated Water Storage Tank (BWST) level has been depleted (recirculation mode). The RB sump screen assembly is designed to minimize the potential for unacceptable debris introduction into the LPI, HPI and RBS equipment during the recirculation mode of operation for long term core cooling. This is accomplished by a grating and screen design that effectively filters material from injection water that could potentially degrade the long term post LOCA performance of the systems. The screen mesh size is 0.132 inches.

During operation, the LPI portion of the DHR/LPI system provides core cooling for large breaks and operates independent of, and in addition to, the HPI system. LPI is accomplished through redundant flow paths. Each path includes one pump and one cooler and enters the reactor vessel through the core flood nozzles, one on each side of the vessel. Crossover lines between the two LPI inlets allow injection of an adequate supply of borated water even if a core flood line ruptures. In the event of a small to intermediate reactor coolant leak, where the BWST has been depleted and RCS pressure remains too high to allow LPI (approximately 170 psig), the system can be operated in the "piggyback" mode. In this mode, the LPI pumps take suction from the RB sump and pump the water to the suction of the HPI pumps, which have sufficient discharge pressure to pump water into the RCS while it is at elevated pressures.

The RBS system consists of two pumps and associated spray headers which are designed to maintain post LOCA RB pressures within acceptable limits. A system secondary function is to remove iodine from the RB atmosphere. The RBS and LPI pumps share common suction lines from both the BWST and the RB sump. The RBS pumps can also be lined up through their test/recirculation lines to inject water into the RCS via the DHR system.

On October 1, 1993, ANO maintenance personnel noted that there were gaps in the RB sump screen where a standpipe for a sump level detector passed through the screen assembly. As a result of this deficiency, detailed sump inspections were conducted which identified additional breaches in the RB sump integrity. These breaches included:

- 22-6 inch diameter by 3 inch high split pipe scuppers through the sump curb at the RB floor elevation.

- 4 conduit penetrations through the screen/grating assembly which were not adequately screened.

- Insufficient seal around a 1 inch conduit which passed through the grout below the screen/grate assembly.

- Two tears in the screen behind the sump grating. One tear was approximately 12 inches long in the horizontal portion of the screen and the other was a 12 inch by 14 inch angled tear in the sump door.

- Floor drains leading to the RB sump were not screened.

With the plant in cold shutdown conditions, the RB sump and its associated systems were not required to be operable by Technical Specifications; therefore, immediate operability was not an issue. However, an engineering evaluation was initiated to determine if the identified deficiencies could have compromised the ability of the LPI, HPI and RBS systems to perform their post LOCA recirculation function. On October 14, a review of the preliminary findings of the on-going evaluation indicated that the LPI system could have been rendered incapable of performing its recirculation mode function if a LOCA had occurred during previous power operation. At approximately 1513 CDT, the NRC Operations Center was notified of this condition in accordance with 10CFR50.72.

C. Root Cause

The original sump screen was designed to be installed within the RB sump over the outlet pipes to the LPI and RBS systems. With this design, any water and/or debris which entered the sump through the scuppers or RB floor drain headers would have been outside the screen and therefore filtered if the screen integrity was intact. However, in May of 1973, prior to commercial operation, the plant designer (Bechtel) modified the sump design to increase the total area of screen surface. This was accomplished by installing the current A-frame type screen/grate assembly on top of the curb surrounding the sump. The new design would allow water passing through the scuppers and floor drain headers to bypass the screen and enter the sump unfiltered. A review of the documentation revealed that the concrete detail drawing for the sump shows the scuppers, but the sump screen assembly drawing does not. In addition, the screen assembly drawing did not provide details for sealing around conduit penetrations and it appears that field construction personnel left the unscreened openings around the penetrations during original construction of the A-frame assembly. Drains were not screened during initial construction; however, the headers entered the sump outside the screen and did not create an unscreened debris path. Although the licensing of ANO-1 preceded the issue of many present day standards and guides regarding RB sump performance, the Safety Analysis Report indicates that the designer had knowledge of sump integrity requirements since it states that "Flow into the recirculation piping from the reactor building sump is totally screened." Therefore, the root cause of this condition was determined to be an inadequate review by the plant designer prior to modifying the RB sump screen assembly design.

Damage to the screen in the form of tears is believed to have occurred during prior outages due to maintenance or modification activities in the area of the sump. The specific activities causing the damage or the time duration of the deficiencies could not be determined.

There have been several NRC communications issued to the industry addressing sump screen blockage and debris intrusion into pump suctions. However, most of this correspondence addressed types of debris and its affect on sump suction blockage with the exception of Information Notice (IN) 89-77, which addressed inadequate screen configurations in addition to debris and screen blockage. ANO's review of NRC correspondence, including IN 89-77, focused primarily on cleanliness and removal of debris present in containment and did not consider sump screen integrity. Consequently, the review resulted in procedure changes to perform building walkdowns and sump inspections at the end of outages to ensure cleanliness, but did not provide guidance addressing sump integrity. The failure to identify the sump integrity deficiencies has been attributed to the narrow review focus regarding IN 89-77 which evaluated debris but failed to consider RB sump integrity. In addition, two contributing factors were identified regarding the failure to identify the torn screen. These factors are:

* The low light levels in the area of the sump would make it more difficult to observe tears in the screen, which is located behind the grating of the screen/grate assembly.

* Plant and contract personnel who had the greatest opportunity to observe the tears in the screen (decon and maintenance workers) were unaware of the design requirements for sump integrity. In addition, the scuppers are obvious design features of the sump and considering their location, it would not be obvious to an observer that they were not screened internally.

D. Corrective Actions

The following actions were implemented prior to plant startup to correct the identified deficiencies.

- The scuppers were covered with a screen assembly fabricated from steel plate and 0.132 inch mesh screen.

- The tears in the screen and openings around the conduit penetrations were repaired using steel plate and 0.132 inch mesh screen.

- The floor drains in the RB basement floor were flushed and then grated and screened with 0.132 inch mesh screen. Drains from the upper levels were modified to direct drainage to the RB basement floor.

- The 1-inch conduit passing through the sump base was sealed with grout.

Additional corrective actions included:

- Inspection criteria to be used in plant procedures for closeout inspections of the RB sumps of both units was better defined in order to ensure sump integrity.

- A surveillance of the process for reviewing INs was performed by ANO Quality Assurance to identify any weaknesses. The results of the surveillance were satisfactory and indicated that the current IN review program is effective.

The relocation of Design Engineering on-site in 1990 allows for increased Design Engineering involvement during the construction, testing and closeout of design change packages. Additionally, the design change procedures in place at the present time require detailed documented reviews of design basis documents for each design change including revisions to design changes and are considered adequate to prevent the occurrence of similar conditions. Therefore, no changes to the design change process are considered necessary to address the root cause of this condition.

E. Safety Significance

The identified breaches in RB sump integrity introduced the potential for degradation in the performance of the LPI, HPI and RBS systems in their recirculation mode of operation after the BWST has been depleted following a LOCA.

A detailed engineering evaluation was performed to determine the impact of debris in the recirculated water on the performance of required safety systems. This evaluation included assessments of:

- The potential for debris and insulation migration to the sump

- The effect of debris and insulation on equipment operation

- Potential for debris and insulation to directly impact the core's ability to maintain a coolable geometry

- The potential for consequential effects such as increased post accident RB temperature and pressure.

Several mitigating factors were identified which limit the potential to impact LPI, HPI and RBS performance. These factors indicate that the breaches in sump integrity would not have been likely to pose a significant increase in the risk to the public health and safety. The mitigating factors include:

- The most probable cause of flow degradation during the recirculation mode of operation would be insulation fiber build up in the Decay Heat cooler outlet valves. Insulation fiber type debris would be generated as a result of a catastrophic failure of RCS piping which is either contained in fibrous insulation or is in close proximity to fibrous insulation covered piping. It has been calculated that approximately 13% of the "A" Steam Generator (SG) side piping and 29% of the "B" SG side piping is contained in fibrous insulation which was installed during refueling outage 1R10 (2/92 - 5/92). Prior to 1R10, there was no fibrous insulation installed on the RCS piping. The remaining RCS piping is contained in metal reflective insulation which will not break up into particles which could potentially degrade mechanical components such as the Decay Heat cooler outlet valves.

- Small Break LOCAs (which are the most likely breaks) should be terminated successfully in the injection mode of LPI and RBS operation by cooldown to cold shutdown on the Decay Heat Removal system (without entering the reactor building sump recirculation mode). This fact can be confirmed by both analytical evaluation and a review of the few LOCA events that have actually occurred in the industry.

- Medium Break LOCAs and Large Break LOCAs (which are the least likely breaks), although likely to proceed to the sump recirculation mode of LPI and RBS operation, result in a depressurized RCS. With a depressurized RCS, the alternate success paths for core cooling are maximized.

- The likelihood of the occurrence of Medium Break LOCAs and Large Break LOCAs is minimized by use of reactor building leak detection systems that will identify incipient failures, allowing the plant to be placed in a cold shutdown condition prior to an RCS pressure boundary rupture.

- For events that do progress to the reactor building sump recirculation mode, decay heat loads are such that significantly reduced LPI flow (compared to that required for the injection mode) will still provide adequate core cooling. Therefore, significant LPI flow degradation margin exists before inadequate core cooling occurs.

- The unscreened flow paths that have been identified for the reactor building sump represent only a small fraction of the total screened area through which recirculating cooling water passes (approximately 3.5% of the screened area) during the recirculation mode of cooling.

- Multiple core cooling flow paths and alternative alignments exist for injection, providing redundancy for components potentially impacted by debris introduction during reactor building sump recirculation mode operation.

- The most likely break scenarios (Small Break LOCAs, Medium Break LOCAs, transient induced LOCAs) are the least likely to generate debris that could be injected into LPI, HPI and RBS equipment through unscreened flow paths to the sump. For example, relief valve flow resulting from transient induced LOCAs would likely be enclosed in piping to the quench tank.

- The reactor building will still serve as an effective fission product barrier even if the RBS system and LPI coolers were to be significantly degraded, since the RB fan coolers are not dependent on the performance of the RB sump. The ANO-1 PRA RB response analysis has shown that with the availability of the RB fan coolers, RB performance (i.e., RB failure probability) is not challenged significantly.

The potential accident sequences were also assessed utilizing an event tree format in order to provide a quantitative risk perspective of the multiple success paths that are available to successfully establish post LOCA long term cooling. This evaluation calculated the potential increased core damage risk and associated RB performance for Small Break, Medium Break, Large Break and transient induced LOCAs with Engineered Safeguards equipment potentially degraded by recirculation water entrained debris. The results indicated that the increase in core damage frequency could have been as high as 5.19E-05/rx-yr or as low as 4.82E-06/rx-yr with a nominal IPE assessed core damage frequency of 5E-05/rx-yr. Recognizing that the NRC has established a nuclear plant safety goal of 1E-04/rx-yr in its severe accident policy statement, these results indicate that the identified condition represented a sizable contribution to the potential risk of core damage. However, the risk increase was not above the NRC safety goal.

Based on the above, it has been concluded that although this condition introduced an undesirable increase in the risk of core damage, it did not represent a significant or undue increase in the risk to the public health and safety.

F. Basis For Reportability

The breaches in the integrity of the RB sump created the potential for injected debris to compromise the ability of the LPI, HPI and RBS systems to perform their design functions in the recirculation mode of operation following a LOCA. Therefore, this condition is considered reportable pursuant to 10CFR50.73(a)(2)(ii)(B) as a condition outside the design basis of the plant and 10CFR50.73(a)(2)(v) as a condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident. In addition, this condition is reportable in accordance with 10CFR50.73(a)(2)(i)(B) as operation prohibited by Technical Specifications since the LPI and RBS systems were potentially incapable of performing their design basis function while required to be operable.

G. Additional Information

A similar condition was identified regarding the ANO-2 containment sump which was reported in LER 50-368/93-002-00. No other similar conditions have been reported as LERs by ANO.

Energy Industry Identification System (EIIS) codes are identified in the text by [xx].