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REGION II
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Richmond, Virginia 23261

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Facility Name: Surry 1 and 2

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Team Leader: M B Shymlock September 28, 1988
M. Shymlock, Chief Date Signed
Operational Programs Section

Team Members: T. Collins, Radiation Specialist, Region II
M. DeGraff, Reactor Engineer, Region II
W. Holland, Senior Resident Inspector, Surry
W. LeFave, Senior Reactor Engineer, Plant Systems Branch, NRR
L. Lawyer, Reactor Engineer, Region II
J. Mathis, Resident Inspector, Grand Gulf

Approved by: C. A. Julian 9/30/88
C. A. Julian, Chief Date Signed
Operations Branch
Division of Reactor Safety

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REPORT DETAILS

I. INTRODUCTION - FORMATION AND INITIATION OF AUGMENTED INSPECTION TEAM (AIT)

A. Background

Surry Units 1 and 2 are Westinghouse (W) pressurized water reactors (PWR) with Stone & Webster designed sub-atmospheric containments. The units are located five miles south of Williamsburg, Virginia, on the James River in Surry County, Virginia. Unit 1 went critical in July, 1972 and was declared commercial in December, 1972.

On Tuesday, August 30, 1988, the resident inspectors became aware of a report by the Independent Offsite Evaluation Review (IOER) group relating to an event involving borated water leakage through the Unit 1 refueling cavity floor seal. This event occurred on May 17, 1988, during the Unit 1 refueling and maintenance outage. This information was provided to regional management after preliminary assessment by the residents.

B. Formation of AIT

On the morning of Wednesday, August 31, 1988, the acting Regional Administrator, after further briefing by the regional and resident staff and consultation with senior NRC management, directed the dispatch of an AIT headed by the Section Chief of the Region II Operational Programs Section. The team included participation by the Office of Nuclear Reactor Regulation.

C. AIT Charter - Initiation of Inspection

The Charter for the AIT was prepared on August 31, 1988, and the AIT members arrived at the Surry site on September 1, 1988. Security badging was completed for the team, and the special inspection commenced with an entrance meeting and briefing by licensee management at 1100 hours on September 1, 1988. The Charter for the AIT specified the following:

1. Develop and validate the sequence of events associated with approximately 15,000 gallons of borated water leakage from the Unit 1 refueling cavity through the refueling cavity floor seal which occurred during the approximate time frame of May 17, 1988, while Surry Unit 1 was in a refueling outage. Our specific concerns which require evaluation include: (1) the potential degradation of safety-related instrumentation and equipment resultant from exposure to corrosive borated water, (2) adequacy of operator response during the

incident, (3) adequacy of the positive "J" seal design to prevent leakage of this type on Surry Unit 1 or Unit 2 and potential generic implications, (4) extent and significance of personnel radiation exposures during event, (5) adequacy of low head safety injection to replace the leakage, (6) extent of failure and safety significance of the failure of the instrument air, backup nitrogen supply, and related seals and equipment sufficient to support conclusions regarding the safety of continued plant operations, (7) adequacy of management evaluation of the event both with respect to scope and timeliness, and (8) licensee reporting of the event. Key items the AIT should emphasize include all equipment malfunctions, major plant evolutions/status changes, operator errors, licensee management/support organization response, and reports made to the NRC.

2. Evaluate the significance of the event with regard to radiological consequences, safety system performance, and plant proximity to safety limits as defined in the Technical Specifications.

3. Evaluate the accuracy, timeliness, and effectiveness with which information on this event was reported to the NRC.

4. For each seal or related equipment malfunction, to the extent practical, determine:

   a. Root cause.

   b. If the equipment was known to be deficient prior to the event.

   c. If equipment history would indicate that the equipment had been historically unreliable or if maintenance or modifications had been recently performed.

   d. Any equipment vendor involvement prior to or after the event.

   e. Pre-event status of surveillance, testing, (e.g., Section XI), and/or preventative maintenance.

   f. The extent to which the equipment was covered by existing corrective action programs and the implication of the failure with respect to program effectiveness.

5. Evaluate the licensee's actions taken to verify equipment operability.

6. Identify any human factors/procedural deficiencies related to this event.

7. Through operator and technician interviews, determine if any of the following played a significant role in the event; plant material condition; the quality of maintenance; or the responsiveness of engineering to identified problems. Unless these concerns involve immediate safety issues, team actions should be limited to communicating the concerns to NRC management.

D. Persons Contacted

Those persons contacted by the AIT are identified in Appendix 1.

E. Description of principal Operations Shift Staffing at the Time of the Event

Abbreviations for the principal Operations Staff are used for convenience throughout the report. The following brief explanation of each position is provided:

SS Shift Supervisor - A Senior Reactor Operator (SRO) responsible for both Unit 1 and Unit 2 operations. While on shift he/she is also the unit supervisor for one of the two units.

USS Unit Shift Supervisor - The junior of the two SROs on the shift and responsible for the other unit.

STA Shift Technical Advisor - He/she is assigned to the shift to advise the SS on matters pertaining to the engineering aspects of assuring safe operations of the plant.

CRO Control Room Operator - A licensed reactor operator responsible for the operation of his/her assigned unit.

RS Refueling Supervisor - An SRO responsible for all fuel movement activities.

CS Containment Supervisor - An SRO responsible for overall operations activities in containment other than fuel movement.

UTS Unit Test Supervisor - An SRO responsible for Type C testing of containment penetrations and repair of their associated components. Several non-licensed operators work under the direction of this individual in fulfilling his responsibilities.

NLO Non-Licensed Operator - A non-licensed operations department individual trained in the location, operation, and safety significance of plant equipment in his work area. Reports to one of the CROs or supervisors described above.

These and other acronyms and abbreviations used in this report are identified in Appendix 2.

F. Design Description

Design descriptions for the major equipment and systems discussed in the report are provided in Appendix 3.

II. Description of Event

A. Overview of Event for Surry Unit 1

1. Initial Conditions

On the morning of May 17, 1988, Surry, Unit 1 was in the middle of a refueling and maintenance outage. The reactor vessel was defueled (all fuel had been transferred to the spent fuel pool). No fuel movement was in progress. The reactor cavity was flooded to approximately 27 feet and the spent fuel pool was isolated. Contract personnel, W were performing work from the refueling bridge on the upper internals package thermocouple conduits. An operations department group was performing Local Leak Rate Tests (LLRT) and maintenance on various containment penetrations.

Unit 2 had experienced an automatic reactor trip from 100 percent power with a manual safety injection early on the morning of May 16, 1988. Following the trip, the unit experienced problems associated with auxiliary feedwater (AFW) flow to the "A" steam generator. Evaluation and troubleshooting of the AFW flow problems were still receiving management and operations staff attention on May 17, 1988.

2. Event Description

On May 17, 1988, while preparing to repair instrument air (IA) valve, 1-IA-849, an NLO requested between 0800 and 0830 hours, via the control room, that IA to Unit 1 containment be isolated (this was necessary to facilitate repair of the valve). Upon entering the "C" loop room (located in containment) he observed water cascading down the walls through the reactor loop piping penetrations. His first response was to inform the control room and have IA re-established to the containment and his second was to determine if the nitrogen bottles were supplying pressure to the refueling cavity floor seal. He noted that one of the nitrogen bottles was empty and the pressure regulator on the other bottle was misadjusted. This prevented it from being able to pressurize the seal. He then adjusted the regulator to supply pressure to the seal and noted that the leak decreased.

This series of events resulted in nearly 30,000 gallons (which equates to approximately three feet of water in the refueling cavity) of water being drained through the deflated refueling cavity floor seal. The reduction in cavity level resulted in increased radiation level on the Unit 1 operating deck. Work on the upper internals package had been suspended by the health physics (HP) technician due to increased radiation levels.

3. Licensee Actions Following the Event

The NLO, noting the water leak in the "C" loop room informed the control room. The CRO and SS were informed of the problem. The CRO noted that the incore sump high level alarm had alarmed. Attempts to start the incore instrument room sump pump failed. An NLO was sent to check out the incore instrument room sump pump breaker at the motor control center. It was noted that the breaker had tripped on thermal overload and it would not reset. This may have indicated that the motor was submerged.

Sometime between 0830 hours and 0900 hours the Operations Coordinator entered the control room. The SS informed him that there had been "a foot or so drop" in cavity water level, and that HP had suspended work on the operating deck due to increased radiation levels. The Operations Coordinator immediately went to inform the Assistant Station Manager.

Sometime later on May 17, 1988, the Superintendent of Operations and the Operations Coordinator inspected the "C" loop room. No problems were identified. The Station Manager was also informed of the event that day.

The team was unable to determine whether any additional evaluations or other actions pertinent to the event were taken by the licensee until two days later when an STA prepared deviation report S1-88-422.

B. Detailed Sequence of Events

SURRY UNIT 1 - REACTOR CAVITY SEAL FAILURE

MAY 15, 1988

Time (EST Hours)	Data Source	Item
0554	CRO Log	Periodic Test, PT-10 Reactor Coolant Leakage walkdown completed satisfactory. Containment sump in-leakage calculated to be

Time (EST Hours)	Data Source	Item
		8.2 gpm. Nitrogen bottle pressures (to the refueling cavity floor seal) found to be 1800 psig and 2200 psig.
MAY 16, 1988		
0200	SS Log	Verified Instrument Air supply to Unit 1 Containment instrument air header being supplied through valves 1-IA-446, and 447.
0223	CRO Log	PT-10 Reactor Coolant Leakage walkdown completed satisfactory. Containment sump in-leakage calculated to be 9.9 gpm.
1433	SS Log CRO Log	Instrument air to Unit 1 containment isolated to investigate a problem associated with instrument air valve, 1-IA-849. The nitrogen bottles were verified as being aligned to the refueling cavity floor seal.
1504	SS Log CRO Log	Instrument air re-established to Unit 1 containment.
1610	Type C Test Log	Instrument air was valved out to Unit 1 containment (verified that the refueling cavity floor seal was being supplied by the nitrogen bottles). Attempted to seat valve 1-IA-849, the valve still leaks. Instrument air returned to service.
	* NLO Interview	The NLO performing the valve manipulations indicated that upon exiting containment he requested that his relief perform "independent verification" associated with instrument air and nitrogen back-up supply valve line-ups.

Time (EST Hours)	Data Source	Item
MAY 17, 1988		
	* NLO Interview	This NLO indicated that the verification requested from the previous shift was completed between 0130 hours and 0200 hours. The verification consisted of insuring nitrogen and instrument air supply valve lineups were correct as well as verifying that, if required, the nitrogen bottles would in fact supply the seal.
0223	SS Log	PT-10 Reactor Coolant Leakage walkdown completed satisfactory. Containment sump in-leakage calculated to be 13.4 gpm. The nitrogen bottle pressures to the refueling cavity floor seal were verified to be 1500 psig and 1800 psig.
0230	Chart recorder	The chart recorder in the control room which plots input from RMS-162, manipulator crane radiation monitor, indicated the measured radiation levels to be approximately 5mr/hr.
0830	CRO Log	Isolated instrument air to containment for PT-16.4, Containment Isolation Valve Leakage.
	* CRO Interview	The operator recalled that during the event, the incore room sump high level alarm did annunciate. The setpoint for this alarm is 18 inches.
	* USS Interview	He recalled the incore instrument room sump high level alarm being illuminated. Additionally, he noted attempts to start the incore room sump pump failed.

Time (EST Hours)	Data Source	Item
0852	CRO Log	Valved instrument air back into containment after noting containment sump level increasing and discovered that one back-up nitrogen bottle to the refueling cavity floor seal was empty and the other was at 1500 psig but did not appear to be supplying the refueling cavity floor seal.
	* Chart Recorder	The chart recorder in the control room which plots input from RMS-162, manipulator crane radiation monitor, indicated the measured radiation levels increasing to approximately 35mr/hr. This increase occurred between 0830 and 0900.
0855	SS Log	Received a report of refueling cavity floor seal inflatable seal leakage. Instrument air to containment had been isolated to support operations work.
0857	SS Log	Instrument air restored to containment.
0911	HP Supervisor Log	Water level in the reactor cavity has dropped. General area around the cavity was up to 100mr/hr. Work on RWP No. 88-RWP-1507 was stopped until water level is raised by operations.
0921	SS Log	NLO reports that one nitrogen bottle to the refueling cavity floor seal ring was completely depressurized. The second bottle gage indicated 1500 psig, however the regulator was misadjusted. The seal was repressurized and preparations are in progress to restore level in the refueling cavity using primary grade (PG) water.

Time (EST Hours)	Data Source	Item
0958	SS Log CRO Log	Received a report from HP that water is leaking around concrete access plug to the incore instrument sump room located in the containment basement. Leak rate is estimated at four gpm. Containment sump pump is keeping up with the leakage. Incore sump pump motor control breaker thermal overloads tripped. Possible cause, pump motor submerged.
1005	SS Log	The fuel transfer tube isolation valve open approximately five turns to raise refueling cavity level.
	Chart Recorder	Chart recorder for RMS-162, manipulator crane radiation monitor, recorded radiation levels begin a decreasing trend. Starting at 1000 hours at approximately 35mr/hr and ending at 1100 hours with approximately 10mr/hr. (It should be noted that this decreasing trend coincides with the refill of the refueling cavity from the spent fuel pool)
1105	SS Log	Secured filling the refueling cavity through the fuel transfer tube. Spent fuel pool was lowered approximately eight inches.
1105	HP Supervisor Log	Water level in the cavity has returned to normal.

* Indicates event is entered at the approximate time frame.

III. SUBSEQUENT LICENSEE ACTIONS

A. Refill of the Reactor Cavity From the Spent Fuel Pool

Following the event the licensee, in order to recover level in the refueling cavity, opened the fuel transfer tube isolation valve five turns. This occurred approximately two hours after the event and the valve remained opened for approximately one hour. This resulted in an eight-inch drop in spent fuel pool water level. The following day, May 18, the valve was opened again. This time, the level in the spent fuel pool was reduced five inches. The AIT was provided information which indicated that a one-inch decrease in spent fuel pool water level was equal to approximately 1440 gallons. Using the above information, the inspectors determined that approximately 18,700 gallons of water was transferred from the spent fuel pool to the refueling cavity to partially make up for the water lost during the seal leak.

The AIT questioned the licensee's level recovery method and requested a copy of the procedure which was used. The licensee indicated that there was no operating procedure which addressed this evolution. During the time period between the two cavity fills, the spent fuel pool was refilled with PG water. After, refilling, the spent fuel pool was sampled to verify boron concentration to be greater than 2000 ppm. The AIT expressed concern that the above method of refilling the cavity was performed without assurance that the seal could meet its intended design function and without procedures.

The AIT also noted that additional water was added to the refueling cavity from the refueling water storage tank just prior to refueling the vessel in order to establish proper refueling level.

B. Deviation and Human Performance Evaluation System Reports

Two days after the event, May 19, 1988, an STA prepared Deviation Report (DR) S1-88-422. Within the report possible causes leading to the event were identified. They were human error, procedure/drawing error and/or design. The Corrective Action section of the report dated June 13, 1988, indicated "no corrective action, management informed - Surry Human Performance Evaluation System (HPES) report 88-012, unresolved for human error. Implementation of design change similar to Unit 2, Engineering Work Request (EWR) 85-200." This DR was subsequently reviewed by the Site Nuclear Safety Operation Committee (SNSOC) on July 7, 1988.

HPES report 88-012 was forwarded to the SNSOC chairman on June 13, 1988. As in the case of the DR, the HPES report concerned itself primarily with human error surrounding the event. The conclusion reached in this report was that during the time frame of 0100 hours to 0830 hours on May 17, 1988, "an unauthorized, non-recorded valve isolation occurred on the normal air supply to the cavity seal ring resulting in partial loss of cavity level." In addition, the statement was made that "the potential hazards that can be created due to this activity cannot be understated." Finally, the report stated that "unfortunately no specific corrective actions can be generated from this office".

The cover letter indicated that "HPES evaluation could not determine specific causal human factors that would have contributed to the event and that the report was being submitted to SNSOC for information and to assure that appropriate management personnel were aware of the results."

The AIT reviewed the DR and HPES reports. The DR disposition was considered superficial in that it failed to recognize other important factors such as:

- possible design deficiencies in the seal;
- the failure of the "J" seal and operability concerns regarding refueling activities;
- the failure of the back-up nitrogen system to supply the seal;
- inadequate procedures to operate the air and nitrogen systems;
- inadequate drawings to indicate system configuration; and
- the generic implications associated with the same seal arrangement used on Unit 2.

The HPES report was considered to be inconclusive in that the only "Human Performance" indicator identified was a "non-authorized, non-recorded valve manipulation." The AIT considers other "Human Performance" indicators pertinent to the event which were not discussed to be:

° no identification tags on IA or nitrogen system valves;

° no drawings depicting IA or nitrogen system configuration.

The AIT also noted that repairs to valve 1-IA-849 were conducted without procedural guidance.

On August 16, 1988 following the Independent Offsite Evaluation Review (IOER) group investigation of the event, a second DR, S1-88-0873, was written. The deviation description identified that the "J" seal portion of the refueling cavity floor seal would not preclude leakage from the refueling cavity as stated in the Updated Final Analysis Report (UFSAR) 9.12-3 and in Surry's response to IEB 84-03, dated October 9, 1984. The DR indicated that "the "J" seal design is inadequate and would allow leakage greater than the available make-up source from one low head safety injection (LHSI) pump if the inflatable seal failed." This second DR appeared to be more complete in its identification and analysis of the deviation.

C. Incore Instrument Room Cleanup

The failure of the refueling cavity floor seal resulted in borated water being deposited in various locations in containment but primarily in the incore instrument room located below the reactor vessel. The water that accumulated was calculated to have achieved a level of approximately five to six feet above the floor.

The borated water remained in the room for approximately 30 days. The incore detector guide thimbles were retracted from the vessel and radiological conditions precluded entry. Once refueling was completed with the guide thimbles reinserted, radiation levels were low enough to allow access.

The room was pumped out and the area washed down with purified water. There were no special tests performed or wall smears taken to determine actual water levels. Following the clean-up effort, the room was inspected and the results determined to be satisfactory. The room was sealed and preparations were made to return to power.

D. Lack of Engineering Review and Subsequent Fuel Reload

Following the event, even though DR SI-88-422 indicated a probable cause of the deficiency to be design related, no engineering evaluation of the seal design or failure modes were performed. This oversight was apparently caused by the station's belief that the leak was small and over a long period of time, and that the event stemmed from human error. As a result, the licensee did not question the design or its ability to perform its intended function. Therefore, no corrective actions or compensatory measures were implemented by the licensee prior to refueling the reactor vessel.

The AIT and the licensee determined that the ability to make up for a refueling cavity floor seal leak exceeded the capacity of one LHSI pump. It is uncertain how much of the "J" seal was actually displaced, thus the anticipated leak rate could be much higher.

The most significant concern is that refueling operations were conducted (all the fuel was reloaded) without confirming that the seal would perform its intended design function.

E. IOER Evaluation of the Event

The IOER group received HPES report 88-012 on July 14, 1988. Their subsequent review deemed the report to be inconclusive and due to the concerns raised, an investigation into the event was initiated.

On August 17, 1988, the IOER group forwarded to plant management their findings regarding their investigation of the refueling cavity floor seal failure at Surry Unit 1. This investigation report included a summary of events surrounding the failure of the seal, the implications of the event, and finally, identification of concerns and proposed actions in the areas of administrative controls, technical issues, and operations response.

The report identified the most important question as being "why the passive "J" seal did not prevent catastrophic leakage and would the resulting leakage exceed the capacity of the make-up capability provided?" Additionally, it was stated that "this concern focuses on the potential for uncovering a suspended fuel assembly and the time necessary to relocate an assembly into a safe position." It should be noted that both the licensee's UFSAR and response to IEB 84-03 state that the passive "J" seal will prevent this from occurring.

In line with their investigation, the IOER group initiated a detailed design review of the "J" seal. They concluded that "the lack of a positive backing plate on the seal can allow the upward displacement of the bulbous portion of the seal due to forces exerted underneath it. These forces result from the flow of water past the seal due to surface imperfections or seal deterioration." Additionally, it was stated that "the calculated buoyancy of the seal in borated water with an air hole in the center is very near the buoyancy point."

The report went on to further discuss the refueling cavity floor seal installation procedure, MMP-C-RC-37. MMP-C-RC-37 requires the stand-off supports be set at 1 3/16 inches. Calculations performed by the licensee show that for a 30% compression (which results from the stand-offs being set at 1 3/16 inches) there will be about 3/8-inch of surface contact between the "J" seal and mating surface. Due to the lack of a backing plate, there is nothing to guard against a reduction in this surface contact. In addition, it was stated that "if the seal were not regularly replaced, resiliency is lost and the ability of the seal to accommodate surface imperfections is lost."

As previously stated, the final portion of the investigation report identified several concerns and proposed actions in the areas of administrative controls, technical issues and operations response.

In the area of administrative controls the following concerns were identified:

- "A station deviation report was not immediately submitted to initiate a review of the event and to ascertain reportability."

- "HPES report 88-012 was submitted without fully assessing the event or the ramifications of the event."

- "The UFSAR states that the "J" seal will prevent leakage in the event of a failure of the inflatable seal. This design basis was reiterated in response to IEB Significant Operating Experience Report, 84-03, 85-01 and internally to IEIN 84-93."

To resolve these concerns the proposed actions included:

- "Submit a station deviation report to document evaluations and determine reportability."
- "Station personnel should be reminded of the requirements for submitting deviation reports."
- "HPES evaluators should be reinstructed on the necessity of performing detailed evaluations, submitting deviation reports for operational events beyond human factors and the need to have a thorough review of reports prior to issuance."

In the area of technical issues the following concerns were noted:

- "The potential flowrate past the "J" seal in its current design, may exceed the capacity of a single LHSI pump."
- "Based on leakage experienced and discussion with the vendor it appears the design of the seal ring is not an acceptable application."
- "Evaluate the residue of boric acid that accumulated on the reactor vessel walls which was not removed."
- "The vendor recommends the inflatable seal should have increased strength provided by fiber reinforcement. In addition, the sealing surface contact surface area should be increased."

To resolve these concerns the proposed actions included:

- "Determine the maximum flowrate that can occur and compare this to the make-up capability of a LHSI pump."
- "Investigate possible seal ring design improvements that can be implemented."
- "Evaluate the impact of boric acid residue on the carbon steel reactor vessel exterior walls."
- "Review the design of the nitrogen back-up system and implement improvements as required."
- "Review vendor recommendations for need."

In the area of operations the following concerns were noted:

- "How did IA to the inflatable seal become isolated. It was indicated that the local IA supply valve was found closed, however logs indicate the problem occurred when IA was valved out to support Type C LLRT on penetration 47."

- "Operators did not enter the appropriate procedure for a loss of refueling cavity level."

- "The fuel transfer gate valve was opened to restore level in the refueling cavity. Given the potential for further leakage and a potential failure of the gate valve to close, spent fuel pool level could have been significantly reduced. This action was not based on procedural guidance, is an unadvisable method and contradicts the requirements of Technical Specification (TS) 5.4.D."

- "AP-22 "Fuel Handling Abnormal Conditions" and AP-27 "Loss of Decay Heat Removal Capability" provide inadequate guidance to operations personnel on a rapid loss of refueling cavity water level."

To resolve these concerns the following action was identified:

- "A review of current procedure controls for a loss of refueling cavity level should be performed."

F. IOER Evaluation Presented to Station Management and NRC

After identification of the above concerns by the IOER group, the engineer who prepared the IOER report submitted a station DR in accordance with procedure. That DR, S1-88-0873, which is discussed in section III B of this report, identified a design problem associated with the "J" seal portion of the refueling cavity floor seal. After receiving the DR, the station safety committee requested and received a presentation on the IOER concerns which resulted in the DR. This presentation was made at the Surry Power Station on Thursday, August 25, 1988. On that day, one of the NRC residents walked into the meeting near the end of the presentation, but was not aware of the problem at that point.

The safety committee concluded that some of the information presented was incorrect and requested the IOER engineer to provide additional information to justify some of the IOER concerns. On Friday, August 26, 1988, the Assistant Station Manager for Licensing and Safety provided information on the IOER presentation to the NRC residents; however, a copy of the IOER report was not provided.

This Manager indicated that design deficiencies identified in the report were under review and would be addressed the following week. The resident inspectors became aware of the IOER report on August 30, 1988. The Station Manager provided the residents a copy of the report on August 31, 1988. The licensee made a 10 CFR 50.72 report of the rapid decrease in refueling cavity water level on September 1, 1988.

G. Justification for Continued Operation

On September 2, 1988, the licensee provided a Justification for Continued Operation (JCO) of Unit 1 as requested by the NRC. This JCO relied on and transmitted the licensee's engineering evaluation, Technical Report PE-0005 dated September 1, 1988, of the potential effects of borated water flooding of the incore instrument room as related to the then present and continued safe operation of the facility.

The licensee's engineering evaluation assumes that initial leakage past the "J" seal would have been collected by the drip pan. This leakage would have then been carried away via drain piping to the containment sump. However, as leak flow increased beyond the capacity of the drip pan, the flow path would have been primarily down the exterior of the reflective insulation, over the neutron shield tank and into the incore instrument room. In addition, the licensee indicated a small amount of leakage could have flowed onto the reactor vessel nozzle reflective insulation and flowed and/or splashed along the reactor coolant piping into the loop rooms.

The licensee's analysis states that all equipment in the loop rooms is qualified for chemical spray. Therefore, the subject leakage/flooding would in no way prevent any equipment in the loop rooms from performing their design functions.

Within the incore instrument room, 11 critical components were identified. Of these, three are constructed of austenitic stainless steel which is not adversely affected by wetting with borated water. These three are the reflective insulation, the reactor coolant piping, and the incore instrumentation guide tubes.

One component received spray but was probably not submerged. Its exposed surface is a 347 stainless steel sheath and other non-stainless steel components are hermetically sealed in this sheath.

Four of the components were coated with design bases accident qualified paint which is not adversely affected by boric acid. These were the supply and return lines for the neutron shield tank coolers, the containment mat liner plate, the neutron shield tank and the incore instrumentation guide tube supports.

One component, the Gamma Metrics Excore Neutron Detector is composed of a signal cable and junction box. The junction box is unprotected carbon steel with SS cabling attached. This junction box is sealed with a silicon O-ring. (It is not clear whether this junction box is above or below the six foot water level.) Even if the junction box was submerged, it should have only suffered a loss of some 0.001 inch of its 3/8 inch thickness. The cabling consists of a solid copper coaxial conductor insulated with Kapton tape encased in a flexible stainless steel hose and covered by woven glass fiber.

The remaining two components were briefly wetted, protected by geometry, and would have suffered less than 0.001 inch material loss. These two were the reactor vessel (primarily the flange) and the reactor vessel sliding supports. These latter supports were also protected by a lubricant.

The licensee concluded in its JCO that "As a result of these investigations (described above), the flooding of the incore instrument room with borated water will have no adverse effect on continued safe operation of the plant."

The AIT concluded, following an evaluation of the JCO that the licensee had adequately addressed the potential degradation of safety-related instrumentation and equipment from exposure of corrosive borated water.

IV. EQUIPMENT STATUS, FAILURES/MALFUNCTIONS, AND ANOMALIES

A. IEB 84-03 Licensee Response and Modification

The licensee's response to IEB 84-03 dated October 9, 1984, indicated an evaluation of the potential for and the consequences of a refueling cavity floor seal failure had been performed.

Their response contained a brief design description detailing the operation of both the inflatable seal, and the passive "J" seal. In addition, the licensee indicated that procedures require a pressure drop test on the inflatable seal as well as a visual inspection of the "J" seal prior to installation. Although not stated, it appears this information was provided to assure the NRC that even if seal degradation were occurring, it would be discovered prior to seal use.

They indicated that at least one makeup path was available at all times during refueling. Therefore should the pressurized seal fail, any of the available makeup paths could be used to maintain water level, while the passive "J-seal" would preclude leakage.

They further explained that although a catastrophic failure is not credible because of the design, should such a failure occur, the elevation of the spent fuel transfer system would prevent a fuel assembly from being uncovered. Additionally, a barrier in the spent fuel storage pool precludes the draining of the pool's water to less than 13 inches above the fuel racks.

The licensee concluded that a complete failure of the refueling cavity floor seal was not a credible event. In addition, based on their evaluation and seal design differences between the two facilities (Surry and Haddam Neck) they believed the seal assembly employed at Surry to be adequate. Finally, as a result of the IEB review, the licensee revised AP-22, Fuel Transfer Equipment Malfunction, to provide operator actions to be taken in the event of a rapid decrease in refueling cavity water level.

The procedure delineated immediate operator actions which consisted of the following:

- Providing makeup by several means,

- Placing the fuel assembly in the safest position possible. If a fuel assemble was in the manipulator the procedure required returning it to the core, and

- Instruction to close the fuel transfer tube gate valve, isolating the spent fuel pool from the refueling cavity.

Additional procedural actions provided were:

- Isolation of the leak or rupture,

- Monitoring residual heat removal (RHR) pumps for proper operation and signs of cavitation, and

- Rectification of the problem and resumption of normal activities as directed by the SS.

In April of 1987 as part of an intended procedure upgrade program, many of the corrective actions were deleted from Abnormal Procedure AP-22. The AIT determined that AP-22, which was available to operators on May 17, 1988, was inadequate to deal with a decrease in refueling cavity level.

It was also noted that there were no directions in the procedure for inspection of the IA and/or back-up nitrogen supply systems either prior to or after the procedure upgrade.

The AIT concluded that the licensee was not in compliance with the IEB 84-03 response. A catastrophic failure was probable, operating procedures were not adequate to address the event, and an LHSI pump (3250 gpm) will not be able to maintain cavity water level. The AIT also identified an inadequacy in the licensee's administrative control process that assures that commitments to the NRC are maintained.

System Modification

Subsequent to the issuance of IEB 84-03, the licensee performed a review of the facility's refueling cavity floor seal. After the review the licensee concluded that it would be desirable to incorporate a back-up air supply for the refueling cavity floor seal. This would provide redundancy and thus maintain the inflatable seal inflated in case of IA failure.

On April 4, 1985, EWR 85-200 was approved to support the design and installation of a nitrogen back-up supply system on Unit 2. During this inspection the AIT determined that a similar nitrogen back-up supply system was installed on Unit 1. It appears that the system was installed under a temporary modification during the 1984 Unit 1 refueling outage. However, the licensee could not produce any documentation which supported the finalized installation similar to EWR-85-200 used on Unit 2. Since the AIT, the licensee has provided information which indicates that the temporary modification was closed out following the outage with no followup action.

EWR 85-200 discussed several conclusions and recommendations. A review of the available documentation indicates several problems with the Unit 1 and Unit 2 nitrogen back-up supply systems. These problems are enumerated below:

- EWR 85-200 recommended that check valves to prevent air backflow and relief valves to prevent overpressurization be installed. Discussions with the licensee indicated that these components are installed on Unit 2 but not on Unit 1. In either case (Unit 1 or Unit 2) it is difficult to ascertain specific system configuration due to the lack of as-built drawings.

- EWR 85-200 recommended that procedure MMP-C-RC-037, Installation, Inflation and Removal of Reactor Cavity Inner Seal Ring, be revised to include steps for setting and testing the pressure regulators, and relief valve, and steps to install and remove the nitrogen bottles. A review of MMP-C-RC-37, used during the May 1988 Unit 1 refueling outage, indicates that none of these recommendations had been implemented. This procedure is applicable to Units 1 and 2.

- EWR 85-200 recommended that the nitrogen bottles have their pressure regulators set at 20 psig. In addition, it recommended that the existing IA pressure regulators be reset to 25 psig versus 20 psig. A review of MMP-C-RC-037 used during the May 1988 Unit 1 refueling outage indicated that the IA pressure regulator was still set at 20 psig, per step 5.5.2.

B. Significance of Seal Failure

During this inspection it was determined that approximately three feet of water was drained from the refueling cavity over a relatively short period of time. The AIT was provided information that the refueling cavity contained approximately 240,000 gallons of water when filled to a depth of 26 feet. The AIT calculated that a three foot drop in cavity level would result in a loss of approximately 27,800 gallons of water. The licensee stated in the AIT exit on September 3, 1988, that the majority of water was drained in approximately four minutes. Using four minutes as the time in which the water was drained and 27,800 gallons as the quantity of water drained, the AIT determined that the leakage through the refueling cavity seal was approximately 6,950 gallons per minute. The design flow rate of one LHSI pump as specified in the UFSAR is approximately 3250 gallons per minute at a design discharge pressure of 225 feet of water. Therefore, one LHSI pump would not keep up with the calculated leakage identified above. It was noted from operator logs during the event that the LHSI pumps were not available.

C. Maintenance Activities

1. Local Leak Rate Testing

The licensee, on May 16,1988, was conducting LLRT on containment penetration No. 47. The testing was being conducted in accordance with PT-16.4, "Containment Isolation Valve Leakage," dated April 8, 1988. Penetration 47 supplies IA to a two-inch IA pipe header inside containment. This header distributes air to various components including the inflatable refueling cavity floor seal. The purpose of the test being conducted was to measure back leakage through check valve 1-IA-939. See IA description in Appendix 3B and Figure 2.

Problems associated with seat leakage on 1-IA-849 prevented the test from being completed. 1-IA-849 is one of two valves required to be closed in order to isolate the penetration. To expedite repairs, a contractor was brought in but, due to time constraints and a lack of spare parts, repair efforts were scheduled to continue the following day, May 17, 1988.

On May 17, 1988, at about 0830 hours, repair activities commenced on 1-IA-849. Operations, to support this effort, had isolated IA to containment. The NLO assigned to this work entered the "C" loop room to verify the nitrogen bottles were supplying the seal (see Figure 2) at which time he discovered water cascading down the walls.

2. Maintenance History

Maintenance had been performed on both the "J" seals and the inflatable seal. In May of 1986, the "J" seals, associated fasteners and retainers were replaced under work request 333552. This work was accomplished due to natural end of life.

MMP-C-RC-37, Installation, Inflation, and Removal of the Reactor Cavity Inner Seal Ring, dated April 12, 1988, requires: (1) a pressure drop test on the inflatable seal; (2) a visual inspection of the seals; and (3) a visual inspection of the "J" seal seating surface. All of the aforementioned tasks are to be completed prior to seal assembly installation.

The pressure drop test requires that the inflatable seal be inflated to 20 psig. Following inflation, the air source is removed. The acceptance criteria specifies that 20 psig be maintained for 10 minutes.

The test performed during the 1988 Unit 1 refueling outage failed to meet this acceptance criteria. Once the seal was inflated, and the air source removed, the pressure dropped to 16 psig in the first 5 to 6 minutes. It then held at this pressure for the remaining portion of the test. The visual inspection noted evidence of surface nicking and scraping. Each of these deficiencies resulted in a Quality Control (QC) rejection.

The licensee performed an evaluation of the noted deficiencies under EWR 88-116. The EWR indicated the calculated leak rate for the seal to be six to eight scfh. This was determined to be well within the capacity of the nitrogen bottles which are designed for a 25 scfh leak rate over 24 hours. It also concluded that the deficiencies noted during the visual inspection were assumed to be caused by seal handling during decontamination efforts.

Finally, the EWR concluded that even in the event of a complete failure, deflation of the inflatable seal, the back-up passive "J" seal will prevent excessive leakage.

The visual inspection performed on the "J" seal seating surface also resulted in a QC rejection. This was based on evidence of suspected boric acid and oil residue, fixed rust and loose metallic flakes. In addition, a six-inch long scratch was also identified.

The licensee performed an evaluation of these deficiencies under EWR 88-148. While generating the EWR, it was expanded to include pitting, denting, hammer blows, and discoloration as well as a sharp edge apparently caused by metallic contact. The resolution of the identified deficiencies included the following:

- The defects did not encompass the entire width of the seating surface at any location around the circumference of the flange.
- The defects were found to be less than 1/32 of an inch indepth with no sharp edges or burrs.
- The hammer blows were observed to have characteristics similar to the defects.
- The hammer blows were also identified as being located outside the actual seating surface and having no affect on the sealing.

Given the known deficiencies discussed above, contribution to the event from these items cannot be overlooked. No inspection of the seal assembly was performed following the 1988 refueling activities. This was due to the licensee's belief that the event stemmed from personnel errors, and the leak being small over a long period of time.

The AIT concluded that the licensee's evaluation of the capability of the nitrogen bottles to maintain a six scfh seal leak rate was incomplete. The evaluation never addressed the nitrogen bottles as a "finite supply." The operators had no direction to monitor, record, and trend the quantity of nitrogen remaining in the bottles and therefore, the likelihood of nitrogen pressure failure leading to seal failure was much higher than concluded by the licensee.

D. Refueling Cavity Floor Seal

1. Refueling Cavity Floor Seal Design Application

The refueling cavity floor seal design at Surry is seismic category 1 and safety related. The seal is part of the refueling cavity pressure boundary. Refer to Appendix 3A and Figure 1 for a description of the refueling cavity floor seal design.

In an attempt to determine root cause failure, the AIT conducted a review of the available refueling cavity floor seal documentation. There is no design documentation available to describe the mechanics of how the overall seal assembly operated and no documentation available that verifies the design adequacy by testing.

In conversations with the vendor Presray it was determined that the licensee's design was unique in that the seal does not have a backing plate (located in the area identified as dimension B in Figure 1 of Appendix 3A). This backing plate would allow better contact between the "J" seal seating surface and its mating surface.

Initially, the vendor informed the licensee that with the current design configuration, the "J" seal could be easily displaced. This displacement was predicted to be a result of both buoyancy factors and the action of water flowing underneath the seal. Since the AIT inspection, the licensee has concluded that due to design tolerances not being controlled, the "J" seal could have a 1/8 inch gap between the seating surface and its mating surface. With this maximum gap, the "J" seal is flow limiting but to some value in excess of 6000 gpm.

In either case, the AIT concludes that the current design application is inadequate and that this condition has existed since initial installation. There are no design margins identified relating to the vertical and horizontal relationship between the vessel flange and the cavity floor. Thus, without periodic testing, it cannot be assumed that the seal would meet its design bases. Based on this evaluation, the AIT concludes that the licensee must reevaluate the present design of the seal ring. In addition, the seal must be tested to ensure continued compliance with the design bases.

2. Equipment Vendor Involvement

The IOER group contacted Presray on August 1, 1988, and the following items were discussed:

- "Presray stated that, in the current design configuration, the "J" seals could easily be displaced from the seating surface due to the action of water flowing underneath it."
- "Presray stated that the design of the licensee seal ring is unique and in their opinion, requires design improvements to hold the "J" seal in place with a backing plate. In addition, the inflatable seal should have a fiber reinforcement to improve strength and the surface contact area (footprint) should be increased."

- "The material used in the "J" seal should have improved resiliency and should be subject to a frequent inspection and replacement cycle."

- "The original design intent was to use the "J" seal as the primary seal with the inflatable seal as a backup and for "housekeeping" concerns."

- "Presray stated that they manufactured and continue to supply most of the refueling cavity seals used throughout the industry. This is the only seal, to their knowledge, that utilizes a "J" seal without a backing plate."

The AIT concluded that contact with the vendor was only accomplished by the IOER group during their followup investigation of the event. This action was taken several months after the event.

V. RADIOLOGICAL CONSEQUENCES

On May 17, 1988, a health physics (HP) technician providing continuous coverage for contract personnel noticed that the reactor cavity water level had decreased and that the radiation levels had increased from approximately 35mr/hr to 100mr/hr. The HP technician immediately evacuated the Unit 1 containment operating deck (47 foot elevation) and terminated the radiation work permit (RWP) under which the contract personnel were working.

The purpose of RWP 88-RWP-1507 issued on May 12, 1988, was to allow work on the upper internals package thermocouple lead conduit. Appendix 3D indicates how the water level in the refueling cavity varied during the event. The AIT reviewed this RWP and concluded that the appropriate precautions and requirements were adequately specified on the RWP to protect the health and safety of those personnel performing the work on the Unit 1 containment operating deck.

Radiation exposures to personnel were reviewed as a result of this event and noted that all exposures were well below NRC limits and the licensee's administrative limits. After the cavity water level was restored, the RWP was reinstated for normal access.

By reviewing the chart recorder for the Manipulator Crane Radiation Monitor (R1-RMS-162), the AIT determined that the remote read out in the control room did not reach the "Alert" setpoint of 35mr/hr during the event. The monitor was located above the reactor cavity. The setpoints for the radiation monitor, R1-RMS-162 had been changed to 35mr/hr for the "Alert" setpoint and 50mr/hr for the "Alarm" setpoint for refueling operations. The normal setpoints for routine operations are 120mr/hr on "Alert" an 600mr/hr on "Alarm". R1-RMS-162 was calibrated on April 16, 1988, as required by TS prior to removing the Unit 1 reactor vessel head.

The portable radiation survey instrument issuance log for May 17, 1988, was reviewed. During the time of the event it was noted that an operator who entered the Unit 1 containment was issued a survey instrument. The survey meter was adequate (greater than 1r/hr) to survey the high radiation area. The key issuance log for high radiation areas access was reviewed. The HP technician assigned to the Unit 1 containment to provide coverage for various tasks accompanied the operator who entered several high radiation areas and provided positive access control over each entry as required by TS 6.4.

Radiation, contamination and airborne radioactivity survey results for the 47 foot elevation and the -27 foot elevation were reviewed. The airborne radioactivity concentrations were all less than 25% of Maximum Permissible Concentration (MPC). Contamination levels on the 47 foot elevation of Unit 1 containment remained unchanged as a result of the event, i.e., 2,000 - 5,000 disintegrations per minute per one hundred square centimeters ($dpm/100cm^2$). However, in the lower containment, -27 foot elevation, the contamination levels increased from 2,000 - 15,000 $dpm/100cm^2$ to 4,000 - 20,000 $dpm/100cm^2$. This slight increase did not create a health and safety concern.

The personnel contamination log was reviewed for the period of May 16-17, 1988, and the event was discussed with licensee representatives. No personnel contaminations were attributed to this event.

The AIT was informed that the radioactive liquid that drained from the reactor cavity was contained in the incore instrument room sump or in the containment sump. The water was later pumped to the High Level Liquid Waste Tanks and processed as normal radioactive waste.

VI. FINDINGS OF THE AIT

A. Radiological Consequences

- The failure of the Reactor Cavity Seal did not result in any radiological releases to the environment which exceeded regulatory limits.

- Radiation doses received by individuals involved in the event were all below regulatory limits. The one operator who was wetted by the refueling cavity water was surveyed and the water in the sump was sampled and counted for radioactivity. No intakes of radioactivity or personnel contamination resulted from the event.

- Under normal refueling conditions had the seal failed the potential existed for significant personnel exposure had a fuel assembly been in the transfer position (i.e., suspended from the refueling bridge).

- The licensee's UFSAR Chapter 14, "Safety Analysis," does not address the accident or consequences due to loss of refueling cavity or spent fuel pool water level.

B. Failure Investigation

The licensee did not perform a failure evaluation or investigation following the event. An investigation was commenced in July by the IOER group of the event.

C. Modifications

- No documentation exits to support the design and/or installation of the nitrogen system on Unit 1.

- Check valves to prevent backflow and overpressure protection devices installed in Unit 2 nitrogen system are not installed in the Unit 1 system.

- Procedure revisions to include pressure regulators and relief valve settings and testing were not implemented for either unit.

- EWR-85-200 dated April 1985 for Unit 2 recommended procedure revision to change IA pressure regulator settings to 25 psig versus 20 psig. The current revision for procedure MMP-C-RC-037 used for both units still indicates 20 psig.

D. Installation and Test of Refueling Cavity Floor Seal

- The inflatable seal failed to meet the acceptance criteria established for the preinstallation pressure test during the 1988 Unit 1 refueling outage. This test is required by MMP-C-RC-37, Installation and Removal of reactor Cavity Seal Ring.

- Visual inspections performed for seal degradation and of the "J" seal seating surfaces, again a preinstallation requirement of MMP-C-RC-37 noted several deficiencies.

- The licensee evaluated all of the aforementioned conditions as being acceptable under EWRs 116 and 148.

- MMP-C-RC-37 provides no guidance on:

    - Installation and/or removal of the nitrogen bottles; and

    - setting and testing of the relief valves and/or check valves.

E. Local Leak Rate Test

- Operation of system (nitrogen and IA) valves and regulators outside the boundaries of PT 16.4 were performed without procedures.

- Independent Verification was performed on nitrogen and IA system valves and regulators without documenting actions.

- No procedural method or documentation was implemented or developed for the repair of valve 1-IA-849 performed on May 16 and 17, 1988.

F. Inadequate Instructions and Drawings

- Current abnormal procedures for addressing a decrease in refueling cavity level are inadequate.

The following concerns apply to the nitrogen and IA systems:

- limits and precautions to prevent overpressurization and rupture of the inflatable seal are not available to operators,

- no provision to control valve positions (i.e., locks, tags),

- no directions or setpoints for adjusting the pressure regulators, pressure either high or low,

- no method or procedure for establishing the preferred regulator and nitrogen source, and

- no lower setpoint limit of nitrogen bottle pressure.

- no logkeeping requirements when nitrogen bottle pressures are monitored, and

- no drawing to indicate system configuration for either system.

G. Training

The following are noted training findings:

- the nitrogen back-up system was poorly understood in its design, layout, operation, operational limits and precautions;

- operational features of the refueling cavity floor seal design were not understood by operations personnel; and

- no training on emergency procedures to mitigate refueling cavity floor seal failure had been implemented.

VII. GENERIC IMPLICATION OF SEAL FAILURE

Plants with designs similar to Surry have responded to Bulletin 84-03 as Surry did, basically eliminating catastrophic seal failure as a credible failure mode because of the passive "J" seal function. However, this event indicates that a significant failure can occur even with the passive seal. The vendor has indicated that plants using the passive seal design are not likely to have a similar failure because of a backing plate which tends to maintain a more uniform seating surface between the seal and its mating surface (as discussed in Section IV.D. of this report). The licensee could not locate documentation of any acceptance tests (including initial preoperational tests) that verified the passive "J" seal assembly had ever been demonstrated or tested to meet its design bases.

It is appropriate to require plants with similar "J" seal designs to verify through functional test that the original design intent of the seal is maintained. Tests after each installation need to be performed to assure proper installation and integrity of the seals.

VIII. ROOT CAUSE DETERMINATION

The apparent root cause of the inflatable seal failure was due to securing the IA supply to the seal for maintenance with a subsequent loss of nitrogen pressure from the backup system. The loss of nitrogen pressure occurred because one bottle was somehow isolated in that the regulator was misadjusted while the second bottle (which was unisolated with the regulator adjusted properly) bled down in some manner.

The "J" seal root cause failure is much more difficult to determine because there is no assurance that the "J" seal was ever completely functional. Therefore, a design application deficiency may have contributed to the failure. Also, dimension changes between the reactor vessel flange (either vertical or horizontal) may have contributed to or caused the inability of the seal to perform its intended design function. Additionally, in May of 1986 the "J" seal was replaced. There are no specific procedures for replacing or repairing the seal. Replacement and repairs were made using the associated design drawings. This is another possible root cause of the seal failure if the replacement was improperly performed and resulted in the "J" seals not being installed in accordance with the original design.

IX. CONCLUSIONS

The overall conclusion of the AIT is that the root cause of the seal assembly failure was a combination of inadequate administrative controls, operator error, coupled with inadequate design application, maintenance and testing of the "J" seal assembly. The primary root cause of the "J" seal failure appears to be design related. Inadequate maintenance, testing, and installation procedures may have contributed to the severity of the event. Operator error was induced by inadequate operator aids and training.

Adequate functional testing of the "J" seal would have discovered the inadequacy of the initial design application and its ability to perform its design function.

X. EXIT INTERVIEW

The findings and conclusion of this special inspection were discussed on September 3, 1988, with those persons indicated in Appendix 1. No dissenting comments were received.

APPENDIX 1 - PERSONS CONTACTED

Licensee Employees

* J. Bailey, Superintendent of Operations
* R. Bilyeu, Licensing Engineer
* D. Benson, Station Manager
H. Blake, Superintendent of Site Services
R. Bracey, Control Room Operator (Unlicensed)
* W. Cartwright, Vice President-Nuclear
B. Cox, Control Room Operator (Unlicensed)
* S. Eisenhart, Staff Engineer, Independent Offsite Evaluation Review
* E. Grecheck, Assistant Station Manager for Licensing and Safety
M. Hotchkiss, Shift Supervisor
R. Johnson, Operations Supervisor
T. Kendzie, Containment Coordinator
* J. Logan, Supervisor, Safety Engineering Staff
* G. Miller, Licensing Coordinator, Surry
* H. Miller, Assistant Station Manager for Operations and Maintenance
* L. Morris, Supervisor, Health Physics and Radwaste
R. Mushenheim, Control Room Operator (licensed)
* G. Pannell, Director, Safety Evaluation and Control
W. Patterson, Human Performance Evaluation System Coordinator, Surry Power Station
* T. Shaub, Licensing Engineer
J. Simpson, Shift Supervisor
K. Sloane, Shift Supervisor
* J. Smith, Supervisor, Independent Offsite Evaluation Review

NRC Employees

L. Nicholson, NRC Resident Inspector

* Attended exit interview on September 3, 1988.

APPENDIX 2 - ACRONYMS AND ABBREVIATIONS

AFW	Auxiliary Feedwater
AIT	Augmented Inspection Team
AP	Abnormal Procedure
CRO	Control Room Operator
CS	Containment Supervisor
DR	Deviation Report
EST	Eastern Standard Time
EWR	Engineering Work Request
HP	Health Physics
HPES	Human Performance Evaluation System
IA	Instrument Air
IOER	Independent Offsite Evaluation Review
JCO	Justification for Continued Operation
LHSI	Low Head Safety Injection
LLRT	Local Leak Rate Test
MPC	Maximum Permissible Concentration
NLO	Non-Licensed Operator
PG	Primary Grade
PT	Periodic Test
PWR	Pressurized Water Reactor
QC	Quality Control
RHR	Residual Heat Removal
RS	Refueling Supervisor
RWP	Radiation Work Permit
SD	Station Deviation
SNSOC	Station Nuclear Safety Operations Committee
SS	Shift Supervisor
STA	Shift Technical Advisor
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USS	Unit Shift Supervisor
UTS	Unit Test Supervisor
W	Westinghouse Electric Corporation

APPENDIX 3 - DESIGN DESCRIPTIONS

APPENDIX 3A - REFUELING CAVITY FLOOR SEAL

GENERAL DESCRIPTION

The refueling cavity floor seal (Figure 1) is intended to seal the opening between the reactor vessel flange and the refueling cavity floor. This allows the refueling cavity to be filled with borated water so that refueling operations can be accomplished under water.

The seal assembly consists of two separate sealing devices; an active or inflatable seal and a passive or "J" seal.

The inflatable seals are manufactured from a nitride rubber material and are designed to seal against a hydrostatic head of 27 feet of water. A design operating pressure of 25 psig is specified under ambient conditions of 60°F to 120°F. The design pressure is 50 psig.

Figure 1 shows the inflatable seal in both the inflated and deflated conditions (inner seal deflated, outer seal inflated). Compressed air or nitrogen is introduced into the inflatable seal via air connections on the bottom of the seal ring. The seal ring contains two air passages which direct the air to the seal.

The "J" seals provide a passive sealing function and are intended to minimize and/or preclude leakage in case of inflatable seal failure. The "J" seals are fabricated from a high grade, thread-type natural rubber compound. They are 7/8-inch in diameter with a 3/8-inch hollow inner core. When the assembly is lowered into place, the seal supports are required to be adjusted to achieve a 1 3/16-inch gap (dimension "A", Figure 1).

Permanently attached to the vessel flange and refueling cavity floor are drip pans which collect leakage past the seals. This leakage is directed to the reactor coolant loop rooms, through the telltale drains to the containment sump. The drip pans and associated small drain lines (3/4-inch) are capable of handling small leakage by the seals.

APPENDIX 3B - INSTRUMENT AIR SYSTEM

DESCRIPTION

The containment IA system (Figure 2) consists of two water-sealed, rotary compressors and associated refrigerant air driers installed on the 11'6" elevation of the main steam valve buildings for Units 1 and 2. The compressors take a suction from the containment via a 3" penetration. Containment trip valves are provided on both sides of the penetration. Each compressor has a minimum capacity of 24-scfm at 90 psig. A shell and tube heat exchanger is provided on each compressor to cool the seal water. Cooling water for these heat exchangers comes from the containment cooling chilled water system. The alternate supply of cooling water is the component cooling system. A connection to component cooling water is also provided for seal-water make-up. One compressor is in continuous service and automatically loads or unloads to meet system demand. The other compressor is on standby and starts automatically if system pressure decreases to 85 psig.

Each compressor discharges to its own moisture separator and filter. Water removed from the air by the separators and air driers is directed to a sump, where a small sump pump transfers the water to the liquid waste system. Each air compressor discharges to its own refrigerant air drier. The piping allows the air compressors to be cross-connected with the air driers as well as allowing them to bypass the driers completely. Air exiting the driers has a dewpoint of 35°F. The air enters the containment through a containment trip valve using containment penetration 47 for Unit 1 IA.

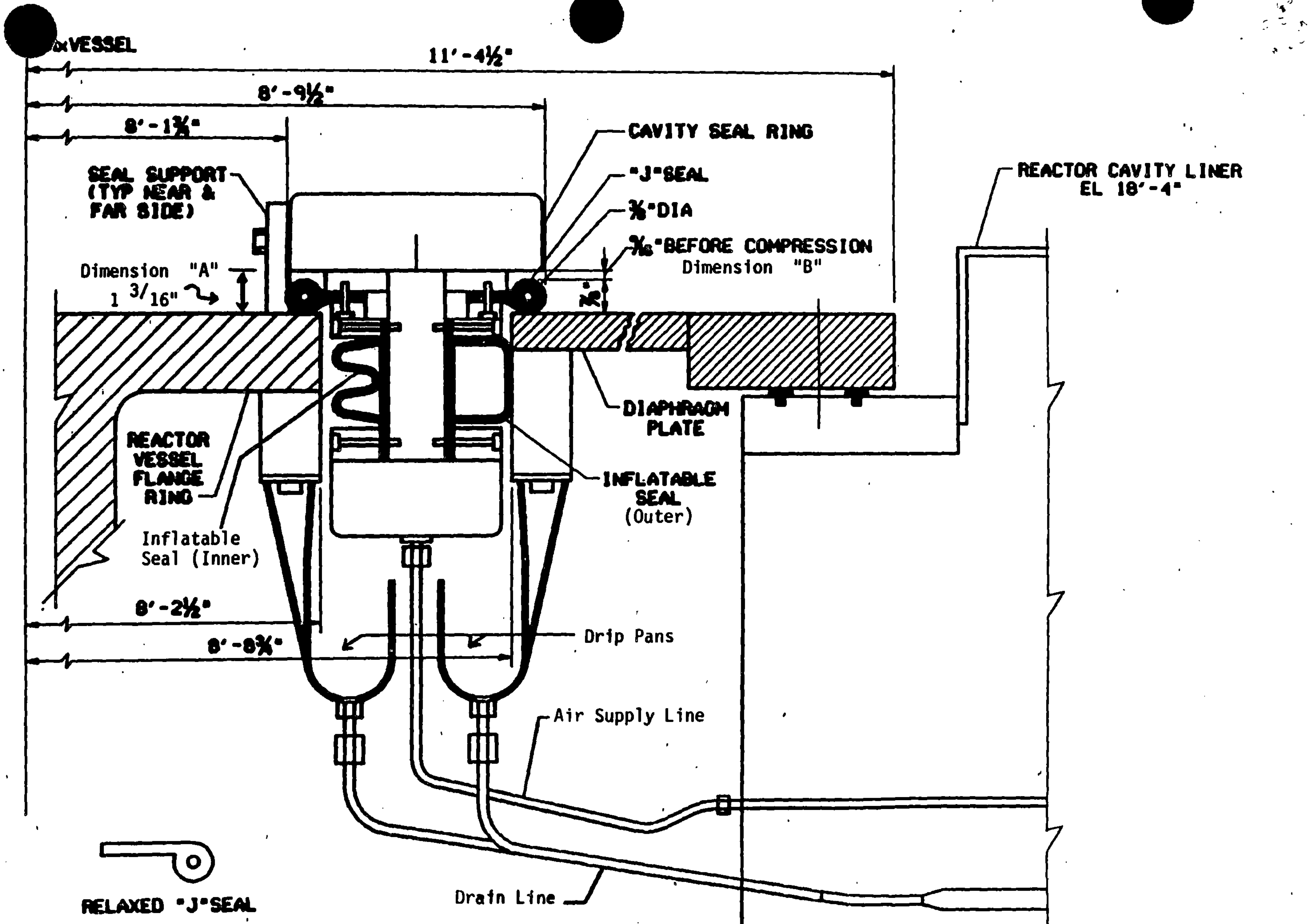
APPENDIX 3C - NITROGEN BACK-UP SYSTEM

The nitrogen back-up system, as shown in Figure 2 (configuration based on personnel interviews), consists of two portable nitrogen cylinders each containing 301 cubic feet of nitrogen at approximately 2200 psig when full. These bottles supply nitrogen to 2200/20 psig variable pressure regulators. Flexible tubing connects the downstream pressure of the regulator through an isolation valve to a junction from the containment IA system. This nitrogen pressure supply is then supplied to the refueling cavity floor seal if IA pressure is not available. The pressure regulators on the nitrogen bottles should be set at 20 psig and the IA pressure regulator should be set at 25 psig.

APPENDIX 3D - UPPER CORE INTERNALS STORAGE

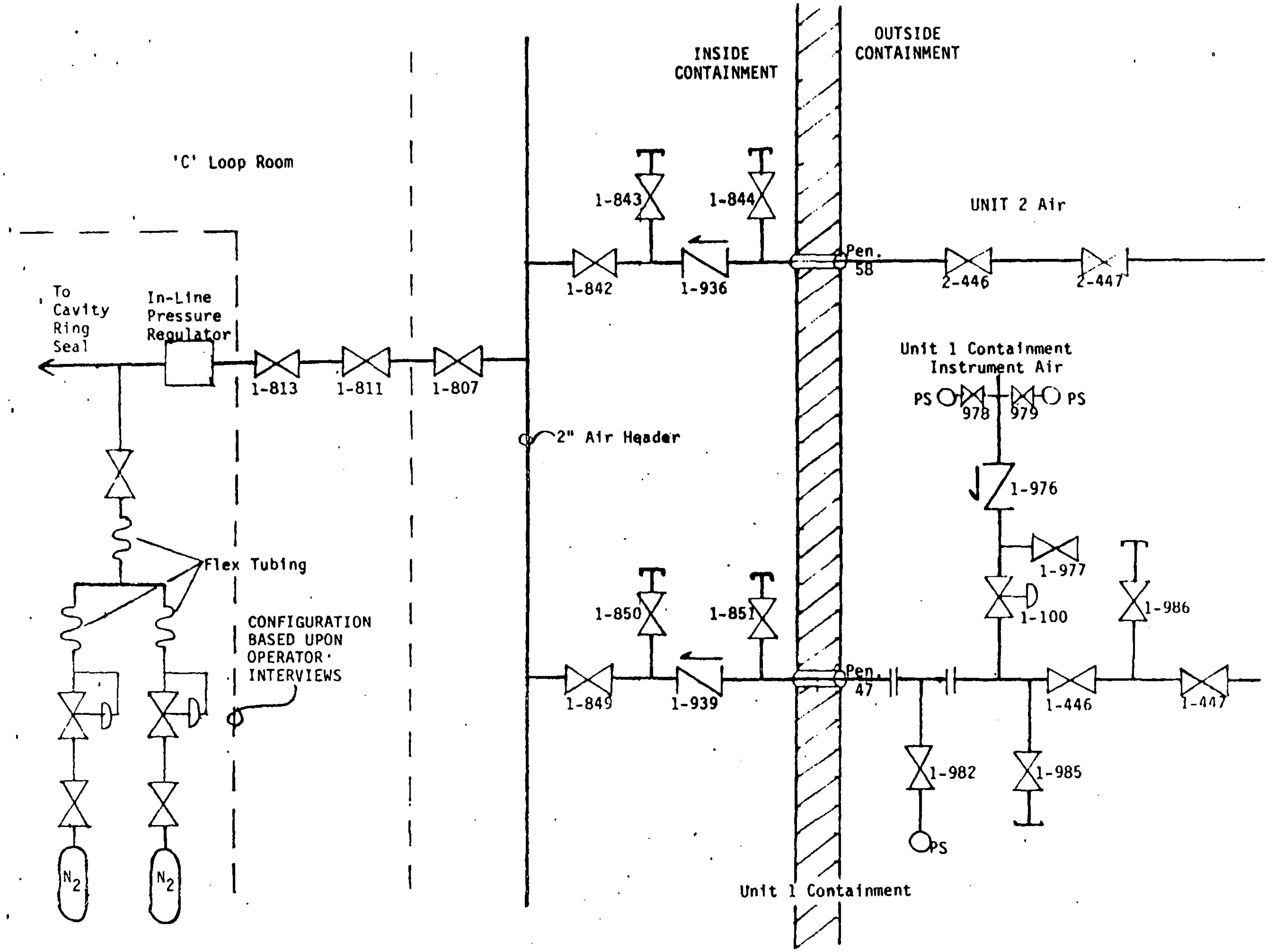
As shown on the attached drawing, Figure 3, the upper internals package (item 1) rests in the storage area on a stand (item 2) which holds the internals up some 6" off the bottom of the reactor cavity. The total height of the upper internals package, including the 6" offset provided by the stand, is 26'. When the refueling cavity water level is "normal" (item 3) the top of the upper internals package is about 1'6" below the surface of the water.

On March 17, 1988, W contractor personnel were working on upper internals thermocouple lead conduit. This work was being performed from the refueling bridge positioned directly over the upper internals package storage location. The water level had been reduced about one foot below (item 4) the normal level thus reducing the remaining shielding to six inches. Due to the loss of refueling cavity water through the refueling cavity floor seal, this shielding water was reduced an additional three feet. Following the event the water level could have dropped to approximately twenty-four feet. Thus allowing about two feet six inches of the upper internals package to be out of the water.



SURRY REACTOR CAVITY SEAL RING ASSEMBLY

Figure 1



Instrument Air System/Nitrogen Backup System

Figure 2

UPPER CORE INTERNALS STORAGE

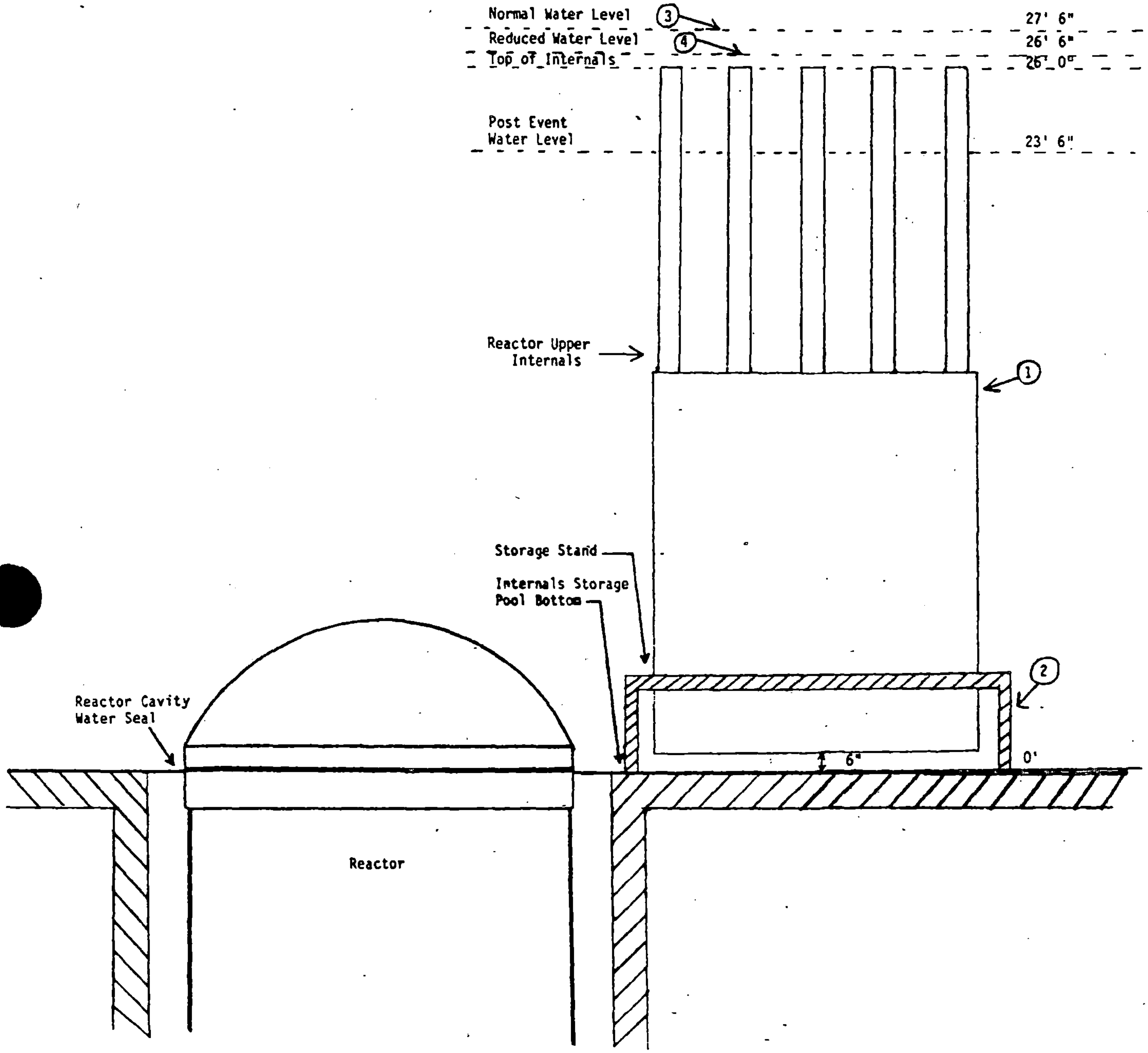


Figure 3