## ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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| ATTACHMENTS: | 1. Supplemental Information<br>2. Presentation material from July 2, 1998, meeting |

# EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/98-13

## Operations

- The licensee was well prepared for plant restart from the 1998 refueling outage as evidenced by proper closure of outage activities, completion of required Technical Specification (TS) surveillances, and adequate configuration of plant systems to support power operation. This was improved performance over previous refueling outages (Section O1.1).

- Control room operators took appropriate steps to limit outside interference and maintain control of the plant during the performance of postmaintenance testing on the reactor feedwater pumps following modifications to their associated hydraulic control system. Effective command and control and three-way communication were observed (Section O1.2).

- Poor procedure use during the restoration from an inadvertent engineered safety feature (ESF) actuation resulted in the mispositioning of the minimum flow bypass valve for the low pressure core spray (LPCS) system. Numerous control board walkdowns performed by operators failed to identify the discrepancy. A violation of TS 5.4.1.a was identified for failure to follow procedure when returning the low pressure core spray system to its standby lineup (Section O2.1).

## Maintenance

- The licensee's actions were comprehensive in identifying and inspecting equipment in the emergency core cooling system pump rooms that was affected by the June 17 flooding event. Efforts to dry equipment and conduct calibrations and functional tests were sufficient to verify operability. However, walkdown inspections of the fire protection system were weak in that subsequent to the walkdowns the inspectors identified ten failed system pressure gauges and a loose pipe hanger on the standby gas treatment system deluge supply piping (Section M1.1).

- A cognitive error on the part of maintenance personnel installing the traversing incore probe (TIP) instrument tubing resulted in the separation of the undervessel connection on one of the 41 tubes. Consequently, the drive cable for one of the probes became mechanically bound when it was inadvertently spooled into the undervessel area during a system alignment. The failure of the drive cable precluded the ability to close its associated containment isolation ball valve and necessitated a plant shutdown in accordance with TS (Section M1.2).

Engineering

- The licensee has maintained an appropriate program to address the requirements of 10 CFR 50.59. Program implementing procedures were generally of sufficient detail to ensure that proposed activities would precipitate safety evaluations (SEs). However, two areas were noted where procedure guidance was either weak or inconsistent with requirements. Although the quality of the 12 SEs reviewed was not always consistent, overall the quality was good. Strengths were noted in the training and oversight programs with regards to maintaining a sufficient pool of qualified SE preparers and providing timely, critical feedback on their products (Section E1.1).

- The configuration of the reactor building equipment drains did not conform to the description in the Final Safety Analysis Report (FSAR) in that a cap was not installed on the drain line from residual heat removal (RHR) pump Room B. The cap was required as part of the licensee's physical controls to protect against common mode flooding. A violation of 10 CFR 50.59 was identified for failure to document a written SE for this defacto change to the facility. The licensee's corrective actions to install a cap on the drain line and review the generic implications for other portions of the drain systems, were found to be appropriate (Section E1.2).

- Compensatory and corrective actions taken to address design deficiencies in the fire protection system and minimize dynamic loads were generally appropriate. However, the licensee's evaluation of the modified system's performance failed to identify a vulnerability to waterhammer following a loss of offsite power. The vulnerability was adequately addressed when the system configuration was modified to maintain a diesel-driven fire water pump operating (Section M1.1).

Plant Support

- As-low-as-reasonably-achieveable (ALARA) planning for the troubleshooting and repair of a traversing incore probe drive cable was effective in evaluating the potential radiological hazards and communicating them to the involved personnel. Good radiological controls practices and health physics support also contributed to dose reduction for the work (Section M1.2).

## Report Details

### Summary of Plant Status

The plant began the inspection period in Mode 4, completing activities from Refueling Outage R13. The plant entered Mode 2 on June 8 to begin Cycle 14. On June 15, power ascension was halted at approximately 35 percent and the plant was returned to Mode 4 when it was discovered that a TIP drive cable was mechanically bound and could not be withdrawn. On June 17, repairs to the TIP system were completed. However, prior to plant restart, a fire main ruptured in a reactor building stairwell resulting in significant flooding and equipment impact in the RHR Train C and LPCS pump rooms. The plant remained in Mode 4 while the licensee implemented recovery actions for the damaged equipment and interim corrective actions for the fire water system.[1]

With recovery actions completed, the plant reentered Mode 2 on July 3. The plant achieved full power operation on July 8 and remained there for the balance of the inspection period.

## I. Operations

### O1 Conduct of Operations

#### O1.1 Plant Startup From Refueling Outage R13

a. Inspection Scope (71707)

The inspectors reviewed the licensee's readiness for restart of the plant following Refueling Outage R13. The review included verification on a sampling basis of TS-required surveillances, closure of maintenance activities, and walkdowns of selected systems to independently verify proper configuration for return to power operations. Portions of the startup evolution were also observed.

b. Observations and Findings

A review of randomly selected TS surveillances found that each was current for transition of the plant to power operations. An independent verification of selected prerequisites for plant startup, as defined in Procedures 3.1.2, "Reactor Plant Startup," Revision 42, and 3.1.1, "Master Startup Checklist," Revision 23, was also performed, including a control room board walkdown. With the exception of the LPCS system, prerequisites were properly completed and plant systems were aligned for plant startup. A misaligned valve in the LPCS system was identified during the control board

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[1]The facts surrounding the fire main flooding event are documented in NRC Augmented Inspection Team Report 50-397/98-16.

walkdown. The misalignment did not render the LPCS system inoperable. This issue is discussed further in Section O1.2. Overall, the licensee's readiness for restart demonstrated improved performance from that observed during previous refueling outages.

Observation of portions of the startup found that the evolution was properly briefed and controlled.

c. Conclusions

The licensee was well prepared for plant restart from the 1998 refueling outage as evidenced by proper closure of outage activities, completion of required TS surveillances, and adequate configuration of plant systems to support power operation. This was improved performance over previous refueling outages.

O1.2 Reactor Feedwater System Postmaintenance Testing

a. Inspection Scope (61726)

The inspector observed the control room staff conduct postmaintenance testing on the reactor feedwater system.

b. Observations and Findings

On July 7, 1998, the inspector observed the licensee conduct Procedure 8.3.386, "Test Instructions - RFW Governor Post Maintenance Test and Tuning," Revision 1. Just prior to performing the test, the licensee modified the reactor feedwater pump hydraulic control system during a planned maintenance outage. The licensee used Procedure 8.3.386 to verify proper operation of the reactor feedwater pumps and control system.

During the performance of the test the inspector noted that the control room staff was knowledgeable and well prepared. The operators took steps to limit interference from outside the control room during the test, predetermined specific assignments during procedural steps that could produce plant transients, and established formal contingency plans for possible reactor feedwater system malfunctions. The inspector observed that operators maintained good command and control and used effective three-way communication. This preparation allowed operators to focus their attention on the plant during the procedure and demonstrated their preparedness to deal with undesired system response.

c. Conclusions

Control room operators took appropriate steps to limit outside interference and maintain control of the plant during the performance of postmaintenance testing on the reactor feedwater pumps following modifications to their associated hydraulic control system. Effective command and control and three-way communication were observed.

## O2 Operational Status of Facilities and Equipment

### O2.1 ESF System Walkdowns (71707)

a. Inspection Scope

The inspectors walked down accessible portions of the following systems:

- LPCS
- RHR A, B, and C
- 4160V Critical Switchgear
- Standby Liquid Control

b. Observations and Findings

With the exception of the LPCS system, each of the systems was found to be properly aligned for the plant conditions at the time of the walkdown. Where required, locks were found properly installed. No material condition deficiencies were identified. Excluding the June 17 flooding event, which impacted the availability of the RHR C and LPCS systems, the reliability and availability of these systems remained high (i.e., well within the assumptions of the plant's probabilistic safety assessment).

On June 7, 1998, a control room board walkdown performed by the inspectors as part of the verification of plant readiness for restart from Refueling Outage R13 identified that the minimum flow bypass valve for the LPCS system was open, contrary to the system's standby alignment defined in Section 5.2 of Procedure 2.4.3, "LPCS System," Revision 17. In response to this finding, the operating crew reverified the alignment of the emergency core cooling system prior to entering Mode 2. No other discrepancies were identified. The licensee also reviewed the operational history of the LPCS system and found that the minimum flow bypass valve was automatically opened in response to an inadvertent ESF actuation signal that was initiated on May 30, due to maintenance. During plant restoration from that event, operators failed to reference Procedure 2.4.3 for returning the LPCS system to its standby lineup. Subsequently, 16 shift turnovers, each including board walkdowns by the oncoming and offgoing control room staff, and a panel walkdown performed by the control room supervisor and shift manager, as required for plant startup by Procedure 3.1.2, failed to identify the mispositioned valve.

The function of the minimum flow bypass valve is to provide limited pump flow for cooling the LPCS pump when the pump is operating without its normal flow path available (e.g., reactor pressure greater than pump shutoff head). The valve repositions automatically in response to system flow. The operability of the valve is periodically verified through a TS-required surveillance. Therefore, based upon the ability of the minimum flow bypass valve to automatically reposition as needed, the mispositioning of

the valve did not impact the ability of the LPCS system to perform its safety function. However, both the failure to restore the valve to its standby position following the ESF actuation and the subsequent missed opportunities to identify the discrepancy indicated weaknesses in procedure use and operator attentiveness. The failure to maintain the minimum flow bypass valve in the closed position, as required by Procedure 2.4.3, was identified as a violation of TS 5.4.1.a (VIO 50-397/98013-01).

To address the performance issues regarding procedure use and board walkdowns, the licensee implemented and planned several corrective actions. The crew involved with the mispositioning of the LPCS valve was counseled on the importance of proper system restoration with verification, as necessary, through the Volume 2 system operating procedures. The license also identified weak use of the Volume 2 procedures in the simulator during simulated abnormal operating conditions. Expectations have been reemphasized and performance criteria revised in this area during simulator training. In evaluating the failure of operators to subsequently identify the mispositioned valve, the training department included a board awareness scenario in the most recent training cycle. The scenario included the removal of control power for Valve RHR-V-64C (RHR C minimum flow bypass valve) which eliminated valve position indication on the control board. Several crews failed to identify the discrepancy, highlighting the need to improve human factors aspects for control board indications. The licensee is implementing a program to address this area which will include consideration of visual discriminators for components out of their normal alignment. The inspectors considered these corrective actions to be appropriate to address the performance weaknesses that led to the violation.

c. Conclusions

Poor procedure use during the restoration from an inadvertent ESF actuation resulted in the mispositioning of the minimum flow bypass valve for the LPCS system. Numerous control board walkdowns performed by operators failed to identify the discrepancy. A violation of TS 5.4.1.a was identified for failure to follow procedure when returning the LPCS system to its standby lineup.

## II. Maintenance

### M1 Conduct of Maintenance

M1.1 Recovery Actions For Fire Main Flooding in the Reactor Building

a. Inspection Scope (62707, 61726)

The inspectors observed and/or reviewed the licensee's work activities to recover plant equipment and prepare the plant for restart following the June 17, 1998, fire main flooding event. This review included independent system walkdowns and observations. The following work activities were also reviewed:

- OSP-RHR/IST-Q704, Revision 2, "RHR Loop C Operability Test"
- OSP-LPCS/IST-Q702, Revision 3, "LPCS System Operability Test"
- Work Order # KTJ1, Repair of Valve FDR-V-609
- Work Order # MBP0, Assessment and Repair of Pump Motor RHR-M-P-2C
- Procedure 8.3.403, Fire Protection Water Hammer Test
- Selected Work Orders for Drying, Inspecting, and Testing Flooded Equipment
- Temporary Modification Request 98-20, Reactor Building Firemain Riser Nitrogen Blanket

b. Observations and Findings

Equipment Recovery

Both the Division I and II keepfill system pumps were replaced shortly after the event. The prioritization of this repair allowed the licensee to restore the normal keepfill function and return one of the RHR cooling loops to a normal standby lineup.

The inspectors walked down RHR pump Room C, the LPCS pump room, and applicable stairwell/hallway spaces on June 28 and 29. These walkdowns included visual inspections after the licensee removed various conduit covers, pressure switch covers, electrical junction and contactor covers, and opened breaker boxes. The inspectors also reviewed various work orders conducted to verify operability of affected components and an interoffice memorandum dated June 29, 1998, from J. E. Parker, Supervisor of Project Controls, "Certification for Restart Following Flood of ECCS Rooms."

The components (including mechanical, civil, and electrical) located in the affected spaces were adequately inspected, dried, if appropriate, and checked for operability. The RHR Pump C motor was sent to a vendor facility to be more thoroughly dried, verified as undamaged, and subsequently reinstalled. A review of the postmaintenance testing on the motor found that the testing was adequate to demonstrate operability. Where appropriate, other components were replaced. Under Work Order MCF6, a technician recommended replacing the contactor for the LPCS pump motor heaters. This recommendation was not evaluated by licensee supervision until brought to their attention by the inspectors. The old contactor had been dried, cleaned, reinstalled, checked for resistance to ground, and placed in service (all indicating acceptable performance). Following the inspectors' question, the licensee replaced the contactor, based on the initial recommendation.

Fire Protection System Inspection and Restoration

Due to the identified susceptibility of the fire protection system to significant dynamic loads generated by waterhammer, the licensee conducted a comprehensive visual inspection of the system and implemented several corrective actions and compensatory measures. The inspectors conducted independent walkdowns of the fire protection system to verify that visual damage to equipment had been identified and was being properly tracked for resolution.

On June 26, the inspectors identified 8 pressure gauges associated with the standby gas treatment and reactor building exhaust air deluge systems that had apparently failed due to overranging. The inspectors also identified a hanger associated with the standby gas treatment deluge system supply piping whose baseplate anchors were loose (i.e., finger tight). None of the items were being tracked through the licensee's work control process. Subsequently, the inspectors identified that two pressure gauges on the deluge supply to the control room emergency charcoal filters had also failed. Again, these gauges were not being tracked for resolution. Further review found that none of the identified deficiencies impacted operability of the fire protection system. However, the number of deficiencies identified by the inspectors indicated weak performance in the licensee's efforts to inspect the system and identify equipment requiring maintenance.

Recognizing the potential for significant dynamic loads at the bottom of the reactor building fire main risers, the licensee replaced the riser isolation valves (including the valve that failed) with valves fabricated from cast steel.

In evaluating the root cause of the flooding event, the licensee determined that the waterhammer was the result of the inability of the fire system to maintain pressure immediately following a large system demand. That is, the operating jockey pump, with the subsequent start of main system pumps on low pressure, was insufficient to preclude short-term voiding in the upper portions of the system (i.e., reactor building risers). To minimize voiding of the system upon initial system demands, the licensee implemented a compensatory measure to maintain both electric motor-driven fire water pumps operating. Because voiding in the reactor building risers could not be completely eliminated with this measure, the licensee also implemented a temporary modification to apply a small nitrogen blanket on the top of the risers. The nitrogen provides a compressible volume of gas to dampen system dynamic loads.

The inspectors found that the licensee's corrective actions were properly implemented and controlled. With the exception of the fire water pump configuration, the corrective actions, coupled with the compensatory measures, were found to be adequate to address the identified deficiencies in the fire protection system. In regards to the fire water pumps, the inspectors raised a concern with the ability of the modified configuration to withstand a loss of offsite power. Specifically, with the unavailability of the normal transformer, due to the plant being shut down, a loss of the startup transformer would result in the loss of both electric motor-driven fire water pumps. A subsequent system demand could potentially lead to significant voiding in the reactor building risers prior to the automatic starting of the diesel-driven fire water pumps. The licensee agreed that the scenario presented a vulnerability to system operation and modified their compensatory measure to have one of the two operating fire water pumps be a diesel-driven pump. Following plant startup, with the normal transformer placed in service, the configuration was returned to two electric motor-driven pumps operating.

Fire Protection System Testing

To verify the adequacy of the short-term actions taken for the fire protection system, the licensee tested the system's response to both a single and dual preaction initiation. The testing was performed in accordance with Procedure 8.3.403, "Fire Protection Water Hammer Test," Revision 0. A review of Procedure 8.3.403 found that it established sufficient conditions to bound the design conditions that could be experienced by the fire protection system. By analysis, the licensee determined that the bottom of the reactor building fire main risers experience the highest dynamic loads in the system. As such, strain gauges were installed at these points to evaluate acceptability of the dynamic loading. The acceptance criterion established for the strain at these points was found to be adequate to demonstrate operability under design conditions. The testing was generally well controlled with test results well within the established acceptance criterion.

c. Conclusions

The licensee's actions were comprehensive in identifying and inspecting equipment in the emergency core cooling system pump rooms that was affected by the June 17 flooding event. Efforts to dry equipment and conduct calibrations and functional tests were sufficient to verify operability. However, walkdown inspections of the fire protection system were weak in that subsequent to the walkdowns the inspectors identified ten failed system pressure gauges and a loose pipe hanger on the standby gas treatment system deluge supply piping.

Compensatory and corrective actions taken to address design deficiencies in the fire protection system and minimize dynamic loads were generally appropriate. However, the licensee's evaluation of the modified system's performance failed to identify a vulnerability to waterhammer following a loss of offsite power. The vulnerability was adequately addressed when the system configuration was modified to maintain a diesel-driven fire water pump operating.

M1.2 Failure of TIP Drive Cable to Retract

a. Inspection Scope (62707)

On June 15, the licensee performed a normal shutdown of the plant when it was identified that the drive cable for TIP Machine B would not retract past its associated containment isolation valve. The inspectors reviewed the licensee's actions to address the TIP failure, including troubleshooting and repair activities and their associated radiological controls.

b. Observations and Findings

On June 14, during performance of alignment checks of the TIP system, the drive cable to TIP Machine B became mechanically bound and the probe would not retract to its shielded position. Operators appropriately identified that the cable precluded closure of the TIP Machine B's containment isolation ball valve and entered the TS action statement for an inoperable containment isolation valve. The licensee determined not to isolate the penetration with the redundant isolation valve, an explosive squib valve, and commenced an orderly shutdown to Mode 4. Completion of all TS-required actions was verified. The licensee properly reported the TS-required shutdown in accordance with 10 CFR 50.72 and 50.73 (Licensee Event Report (LER) 50-397/98-010-00).

Upon initial entry into the drywell, the licensee identified that the selected TIP tube for drive Cable B had become disconnected from its associated local power range monitor (LPRM) instrument tube under the reactor pressure vessel. The drive cable had spooled out under the vessel and become entangled with other equipment. The probe itself was located approximately 3 feet below the undervessel work platform. Both the drive cable and probe were subsequently replaced due to damage to the cable.

ALARA planning and radiological controls for the investigation and repair of the TIP Machine B drive cable were very good. The potential for high dose rates generated by the TIP was properly evaluated and personnel were well briefed on expected conditions and contingencies. The use of video taping on the initial drywell entries proved to be a valuable tool in developing repair actions. Health physics coverage for the work was excellent and supported good radiological controls practices.

In evaluating the root cause of the TIP failure, the licensee determined that the disconnected tube had been installed backwards. This resulted in alignment problems between the TIP tube and the local power range monitor instrument tube that precluded proper fit-up. Inadequate self-checking on the part of maintenance personnel reinstalling the tubing failed to identify and correct the alignment problem. Subsequently, the tube became disconnected during alignment of the TIP system. The licensee reinspected all of the TIP tube connections in the drywell and did not identify any additional discrepancies. The importance of self-checking was also reemphasized to the individuals involved and the maintenance organization in general. The inspector agreed with the licensee's conclusion that this event was the result of a cognitive error and found that corrective actions were appropriate. As such, LER 50-397/98-010-00 is closed.

c. Conclusions

A cognitive error on the part of maintenance personnel installing the TIP instrument tubing resulted in the separation of the undervessel connection on one of the 41 tubes. Consequently, the drive cable for one of the probes became mechanically bound when it was inadvertently spooled into the undervessel area during a system alignment. The failure of the drive cable precluded the ability to close its associated containment isolation ball valve and necessitated a plant shutdown in accordance with TS.

ALARA planning for the troubleshooting and repair of a TIP drive cable was effective in evaluating the potential radiological hazards and communicating them to the involved personnel. Good radiological controls practices and health physics support also contributed to dose reduction for the work.

## III. Engineering

### E1 Conduct of Engineering

E1.1 WNP-2 Licensing Basis Impact Determination (LBID) Process

a. Inspection Scope (37001)

The inspector examined the licensee's process for evaluating changes to the facility in accordance with 10 CFR 50.59. The inspection included a review of procedures that support the design change process and a random selection of 12 (SEs) associated with specific plant modifications. The inspector also interviewed cognizant licensee personnel and attended a Plant Operations Committee (POC) meeting.

b. Observations and Findings

Procedures and Controls

The inspector reviewed the following procedures:

- Plant Procedure Manual (PPM) 1.3.9, "Temporary Modifications"
- PPM 1.3.42, "Troubleshooting Plant Systems and Equipment"
- PPM 1.3.43, "LBIDs"
- PPM 1.4.1, "Plant Modifications"
- PPM 1.4.5, "Processing of Licensing Document Changes"
- SWP-IRP-01, "POC"
- SWP-PRO-02, "Preparation, Review, Approval and Distribution of Procedures"
- Engineering Instruction 2.8, "Generating Facility Design Change Process"

The procedures were found to be adequate in scope and level of detail and addressed potential areas of activity that would require preparation of a 10 CFR 50.59 screening and analysis.

Procedure PPM 1.3.43 is the governing procedure which addresses the requirements of 10 CFR 50.59. In general, the procedure was comprehensive, with clear delineation of the responsibilities for individuals required to prepare and review LBIDs. An adequate level of guidance was also included to support the preparation and review of LBIDs.

The discussion in Section 4.8, Licensing Basis Acceptance Limit, is a clear and strong statement noting that any change reducing the margin of safety is an unreviewed safety question (USQ). The guidance section, Attachment 6.2, discusses examples of the licensee's implementation of the 10 CFR 50.59 process to highlight desired actions and includes a number of specific questions for consideration when performing a screening.

In determining whether or not a change to the facility constitutes a USQ as defined by 10 CFR 50.59, Attachment 6.3 of Procedure 1.3.43 provides guidance to the preparer of an SE. Two of the questions that must be answered in making that determination are: (1) May the proposed activity create the possibility of an accident of a different type than any evaluated previously in the FSAR, and (2) may the proposed activity create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the FSAR. The guidance in Section 6.3 for answering these questions states that the possible accidents (or malfunctions of equipment important to safety) of a different type are limited to those that are as likely to happen as those considered in the FSAR. The guidance was found to be inconsistent with the definition of a USQ in 10 CFR 50.59. Specifically, the current regulation does not require a comparison between the probability of analyzed accidents or malfunctions and those created from a proposed activity to determine whether or not a USQ exists. The licensee agreed with the inspector's finding and planned to clarify Attachment 6.3.

Section 4.4 of PPM 1.3.43, Design Safety Function, states, "in the event that existing documentation. . .does not provide an adequate description of the design safety function, the function should be developed and addressed in the 10 CFR 50.59 Safety Evaluation documentation." However, the procedure does not provide specific direction for ensuring that the description of a safety function, as developed in an SE, is translated back to the licensing basis document(s). The lack of a clear tracking mechanism for capturing this information was considered a potential weakness in the licensee's program for maintaining the plant's design basis.

<u>Training and Qualifications</u>

The inspector met with the training staff and discussed the training that is provided to new and current employees to address the LBID process. The licensee has a series of five training modules provided to new employees that cover the follolwing: FSAR; TS; industry regulation, codes and standards; analyses in the FSAR; and LBIDs. The revision dates of the training modules range from February 10, 1997, through February 17, 1998. The inspector reviewed a significant portion of the lesson plans and found the subject matter outlined to be a valuable cross-section of information that both a preparer and reviewer of SEs would need to adequately process an evaluation. As of March 1998, all new employees receive a 1-week course, which includes these five modules. Additionally, personnel assigned as preparer or reviewer of LBIDs are also required to take this course. The licensee currently maintains an active list of approximately 225 preparers and 160 reviewers.

Qualified preparers and reviewers are required to annually take a 3-hour refresher course on preparing LBIDs. POC and Corporate Nuclear Safety Review Board (CNSRB) findings about the quality of SEs, including copies of what were deemed to be outstanding SEs, were provided directly to the training staff for use in discussions during the refresher sessions. It was also noted that training staff routinely screen quality department audit and surveillance reports for items to be included in future training modules. The Training Advisory Group provided another means to update and augment the training program for LBIDs. A review of the Training Advisory Group meeting minutes reflected a good level of sensitivity to the need to make changes to training to respond to lessons learned at the plant and to incorporate the latest regulatory guidance.

10 CFR 50.59 SE

The inspector reviewed a random sampling of 12 SEs that were documented in accordance with 10 CFR 50.59. Most of the evaluations were selected from the licensee's 1997 Annual Operating Report, dated February 26, 1998. For selected SEs, the inspector discussed the package with either the preparer or reviewer. Those SEs reviewed were: SE 98-070, SE 98-069, SE 97-019, SE 97-136, SE 96-068, SE 98-048, SE 97-074, SE 96-052, SE 97-017, SE 97-063, SE 96-086, and SE 98-071.

Most of the SEs were found to be adequately prepared in accordance with the licensee's procedure and consistent with the requirements of 10 CFR 50.59. SE 98-048, SE 97-017, and SE 97-063 were found to be very well prepared with an exceptional level of detail and support for the technical and process conclusion. However, SE 97-019 was found lacking in the level of detail in several of the question responses and was not clearly written. Although the technical basis for the conclusion was valid for the six changes being described, the 10 CFR 50.59 package was not adequately written for each of the changes. The licensee also identified this fact through its corrective action program. SE 96-068 was found lacking in the level of detail in its supporting documentation.

Licensee Oversight of the LBID Process

The inspector noted that both the POC and the CNSRB provide oversight to the licensee's LBID process. The CNSRB subcommittee on 10 CFR 50.59 routinely reviews a large number of processed SEs and screenings and provides feedback to training and the responsible organizations. The subcommittee has current membership with strong industry experience. Also, participation by line management at the subcommittee meetings ensures that prompt feedback is obtained. An additional benefit from this subcommittee is that the POC process for review of SEs is also independently evaluated.

The quality department staff had conducted 15 audits or surveillances since 1996 that included a review of specific screenings or safety analyses for the 10 CFR 50.59 process. Most of the sampled packages were found to be adequate by the licensee's auditors and reviewers; however, a few were identified as reflecting weaknesses in procedures or in implementation thereof. Issued reports included a recommendation to revise Procedure EI 2.8 (Audits 296-017,298-024), a finding of a missed LBID screening for a valve out of service for 7 years (Audit 298-022), and a recommendation to revise the 10 CFR 50.59 screening process to reflect current regulatory guidance and industry practices (SR 296-038). Two surveillances of note were 296-038 (Adequacy of 10 CFR 50.59 Review Question 2) and 296-087 (10 CFR 50.59 Screening). The inspectors observed that the quality department has no plans to periodically review the effectiveness of the training program for qualified preparers and reviewers of 10 CFR 50.59 evaluations. There is also no plan to conduct a comprehensive audit of a large sample of 10 CFR 50.59 SEs on a regular basis. Instead, during regular audits of the engineering program, a number of modifications are routinely reviewed along with the 10 CFR 50.59 screenings and SEs. Feedback from audits and surveillances is provided to appropriate management and reports are distributed to the corporate vice presidents and appropriate supervisors.

Corrective Actions Addressing the 10 CFR 50.59 Process

As a result of violations of 10 CFR 50.59 related to instrument response time testing and downgrade of the reactor core isolation cooling (RCIC) system, the licensee initiated Problem Evaluation Requests 297-0982 and 298-0123 to address how licensee personnel were using generic (industry and NRC) guidance in 10 CFR 50.59 evaluations. As a corrective action, the licensee reviewed about 900 SE summaries included in Supply System's annual reports and about 100 SEs in detail to determine if the preparers were adequately implementing generic guidance. It was found that only three SEs had not been appropriately developed in accordance with applicable guidance and two of these were referred back to line management for action. One of these resulted in Problem Evaluation Request 298-0156, which addresses the quality of SE 97-019. The inspector found that this effort was comprehensive and worthwhile in its resulting calibration of one of the quality indicators for SEs.

Special Initiative

In response to an item identified by the most recent Performance Self-Assessment, the licensee has developed a Performance Enhancement Program (PEP) to improve the quality of 10 CFR 50.59 SEs. Specifically, two members of the POC assigned to Engineering have developed a program to independently review and "grade" the quality of each SE prior to its POC review and provide prompt feedback, guidance, and coaching to the preparer/reviewer. Screening criteria and grading guidance have been developed and the quality of a representative sample of recent SEs has been benchmarked using the criteria. The PEP is expected to be initiated during July 1998 and the plan includes a program to review all SEs prepared, not just those from Engineering. The new process will not only provide negative feedback, if appropriate, but also will reward those who consistently prepare high quality SE products. The inspector found this PEP to be a worthwhile process enhancement, which has the potential to improve the quality of SEs in all organizations in the plant.

POC Meeting

On June 30, the inspector attended a POC meeting to review two items: (1) a test of the firewater system, and (2) a SE for a followup assessment of operability concerning the running of two fire water pumps for a period of up to 4 months. The meeting was conducted according to procedure, with the chairman immediately establishing that the required quorum was met. Ground rules for the review were discussed and it was emphasized that any package found to be unacceptable would be rejected. For each item discussed, the chairman permitted the preparer to present a discussion of the safety significance of the package. The inspector observed participation by most members and found the questions to be probing and focused. For the first item discussed, comments were frank and led to the prompt conclusion, as stated by the chairman, that the package was unacceptable as written. More information was needed as to why the Supply System was technically confident that a waterhammer and subsequent firewater system damage would not result from conducting the test. The presenter was told to return with a revised package for discussion at the next POC meeting. The second item was discussed in the same manner and after questions and discussion, a unanimous vote resulted in approval of the package. Overall, the inspector found the meeting to be well organized, efficient, and very effective in its focus.

c. Conclusions

The licensee has maintained an appropriate program to address the requirements of 10 CFR 50.59. Program implementing procedures were generally of sufficient detail to ensure that proposed activities would precipitate SEs. However, two areas were noted where procedure guidance was either weak or inconsistent with requirements. Although the quality of the 12 SEs reviewed was not always consistent, overall the quality was good. Strengths were noted in the training and oversight programs with regards to maintaining a sufficient pool of qualified SE preparers and providing timely, critical feedback on their products.

E1.2 Equipment Drain Modification

a. Inspection Scope (37551)

The inspectors reviewed FSAR Figure 9.3-5, "Equipment Drain System Reactor Building." and Figure 9.3-8, "Floor Drains - Reactor Building." The inspectors also reviewed Drawing M723, "Embedded Floor & Equipment Drains - Reactor Building Elevation 422'-3[inches]," Revision 5. In addition, the inspectors reviewed FSAR Sections 3.4.1.4.1.2, "Internal Flood Protection Requirements," and 9.3.3.2.2.1, "Reactor Building Floor Drains." The inspectors walked down emergency core cooling pump rooms and associated drain systems.

b. Observations and Findings

FSAR Section 3.4.1.4.1.2, "Internal Flood Protection Requirements" states: "In the event of a pipe break of sufficient size to flood sump pumps in one room, common mode flooding between watertight rooms is prevented by the following:

a. Of the safety-related watertight pump rooms at 422'-3 level of the Reactor Building the equipment drain sump serves only the RCIC pump room. Drains to RHR A and B pump rooms are capped."

FSAR Figure 9.3-5, "Equipment Drain System Reactor Building." also shows that the equipment drain lines for RHR pump Rooms A and B are capped. During walkdowns of RHR pump Rooms A and B, the inspectors found that a cap was not installed on the equipment drain line for RHR pump Room B. Subsequently, the licensee performed a leak test of RHR pump Room B equipment drain per Work Order MGF3. The licensee determined that the drain flow path to the R-5 sump, located in the control rod drive pump room, was at least partially opened and capped the line.

The safety significance of the open equipment drain line in RHR pump Room B was evaluated based upon potential flooding in the reactor building. A review of mechanical Drawing M723 showed that RHR pump Room B would communicate directly with RCIC pump room and the control rod drive hydraulic pump room. A single failure of the floor drain isolation valve between the RCIC pump room and RHR pump Room A or the control rod drive hydraulic pump room and the high pressure core spray pump room would also allow flooding to migrate to these areas. Thus, the potential existed for multiple trains of equipment to be affected. However, it was recognized that, due to the size of the various pump rooms, a substantial flooding rate would be required to adversely impact all of the areas. It is likely that operator action would be taken prior to a flooding event affecting multiple trains of emergency core cooling or shutdown cooling.

From a review of plant records, the licensee was unable to determine when or if a cap was ever installed on the equipment drain line from RHR pump Room B. As a result, the root cause of the violation was indeterminate. The system engineer walked down other portions of the equipment and floor drain systems and did not identify any additional discrepancies with system configuration. A written SE could not be found to support that deviation from the facility's description in the FSAR. The inspectors also noted that the discrepancy was not identified through the licensee's ongoing effort to review the accuracy of the FSAR.

10 CFR 50.59 states that a licensee may make changes in the facility as described in the safety analysis report without Commission approval when the proposed changes do not involve a USQ. This section further states that the licensee shall maintain records of changes in the facility to the extent that these changes constitute changes in the facility as described in the safety analysis report. The failure to maintain the equipment drain line to RHR pump Room B capped represented a defacto change to the facility that was not evaluated in accordance with 10 CFR 50.59(b)(1) (VIO 50-397/98013-02). Lacking

a specific root cause for the violation, the licensee's corrective actions to install a cap and verify the drains for other equipment and rooms were found to be reasonable to address the immediate concern of plant configuration.

c. Conclusion

The configuration of the reactor building equipment drains did not conform to the description in the FSAR in that a cap was not installed on the drain line from RHR pump Room B. The cap was required as part of the licensee's physical controls to protect against common mode flooding. A violation of 10 CFR 50.59 was identified for failure to document a written SE for this defacto change to the facility. The licensee's corrective actions to install a cap on the drain line and review the generic implications for other portions of the drain systems, were found to be appropriate.

## IV. Management Meetings

### X1 Plant Restart Meeting

On July 2, licensee representatives met with Region IV management in Arlington, Texas, to discuss their recovery efforts from the June 17 flooding event and the plant's readiness for return to power operation. A brief summary of that meeting is provided in NRC Inspection Report 50-397/98-16. The following is a list of attendees for the meeting:

| NRC | Licensee |
|---|---|
| E. Merschoff, Region IV | V. Parrish |
| W. Bateman, NRR | P. Bemis |
| P. Gwynn, Region IV | R. Webring |
| D. Chamberlain, Region IV | S. Oxenford |
| T. Marsh, NRR | D. Atkinson |
| J. Pellet, Region IV | J. Kane |
| J. Shackelford, Region IV | D. Coleman |
| T. McKernon, Region IV | S. Wood |
| L. Whitney, NRR | W. Harper |
| P. Qualls, NRR | A. Arastu, Bechtel Corporation |
| C. Petrone, NRR | |

Presentation material provided by the licensee is included as Attachment 2 to this report.

### X2 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on July 30, 1998. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

# ATTACHMENT 1

## Supplemental Information

## PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Atkinson, Manager of Quality [recovery manager for flooding event]
D. Coleman, Regulatory Affairs Manager
F. Diya, System Engineering Manager
D. Feldman, Assistant Operations Manager
D. Giroux, System Engineering
D. Hillyer, Radiation Protection Manager
D. Kobus, Fire Protection Supervisor
P. Inserra, Licensing Manager
S. Oxenford, Operations Manager
G. Sanford, Maintenance Manager
G. Smith, Plant General Manager
S. Wood, System Engineering Supervisor

NRC

T. Marsh, Office of Nuclear Reactor Regulation

## INSPECTION PROCEDURES USED

| | |
|---|---|
| IP 37001: | 10 CFR 50.59 Evaluation Program |
| IP 37551: | Onsite Engineering |
| IP 61726: | Surveillance Observations |
| IP 62707: | Maintenance Observations |
| IP 71707: | Plant Operations |
| IP 71750: | Plant Support |
| IP 92901: | Followup - Operations |
| IP 92902: | Followup - Maintenance |

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 50-397/98013-01 | VIO | failure to maintain the minimum flow bypass valve in the closed position |
| 50-397/98013-02 | VIO | failure to document a written safety evaluation for a defacto change to the facility |

Closed

| | | |
|---|---|---|
| 50-397/98-010-00 | LER | TS-required shutdown due to inoperable TIP probe system isolation valve |
| 50-397/98013-02 | VIO | failure to document a written safety evaluation for a defacto change to the facility |

LIST OF ACRONYMS USED

| | |
|---|---|
| ALARA | as low as reasonably achievable |
| CNSRB | Corporate Nuclear Safety Review Board |
| ESF | engineered safety feature |
| FSAR | Final Safety Analysis Report |
| LBID | licensing basis impact determination |
| LER | licensee event report |
| LPCS | low pressure core spray |
| NRC | U.S. Nuclear Regulatory Commission |
| PEP | performance enhancement program |
| POC | Plant Operations Committee |
| PPM | plant procedure manual |
| RCIC | reactor core isolation cooling |
| RHR | residual heat removal |
| SE | safety evaluation |
| TIP | traversing incore probe |
| TS | Technical Specifications |
| USQ | unreviewed safety question |
| VIO | violation |
| WNP-2 | Washington Nuclear Project-2 |

ATTACHMENT

# WNP-2

# FIRE MAIN RUPTURE
# AND
# PUMP ROOM FLOODING
# RECOVERY

# AGENDA

- Introduction
- Event Description
- Root Cause
- Fire Protection
- Plant Flooding
- Plant Repair Status
- Plans

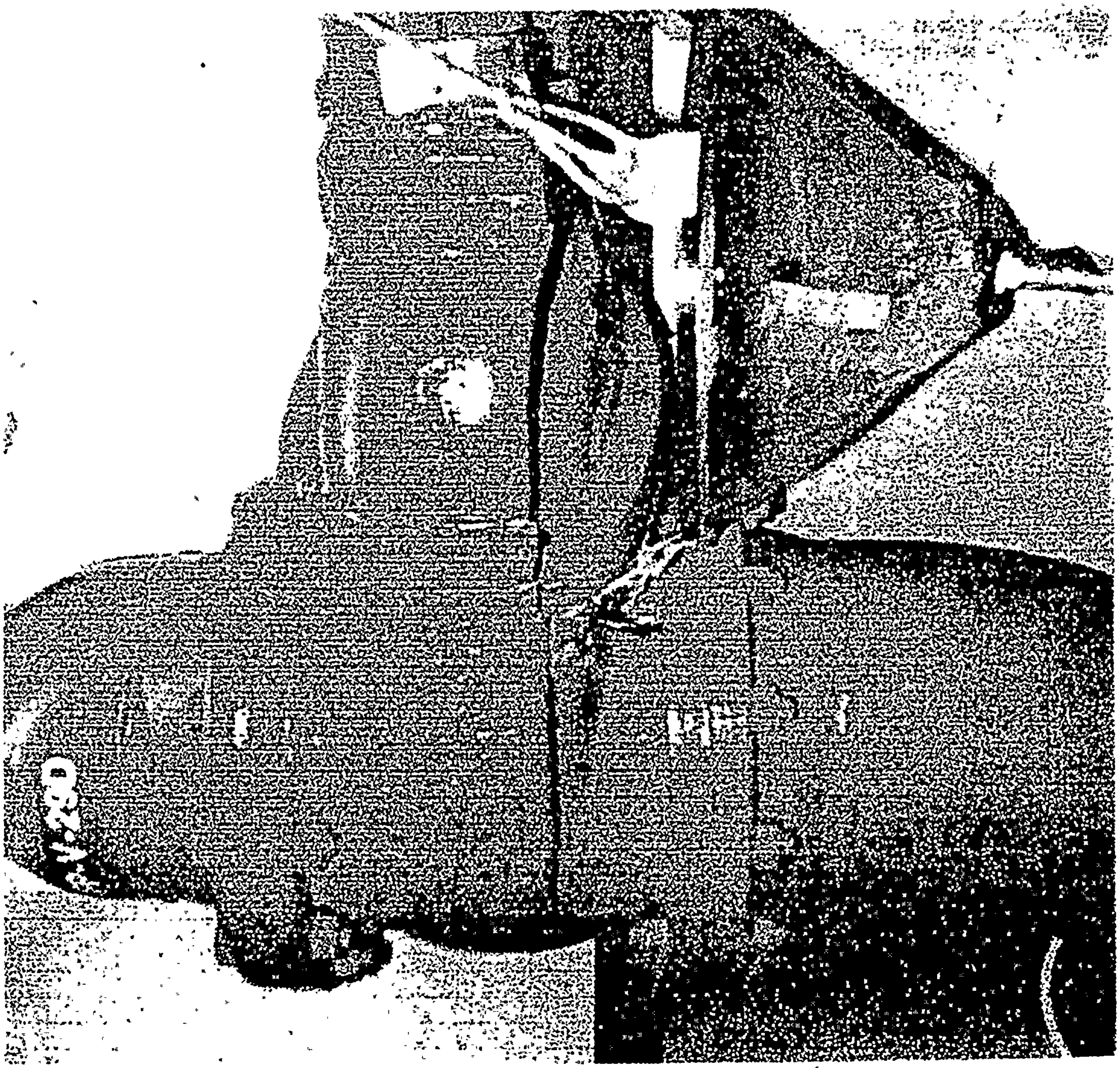
# WNP-2 FLOODING EVENT

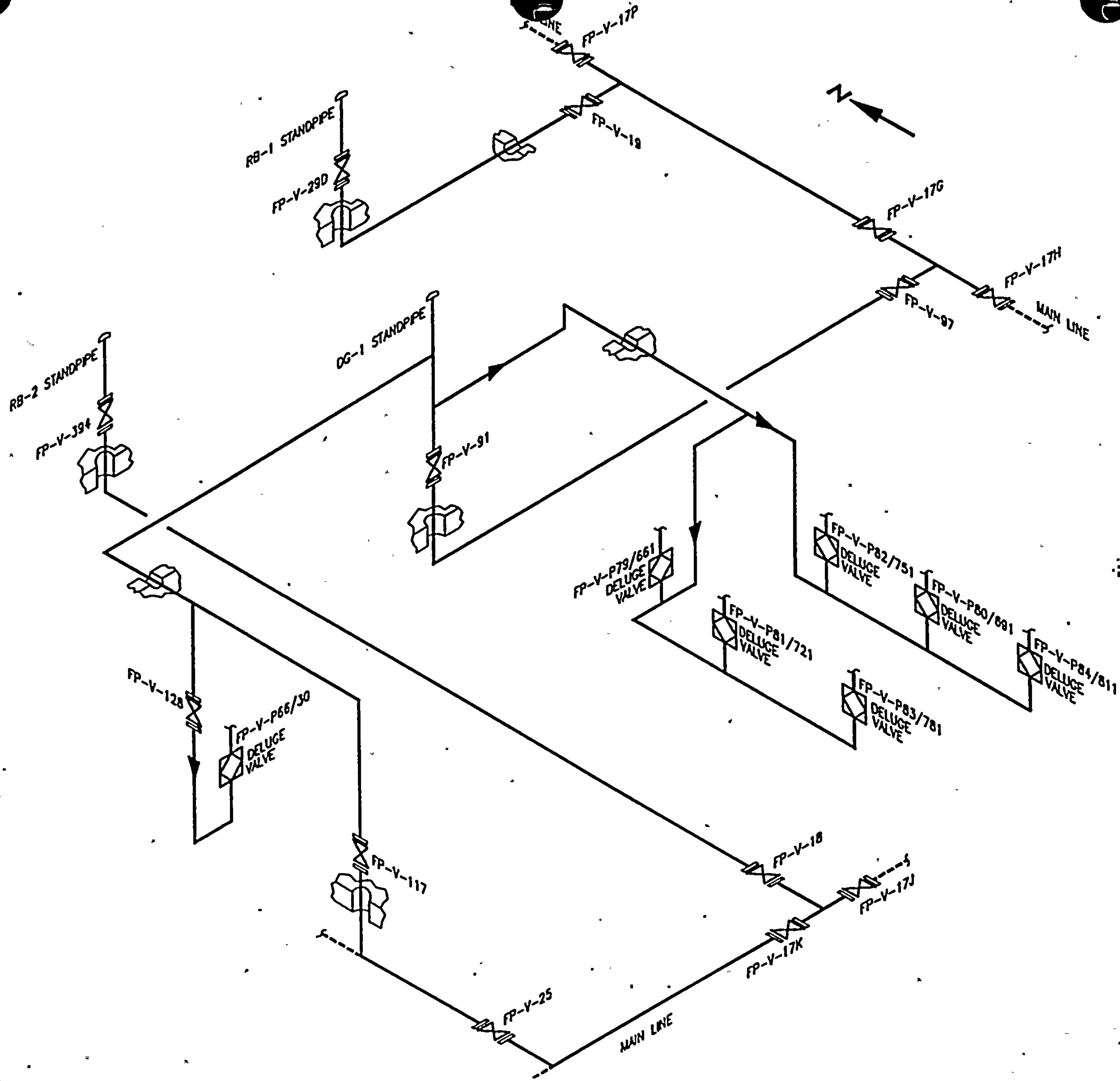
- Cutting torch operation activates smoke detector outside DG-2 room.

- Fire zone 66 pre-action valve opens resulting in a lowering of the water level in the Reactor Building Northeast stairwell fire main and start of three of four main fire pumps.

# FLOODING EVENT (cont)

- As the fire main pressure is restored, a water hammer is created that ruptures a 12" isolation valve near the bottom of the reactor building Northeast stairwell (and fire zone 81 pre-action valve opens "sympathetically").

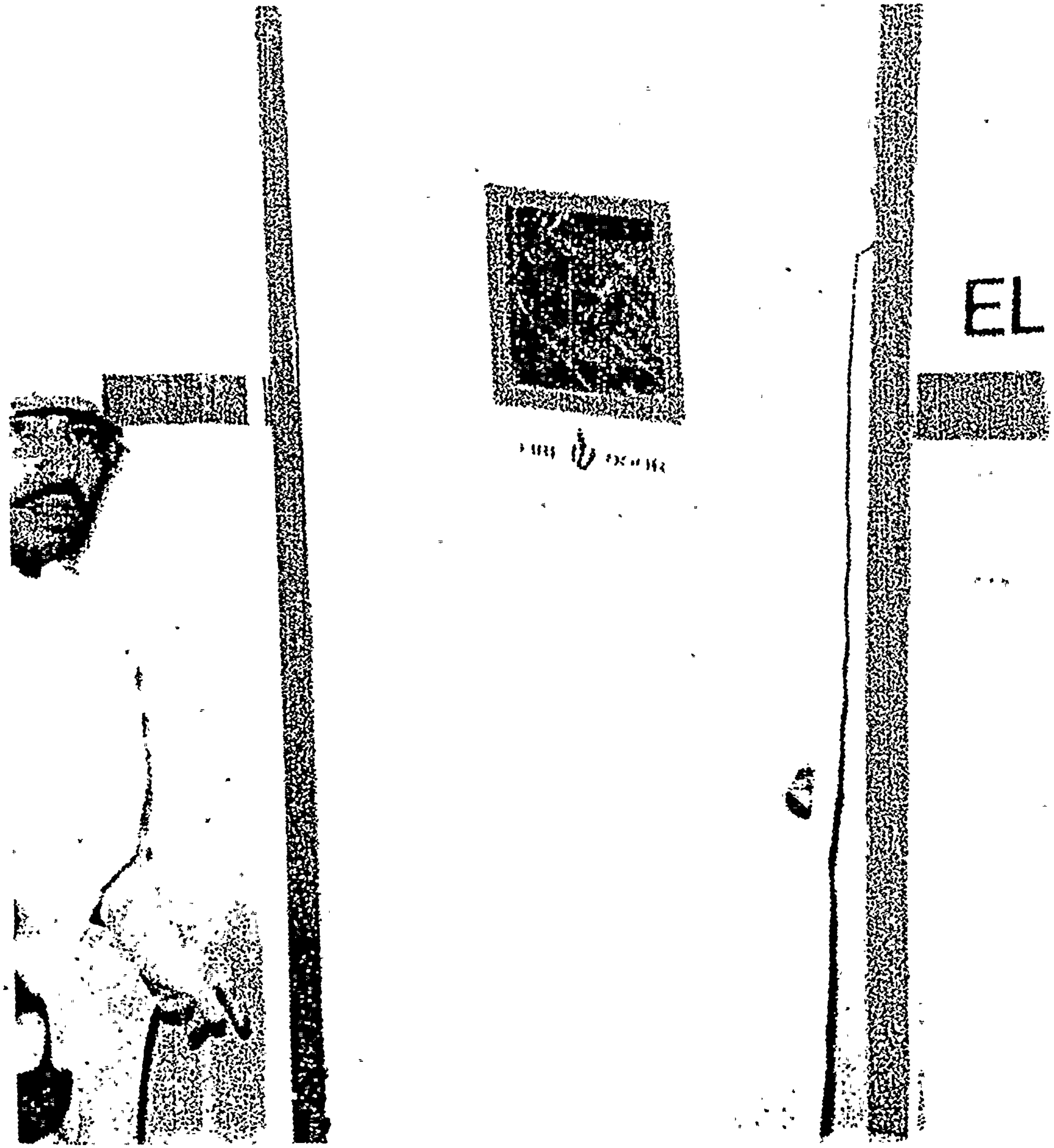


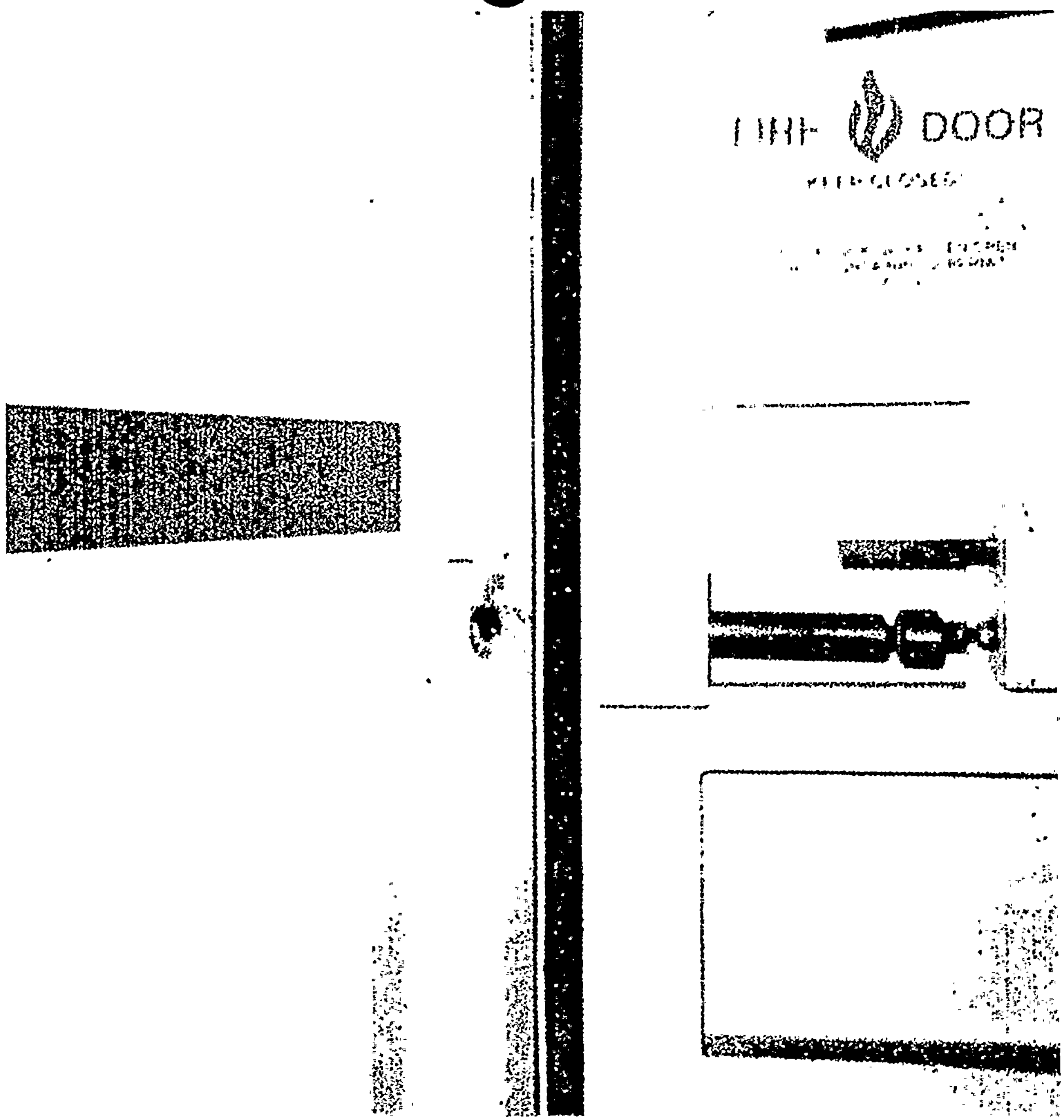


MAIN LINE
FP-V-17P
FP-V-19
RB-1 STANDPIPE
FP-V-29D
FP-V-17G
FP-V-17H
MAIN LINE
FP-V-97
DG-1 STANDPIPE
RB-2 STANDPIPE
FP-V-394
FP-V-91
FP-V-P79/661
DELUGE
VALVE
FP-V-P82/751
DELUGE
VALVE
FP-V-P80/691
DELUGE
VALVE
FP-V-P81/721
DELUGE
VALVE
FP-V-P84/811
DELUGE
VALVE
FP-V-P83/781
DELUGE
VALVE
FP-V-128
FP-V-P66/30
DELUGE
VALVE
FP-V-117
FP-V-18
FP-V-17J
FP-V-17K
FP-V-25
MAIN LINE
N

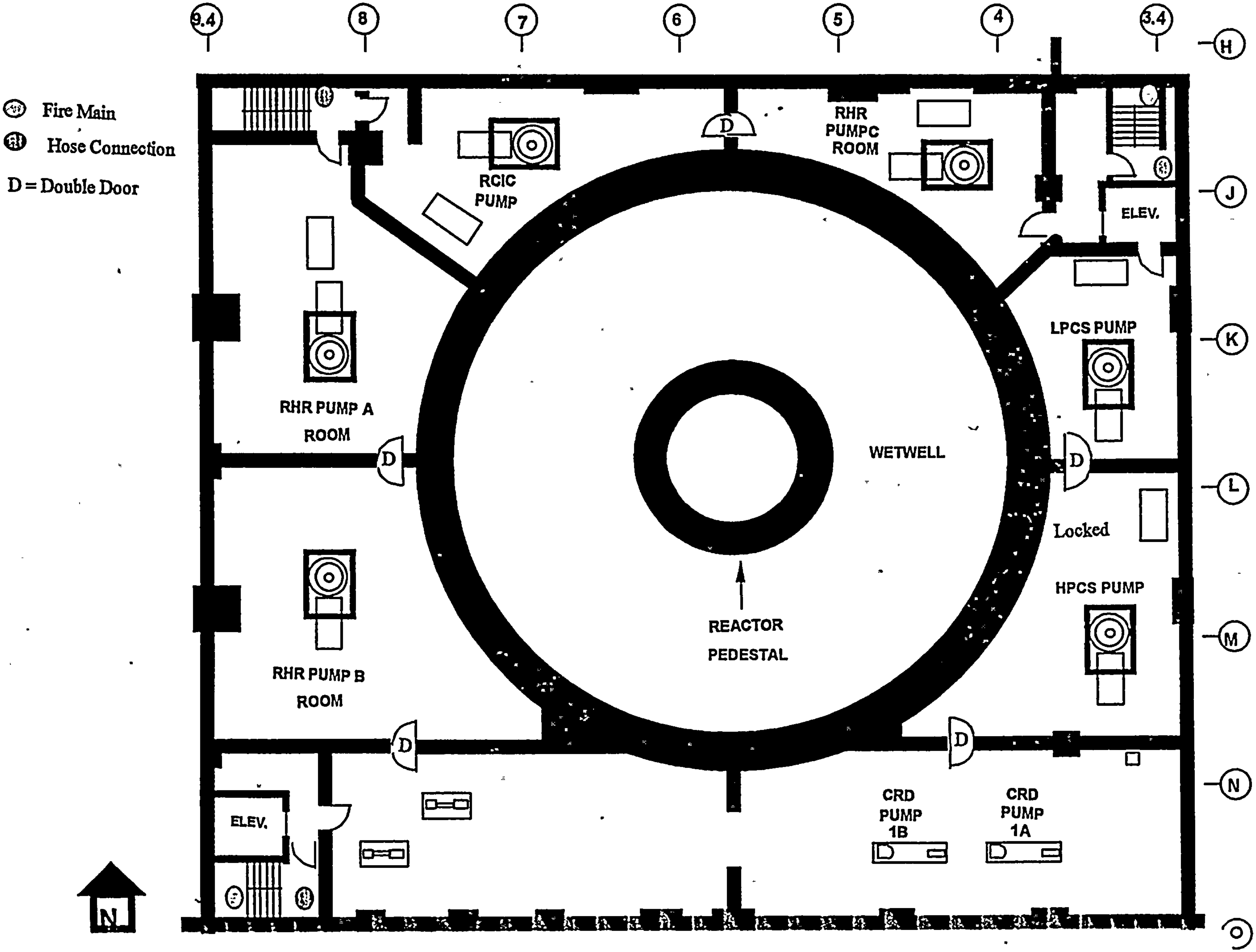
# FLOODING EVENT (cont)

- Stairwell flooding follows with water entering the RHR-C pump room through an undogged door.
- Water flows from the RHR-C pump room to the LPCS pump room through a floor drain valve that fails to close.
- Operators secure the fire pumps and isolate the header after 12 minutes from the start.


EL


DOOR

# REACTOR BLDG. ELEV. 422'



# FLOODING EVENT (cont)

- An UNUSUAL EVENT is declared and the TSC and OSC are staffed.
- Water is pumped from the stairwell to the storm drain after sampling then later discharge is transferred to the condensate sump. (Sample <1.5x10-8)
- Hanford Fire Department is called out for fire suppression support if needed.

# FLOODING EVENT (cont)

- The fire protection system is restored with two main pumps operating after replacing the failed valve.

- The UNUSUAL EVENT is terminated and recovery is commenced.

# ROOT CAUSE ANALYSIS SUMMARY

Terry L. Meade

# RCA SCOPE

- Response of the Fire Protection System
  - Interface with Engineering Teams

- Multiple Preaction system actuation

- Flooding of the ECCS pump rooms and the barriers that should have prevented the multiple room flooding

# RCA METHODOLOGIES

- Comprehensive Timeline
- Event and Causal Factor Chart
- Interviews with station personnel
- Barrier Analysis
  - ECCS Room Flooding
- Fault Tree Analysis
  - Opening of watertight doors
  - FP Clapper Valve failure

# ROOT CAUSE

- INADEQUATE FIRE PROTECTION SYSTEM DESIGN
  - The system is configured such that destructive forces are generated during an anticipated challenge with only the jockey pump running.

# RCA CONTRIBUTING FACTORS

- Material condition of the P81 preaction valve trim allowed a sympathetic unnecessary actuation.
- Human Performance resulted in LTA closure and dogging of the RHR 2C watertight door.
- Material condition of FDR-V-609 was a significant contributor to the flooding of the LPCS room.

# RECOMMENDATIONS cont.

- Strengthen fire impairment and ignition source permit process
- Establish methods to avoid preaction during maintenance activity
- Strengthen knowledge base for potential FP actuations
  - Operations' Supervision
  - Maintenance Crafts & Supervision

# RECOMMENDATIONS cont.

- Strengthen the configuration control of water tight doors.
  - Improve knowledge base and self checking techniques on watertight doors.
  - Enhance PM & periodic testing for watertight doors.
    - Door Alignment
    - Dogging mechanism

# RCA RECOMMENDATIONS

- Enhance FP system design to avoid water hammer
- Provide sufficient online pump volume
- Establish compensatory action on loss of this online volume
- Modify preaction clapper valve trim to assure positive check valve closure
- Enhance PM & periodic testing for clapper valve to detect degradation

# RECOMMENDATIONS cont.

- Strengthen the configuration and operability of the Floor Drain inter-room system.

- Shut FDR-V-607, 608, 609

# FIRE PROTECTION

# FIRE PROTECTION SYSTEM

- Water hammer history:
  - No records found of a simultaneous (sympathetic) actuation of any two pre-action/ deluge valves before this event at WNP-2.
  - The WNP-2 Plant has experienced two previous water hammer events in the fire protection system that have damaged pressure gages.

# FIRE PROTECTION SYSTEM

- Hydraulic Analysis
  - Initial hand calculations indicated that two pumps operating would prevent water hammer in the stairwell fire mains.
  - Follow-up time dependent analysis indicated that the system was more complicated than first assumed and that a void in the stairwell fire main(s) would occur with one or two zones activating.

# FIRE PROTECTION SYSTEM

- Hydraulic Analysis (cont)
  - Since modeling indicated that the normal activation of a pre-action/ deluge valve could cause water hammer, a nitrogen bubble (1-2 ft in a 6” line) was added as a “cushion”.
  - Additionally, cast iron isolation valves in the stairwells (2) were replaced with cast steel isolation valves.

# FIRE PROTECTION SYSTEM

- NFPA Compliance
  - Both the continuous operation of two fire pumps and the addition of nitrogen at the top of the stairwell fire mains was reviewed with respect to the fire code, determined to be in compliance, and was evaluated under 10 CFR 50.59.
  - Discussions with NFPA staff members supported these conclusions.

# FIRE PROTECTION SYSTEM

- Compensatory Actions:
  - The nitrogen bubbles are monitored (ultrasonically) to maintain between 1 and 2 feet of cushion. Nitrogen is manually added as necessary to maintain this volume.
  - The main fire pumps will be evaluated for degradation on an accelerated schedule until the long term solution is implemented.

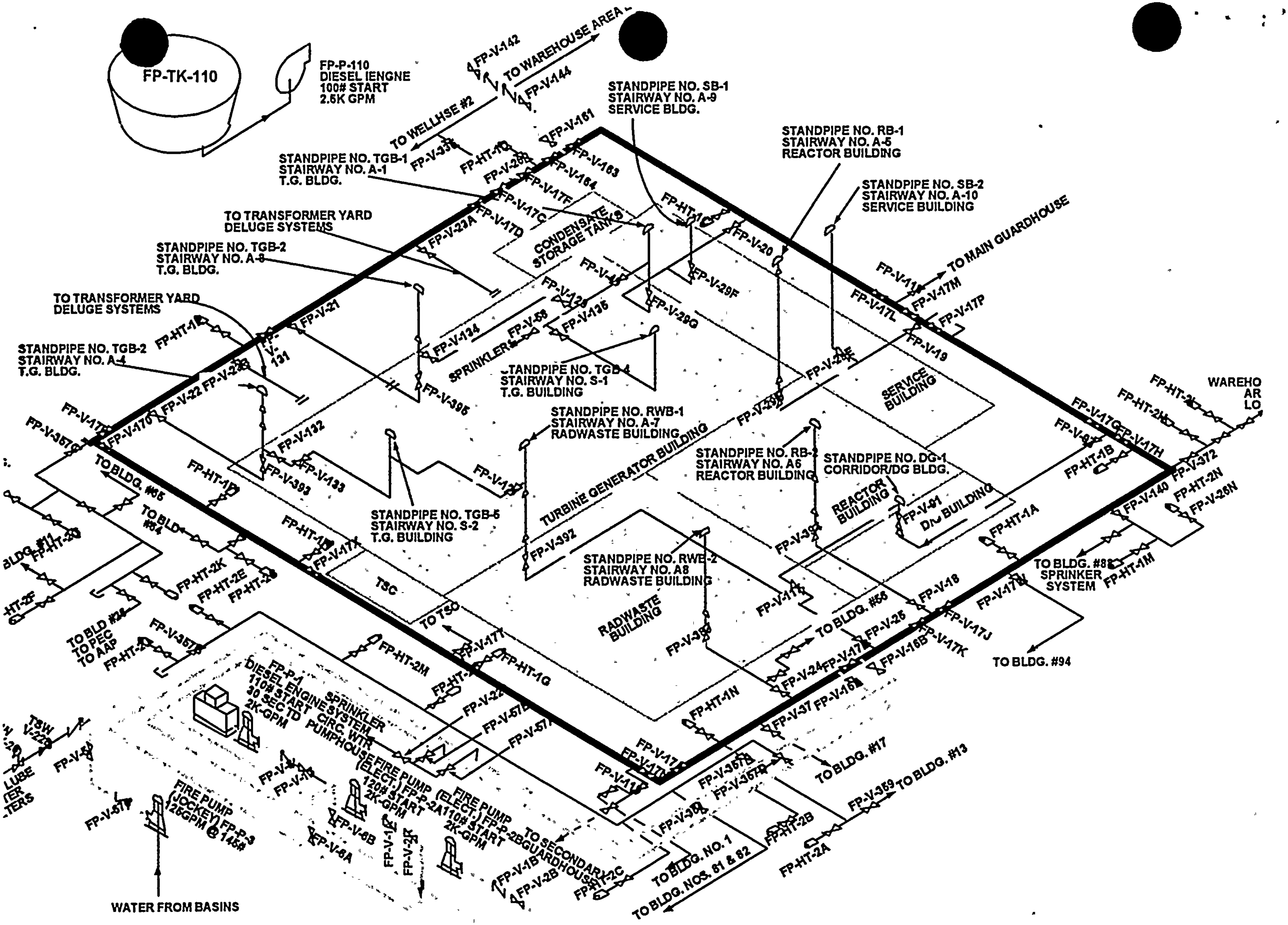
# FIRE PROTECTION SYSTEM

- System Test:
  - The most limiting pressure transient induced by pre-action/ deluge valve actuation is initiated and the strain is measured at various points in the system. (Done with two fire pumps running.)
  - Acceptance is determined by inspection and evaluation of test data.

# FIRE PROTECTION SYSTEM

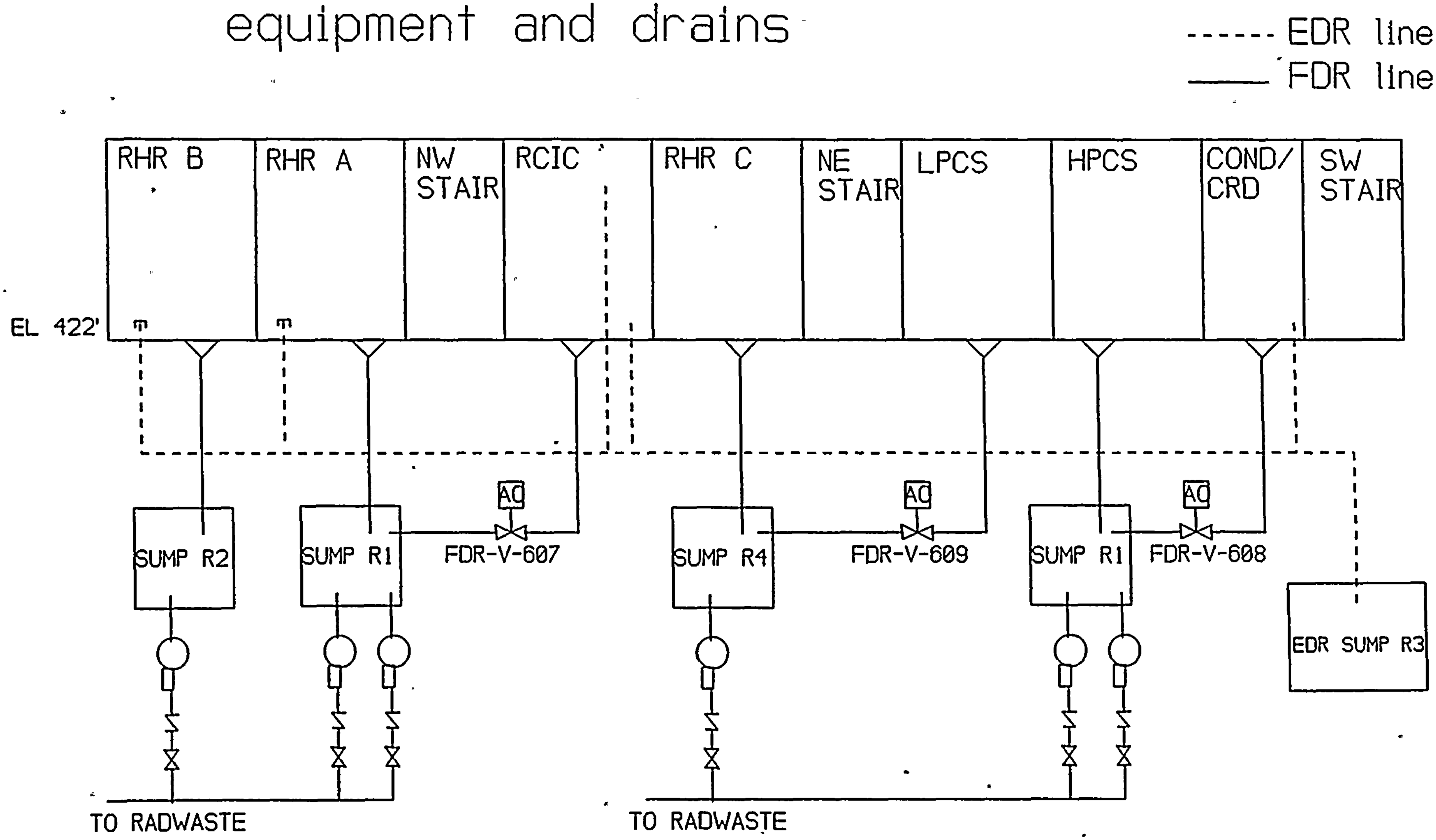
- Long Term Goals:
  - Determine and implement a solution to the water hammer problem that allows:
    - Main fire pumps in standby
    - Removal of nitrogen bubbles

# PLANT FLOODING



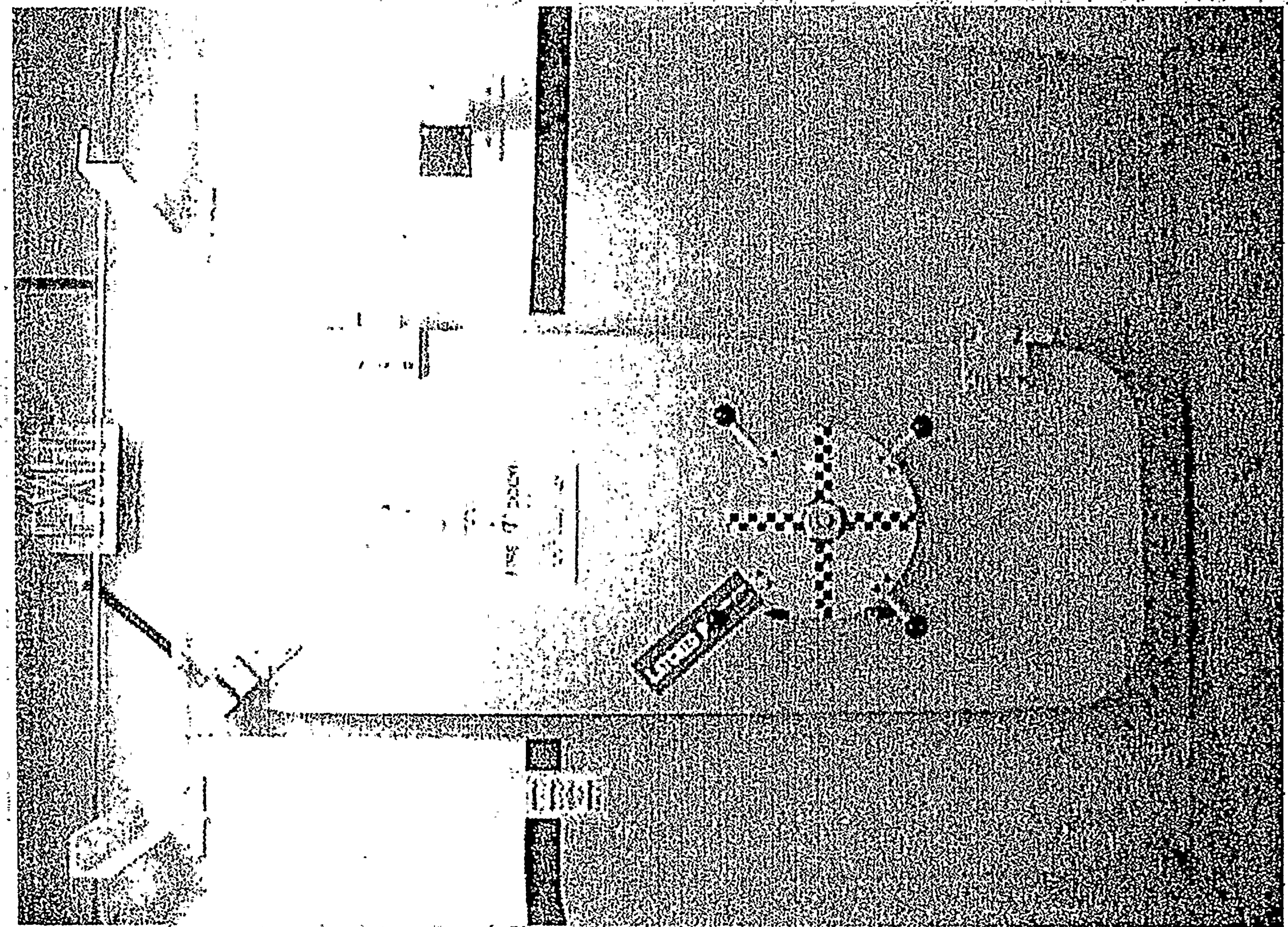
FP-TK-110
FP-P-110
DIESEL IENGNE
100# START
2.5K GPM
FP-V-142
TO WAREHOUSE AREA
FP-V-144
TO WELLHSE #2
STANDPIPE NO. SB-1
STAIRWAY NO. A-9
SERVICE BLDG.
STANDPIPE NO. RB-1
STAIRWAY NO. A-5
REACTOR BUILDING
STANDPIPE NO. TGB-1
STAIRWAY NO. A-1
T.G. BLDG.
STANDPIPE NO. SB-2
STAIRWAY NO. A-10
SERVICE BUILDING
TO TRANSFORMER YARD
DELUGE SYSTEMS
CONDENSATE STORAGE TANKS
TO MAIN GUARDHOUSE
STANDPIPE NO. TGB-2
STAIRWAY NO. A-8
T.G. BLDG.
TO TRANSFORMER YARD
DELUGE SYSTEMS
STANDPIPE NO. TGB-2
STAIRWAY NO. A-4
T.G. BLDG.
SPRINKLERS
STANDPIPE NO. TGB-4
STAIRWAY NO. S-1
T.G. BUILDING
SERVICE BUILDING
STANDPIPE NO. RWB-1
STAIRWAY NO. A-7
RADWASTE BUILDING
TURBINE GENERATOR BUILDING
STANDPIPE NO. RB-2
STAIRWAY NO. A6
REACTOR BUILDING
STANDPIPE NO. DG-1
CORRIDOR/DG BLDG.
REACTOR BUILDING
DG BUILDING
TO BLDG. #65
TO BLDG. #64
STANDPIPE NO. TGB-5
STAIRWAY NO. S-2
T.G. BUILDING
STANDPIPE NO. RWB-2
STAIRWAY NO. A8
RADWASTE BUILDING
TO BLDG. #88
SPRINKER SYSTEM
TSC
TO TSC
RADWASTE BUILDING
TO BLDG. #56
TO BLDG. #94
TO BLD #36
TO PEC
TO AAP
FP-P-1
DIESEL ENGINE
110# START
30 SEC TD
2K-GPM
SPRINKLER SYSTEM
CIRC. WTR
PUMPHOUSE
FIRE PUMP (ELECT.) FP-P-2A 110# START 2K-GPM
FIRE PUMP (ELECT.) FP-P-2B 110# START 2K-GPM
FIRE PUMP (JOCKEY) FP-P-3 125GPM @ 145#
TO SECONDARY GUARDHOUSE
TO BLDG. #17
TO BLDG. #13
TO BLDG. NO. 1
TO BLDG. NOS. 81 & 82
WATER FROM BASINS

# Relationship between pump room equipment and drains



# Plant Flooding

- Comparison of Flooding Event to Licensing & Design Basis
- Safety Basis for Sump Isolation Valves & Their Configuration
  - Close FDR-V-607, 608, & 609 Until Analyzed or Modified




LATCHED POSITION

# Plant Flooding (cont)

- Watertight Door Performance with Respect to Analysis
  - Sensitivity Analysis of Leak Rate
  - Analysis for 70 GPM Leak Rate
  - Maintenance of Door Seals

# Plant Repair Status

- Recovered RHR-2C & LPCS Pump Rooms
  - Established Acceptance Criteria
  - Assessed Damage
  - Repaired, Replaced, Drained & Dried, As Necessary
  - Developed & Implemented Specific PMT for Each Component & System

# Plant Repair Status (cont)

- Replaced LPCS Keep Filled Pump (P2) Motor
- Dried, Cleaned, Tested RHR-2C Keep Filled Pump (P3) Motor
- Dried, Cleaned, Tested RHR-2C Pump Motor (Wyle Labs)

# Plant Repair Status (cont)

- Repaired FDR -V609 (Surveilled V607 & V608)
  - Replaced solenoid pilot valves on FDR-V-607/8/9
  - Established Quarterly Surveillance Requirements
  - Established 10 yr Solenoid Pilot Valve Replacement Interval
- Inspected, Repaired, Tested Seals on All 15 Watertight Compartment Doors

# Plant Repair Status (cont)

- Prepared Engineering Restart Justification Report
- Prepared Engineering Certification for Restart for Each Room Flooded
- Utilized Bechtel Equipment Experts to Inspect Rooms, Review Acceptance Criteria, & Validate Supply System Results
- Currently Ready for Restart

# RESTART PLAN

- Fire protection is operable:
  - Analysis, confirmed by test results, will demonstrate no water hammer
  - two main fire pumps operating
  - two cast iron valves replaced with cast steel valves
  - pre-action/deluge valves check valves replaced with improved design
  - a nitrogen "cushion" in the vertical water tower areas.

# RESTART PLAN (cont)

- Repairs complete to support restart.
- Fire protection test and evaluation.
- Improved fire impairment/ignition source control procedure.

# RESTART PLAN (cont)

- Compensatory actions developed:
  - FDR-V-607/608/609 closed w/admin controls
  - Watertight doors inspected/repaired as necessary and human factors improvements completed.
  - Training implemented on proper closure of watertight doors.
  - Procedures in place for two fire pump operation with nitrogen cushion.

# RESTART PLAN (cont)

- WNP-2 ready for startup.
- Initiate long term corrective action efforts.