ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Attachments:
1. AIT Charter
2. Detailed Sequence of Events
3. Simplified Diagram of Flood Affected Areas
4. List of Documents Reviewed
5. List of Persons Contacted

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EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/98-16

On June 17, 1998, a water hammer in the fire protection system caused the rupture of a fire main valve in the northeast stairwell of the reactor building at the WNP-2 facility. The resulting flood water entered into the Residual Heat Removal C pump room through a watertight door, which had apparently been left in an unsecured condition and propagated to the adjacent low pressure core spray pump room via a sump isolation valve, which failed to close as designed. The flood completely submerged the residual heat removal pump and motor and the associated keepfill pump (which also serves the Residual Heat Removal B train). The level in the low pressure core spray pump room rose to just below the pump motor and completely submerged the low pressure core spray keepfill pump (which also serves the Residual Heat Removal A train).

To isolate the flooding, plant operators secured the fire pumps, which impaired the normal fire suppression capability of the station. On the basis of these events, the plant operators declared a Notification of Unusual Event and activated the plant emergency response organization. As a compensatory measure for the loss of the normal fire suppression capability, the nearby Hanford fire department dispatched emergency equipment to the site. Subsequently, the plant staff placed the normal fire suppression system in an alternate configuration, which was not susceptible to water hammer and terminated the unusual event.

Although the event did not pose a risk to the public health and safety, the team concluded that the event was safety significant. The rupture of the fire protection system and the subsequent flooding and loss-of-equipment function in two safety-related pump rooms powered from different electrical divisions suggests that deficiencies may exist in system design, operation, maintenance, and personnel practices. Specifically, the event appears to have been preventable in that the actuation of the fire main preaction system, which contributed to the water hammer, was caused by a maintenance activity and could have been avoided. Evidence also exists that the fire main system may have been known to be susceptible to water hammer events, and corrective actions have not precluded the recurrence of this phenomenon. Further, the extent of the flooding caused by the fire main rupture was significantly increased by a watertight door left in an unsecured condition and was further complicated by the failure of a sump isolation valve, which was known to be in a degraded condition prior to the event. Additionally, based on the failures, which occurred, and the fact that this event had a significant impact on safety-related equipment powered from two separate divisions, this event calls into question the adequacy of the design of the fire protection system and plant features to ensure protection against flooding.

In response to the flooding, Region IV dispatched an augmented inspection team to investigate the circumstances surrounding the event. The findings of the augmented inspection team are documented in the attached inspection report. A summary of the major issues identified by the augmented inspection team is summarized in Section 3 of the report. Several of these findings are indicative of system design, maintenance, procedural, and personnel performance deficiencies, which may have caused or contributed to the event. In accordance with the NRC's

policies regarding the conduct of augmented inspections, the findings of the augmented inspection team have not been characterized with respect to regulatory compliance. Future NRC inspections will be conducted to determine whether any violations of NRC requirements may have existed.

Report Details

1 Introduction

1.1 Purpose and Scope of the Inspection

The NRC has established a policy for the timely, thorough, systematic inspection and evaluation of significant operational events at commercial nuclear power plants. One aspect of this policy is the use of an augmented inspection team to determine the causes, conditions, and circumstances that may be relevant to a particular event and to communicate the important findings associated with the event to NRC management. In accordance with NRC Management Directive 8.3, "NRC Incident Investigation Procedures," the Region IV Regional Administrator chartered an augmented inspection team to be conducted at the WNP-2 nuclear facility to review the circumstances surrounding a rupture in the fire protection system and subsequent flooding of two safety-related equipment rooms, which occurred on June 17, 1998.

The augmented inspection team was composed of five NRC personnel including a team leader and specialists in operations, fire protection, engineering, and electrical equipment. The augmented inspection team charter (Attachment 1) directed the team to conduct fact finding to determine the following:

- The initial conditions, which existed prior to the event.
- The sequence of events, which transpired during the event.
- The performance of plant equipment during the event.
- The actions taken by the licensee's emergency response organization.
- The actions taken by the plant operators during the event.
- Evaluation of certain issues related to containment integrity, radioactive effluents, and the licensee's proposed plan for root-cause assessment and corrective actions.

The augmented inspection team charter directed the team to use the results of the licensee's own investigations to the maximum extent practicable. Thus, much of the information provided by the licensee was not independently developed. However, the team conducted verification activities to confirm the results of the licensee's investigation in addition to performing a limited amount of independent inspection.

The augmented inspection team developed a detailed sequence of events based on a review of the licensee's documentation, personnel interviews, and reviews of plant equipment traces. This detailed sequence of events is provided in Attachment 2 of this

report. A simplified diagram of the area affected by the flooding is provided in Attachment 3. Attachment 4 provides a summary of the major documents, which were reviewed by the team. Attachment 5 provides a list of the principal licensee personnel who were contacted and a brief summary of inspection findings which require additional inspection followup.

2 Event Description

2.1 Detailed Sequence of Events

As previously indicated, a detailed sequence of events is provided in Attachment 2. A brief summary of the major events is provided here for general reference and to provide a context for the remaining sections of the report.

On June 17, 1998, WNP-2 was in Mode 4 (cold shutdown). Preparations were underway for a plant startup. At approximately 1:45 p.m. (PDT), multiple fire alarms were received in the control room coincident with three main fire pumps automatically starting and several loud water hammer noises being heard throughout the plant. The apparent water hammer caused Fire Protection Isolation Valve FP-V-29D to rupture in the fire protection system header in the northeast stairwell of the reactor building. Water from the stairwell entered Residual Heat Removal (RHR) Pump Room C through a watertight door that apparently had not been adequately closed and began rapidly flooding the room. Valve FDR-V-609, located in a line connecting the sumps of the RHR C and low pressure core spray (LPCS) pump rooms, failed to close as designed and allowed water to flow into the LPCS pump room. The flood water completely submerged the RHR C pump and motor and the keepfill pump for RHR B and C, which is also located in the room. Water in the LPCS pump room rose to a level just below the pump motor and also completely submerged the minimum flow valve and the keepfill pump, which serves both the LPCS and RHR A trains. The water level in the stairwell rose to a height of approximately 19 feet above the pump room floors, which are located at the 422-foot elevation.

The event was initiated by the actuation of fire detectors during cutting and grinding activities, which were taking place in the diesel generator building. The fire detectors, sensing the smoke from the maintenance activities, activated a fire protection preaction station, which caused the associated preaction valves to open and fill the normally dry sprinkler line header. (A second preaction station also actuated due to sympathetic effects.) However, no actuation of the associated sprinklers occurred since they are ultimately initiated by thermal fusible links. The depressurization of the fire water system during the filling of the preaction lines caused voiding in the upper portions of the reactor building vertical fire main risers and generated an autostart signal for all four main fire water pumps to start on low system pressure. Three of the pumps started immediately, and the fourth pump began a 30-second time delay sequence, as designed, for starting. The operation of the pumps resulted in a rapid reflood of the reactor building risers and caused a water hammer (i.e., a hydraulic pressure wave which propagated through the system) that ruptured a 12-inch cast iron fire protection system isolation valve located in the reactor building northeast stairwell.

Flooding, at a rate of between approximately 8,500 to 10,000 gpm occurred in the stairwell. The associated watertight door between the stairwell and the RHR C pump room apparently was not adequately secured and became open during the initial stages of the event and allowed the room to quickly flood. The watertight doors for the northeast stairwell access to the RHR C pump room and LPCS pump room were not designed or installed to prevent flooding from the stairwell from entering the associated pump rooms (the doors were designed to seal from inside the rooms). Upon determining that a fire did not exist, and based on indications and reports of the extensive flooding, plant operators secured the running fire pumps to stem the source of the floodwater. Due to the extensive flooding in the stairwell and the ECCS pumps rooms, coupled with the loss of the normal fire suppression system capability, a Notification of Unusual Event (NOUE) was declared by the licensee. The licensee activated the site emergency response organization and notified state and local officials, as well as, the NRC. To compensate for the reduced fire suppression capability, the Hanford fire department dispatched four fire trucks to the site as a compensatory measure. The Hanford units remained in place until the ruptured valve had been replaced and the normal fire suppression system had been placed in a configuration that was not susceptible to water hammer. At that time, the NOUE was terminated.

During the event, approximately 163,000 gallons of firewater were introduced into the northeast stairwell, RHR C, and LPCS pump rooms. Additionally, some minor leakage of floodwater occurred between the LPCS pump room and the vestibule, which separates that room from the adjacent high pressure core spray (HPCS) pump room. Water also leaked from the RHR C pump room into the adjacent reactor core isolation cooling (RCIC) room through a double watertight door arrangement, which separates those two rooms. During the dewatering of the stairwell and associated emergency core cooling system (ECCS) pump rooms, the licensee detected low levels of radioactive contamination in a sample taken from the floodwater. Ultimately, the licensee made a determination that no release of radioactive material above detectable levels had occurred as a result of the flood and subsequent discharge associated with dewatering the flood affected areas. Since the unit was in Mode 4, containment integrity was not required. Shutdown cooling was maintained throughout the event.

2.2 Initial Conditions

The plant was in Mode 4, cold shutdown, with an average reactor coolant temperature of approximately 133° F. The plant operators were making preparations for a plant startup and shutdown cooling was being supplied by RHR A train. Residual Heat Removal B, C, and LPCS pumps were in a normal standby lineup with both of their associated keepfill pumps running. The fire protection system was also in a normal lineup with all four main fire pumps in standby and system pressure being maintained at approximately 140 psig by the jockey pump. Fire Protection Isolation Valve FP-V-29D was in its normally open position. Maintenance craft personnel were conducting cutting and grinding activities in the Division II emergency diesel generator room in order to remove some supports for some abandoned equipment. The cross-connect valve (FDR-V-609) between the RHR C and LPCS sumps was in its normally open position.

2.2.1 Initiating Event

As mentioned in Section 2.1, the smoke from cutting and grinding activities in the Division II diesel generator room actuated the detectors associated with a fire protection preaction station. A second preaction station also initiated, however, the cause for its actuation appears to be due to a sympathetic response associated with the event. The actuation of the preaction stations caused a drop in system pressure, voiding in the vertical risers in the reactor building, and an autostart of the main fire pumps. The starting of the fire pumps resulted in a rapid reflooding of the now-voided risers. The subsequent void collapse caused a water hammer, which ruptured a fire protection valve in the northeast reactor building stairwell and caused significant flooding of important plant equipment from two separate divisions. A thorough discussion of the design and operation of the fire protection system is provided in Section 2.5.1 of this report. Details related to the design and performance of the plant flood protection features associated with this event are provided in Sections 2.5.2 and 2.5.3.

2.3 Operator Response, Procedural Use, and Adequacy of Procedures

The following sections provide a discussion of the primary operators' actions and procedural issues that were observed during the event.

2.3.1 Operator Response and Procedural Issues Associated with the Event

At approximately 1:45 p.m. (PDT), control room fire alarms for preaction Systems P66 and P81 annunciated in the control room and three of the four main fire pumps started. Several seconds later, the RHR C pump room high level alarm was received in the control room (this alarm annunciates when approximately 6 inches of water is above the floor in the room) and the plant operators entered Emergency Operating Procedure 5.3.1 due to high water level in an ECCS pump room.

Approximately 2 minutes later, RHR-P-3, the keepfill pump for RHR B/C tripped due to being submerged by the floodwater. Operators immediately started Pump RHR-P-2B in the suppression pool cooling mode to maintain system pressure. This action was prescribed by plant Abnormal Conditions Procedure 4.12.4.10, "Reactor Building 422 Area Flooding." The requirement to start RHR B due to flooding in the RHR C pump room is based on the necessity to maintain the system pressure in response to a loss of the keepfill pump which is located in the RHR C pump room. Additionally, the licensee's flooding analysis for the northeast stairwell (See Section 2.5) assumes that the operators will take the necessary actions to start the alternate train pump within 20 minutes following the event. In this case, the operators started the pump almost immediately and, thus, fulfilled the intent of the procedure. However, the augmented inspection team noted that the guidance in the procedure appeared to be weak. Specifically, the guidance directed the operators to "consider" starting the opposite train pump during pump room flooding rather than compelling the operators to do so in a timely manner as assumed in the flooding analysis. The issues associated with the guidance to start the alternate train RHR pumps during flooding scenarios will remain open pending further NRC review (50-397/9816-01).

At 1:51 p.m. (PDT), the operators determined that a fire did not exist and isolated Fire Protection Systems 66 and 81. Approximately 3 minutes later, the control room received a report that the northeast stairwell was flooded and that the fire main had ruptured. The operators gave the order to stop all four fire pumps. The fire pumps (as discussed in Section 2.5.1) can only be stopped at the local controllers and not from the control room. Due to the remote locations of the fire pumps, about 5 minutes were required to stop the three fire pumps in the circulating water pump house and about 7 minutes were required to stop the remaining pump. During the event, plant operators maintained communications with personnel outside the control room through the use of hand-held portable radios. However, several of the radios exhibited poor reception and had to be replaced during the event and in some cases alternate methods of communication were employed. Additionally, the control room telephone lines were congested during the event by inquiries from personnel who were not directly involved in mitigating the event. The licensee indicated that these communications deficiencies did not significantly impair the ability of operators to respond to the event. The issues associated with the performance of radio communications during the event will remain open pending further NRC review (50-397/9816-02).

At 2:01 p.m. (PDT), operators received the LPCS pump room high level alarm and reentered Emergency Operating Procedure 5.3.1 on ECCS pump room high level. At this point the operators started the LPCS pump, ostensibly to maintain system operability even though the room was flooding. However, Abnormal Condition Procedure 4.12.4.10 directs operators to stop any pumps which may be running in an affected room. (Subsequently, as the water continued to rise in the room, the operators stopped both the LPCS pump and the associated keepfill pump.) Discussions with station management indicated that the operators had acted in accordance with management expectations given the circumstances, notwithstanding, the procedural guidance. However, the AIT noted that no formal process had been invoked which authorized the plant operators to deviate from the operating procedures. The issues associated with the operators' decision to start the LPCS pump will remain open pending further NRC review (50-397/9816-03).

At 2:14 p.m. (PDT), based on the loss of the RHR C, LPCS, and fire protection systems, the plant operators declared an Unusual Event due to flooding and activated the site emergency response organizations. A more detailed discussion of the event classification is provided in Section 2.4 of this report.

Following the rupture and loss of normal fire suppression capability, the Hanford fire department was notified. The fire department dispatched fire engines to the site to serve as an interim compensatory measure. Additionally, the licensee began to implement other, plant-specific, compensatory measures due to the loss of normal fire suppression capability. However, the licensee identified in Problem Evaluation Report 298-0800 that not all of the required compensatory measures were implemented within the required time period. The majority of the measures, which were not accomplished, either were of an administrative nature (i.e., not writing fire impairments) or were not applicable due to the unique circumstances involved with this event (i.e., placing additional hoses at impaired fire stations). .However, it was noted that continuous fire watches were not

posted in certain vital areas as required. Specifically, the cable chase area and cable spreading room associated with Fire Protection Systems 65 and 66 were not properly monitored. The issues associated with the compensatory measures associated with the loss of the normal fire suppression system will remain open pending further NRC review (50-397/9816-04).

Once the fire pumps had been stopped, the source of the flood water had been eliminated. Operators removed the equipment plugs from above the rooms and noted that the RHR C pump room had approximately 19 feet of water in the room and had completely submerged the pump and motor. Water in the LPCS pump room had not yet reached the level of the pump motor, but was continuing to rise. Operators determined that the normally open sump cross-connect Valve FDR-V-609 between the RHR C and LPCS pump rooms had failed to close as designed. (This valve had been known to be in a degraded condition prior to the event. See Section 2.5.3 for a discussion of the design and performance of the reactor building floor drains system.) Since Valve FDR-V-609 had failed to close automatically and also would not close when operators manipulated its control switch from the control room, the system engineer developed a method to close the valve remotely. This was accomplished by first bypassing the solenoid valve and pressurizing the accumulator for the valve and then removing the bypass to relieve the supply pressure. However, these activities were not proceduralized and approximately 3 hours transpired from the time of the initiation of flooding until the valve was eventually closed.

Based on their knowledge that the cross-connect valve remained open, operators began activities to lower the water in the LPCS pump room before the level equalized with that in the RHR C pump room and submerged the LPCS motor. Submersible pumps were used to begin pumping water from the LPCS pump room back into the RHR C pump room since the equipment in that room was already under water. Concurrently, the Technical Support Center (TSC) and Operations Support Center (OSC) were being activated and preparations were underway to dewater the flooded stairwell and ECCS pump rooms. Section 2.3.2 of this report provides a detailed description of the activities associated with dewatering the flood affected areas.

A review of the maintenance activity, which caused the actuation of preaction System 66, was conducted. It was determined that the craft personnel performing the evolution acted in accordance with the guidance in effect (Procedure 1.3.10A, "Control of Ignition Sources"), but that the guidance apparently did not provide adequate direction to the craft in order to avoid an almost certain, inadvertent preaction of the fire protection system. Further, it was determined that a similar event had occurred on February 5, 1998, whereby maintenance activities had caused an inadvertent preaction of this same system, which resulted in a water hammer and that corrective actions for the earlier event had not been implemented. The previous event was documented in the licensee's corrective actions process via Problem Evaluation Report 298-0112. The issues associated with the procedural guidance for the control of ignition sources will remain open pending further NRC review (50-397/9816-05).

2.3.2 Operator and Technical Support Center Actions to Dewater the Areas Affected by Flooding

After the rupture of the fire protection valve, the licensee commenced preparations for discharging the water from the stairwell and RHR C and LPCS pump rooms using portable pumps and hoses. The operations shift supervisor, acting in the capacity of emergency director pending the activation of the TSC, dispatched a senior reactor operator and a reactor operator to the stairwell area to oversee the recovery operations.

At 2:15 p.m. (PDT), a one liter dip sample was taken from the surface of the water in the stairwell and a 4000-second count for radioactive contamination was begun. The operations shift supervisor ordered that no water be pumped from the stairwell into uncontaminated systems until the sample was properly counted. Concurrently, personnel were directed to make preparations to pump the stairwell to the T-4 condenser sump. This sump had a capacity of more than 500,000 gallons, and was normally used by the licensee to collect both contaminated and potentially contaminated liquids for processing and eventual release. In preparations for a potential discharge to the storm drain system, a composite sampler was emptied and reset. The composite sampler is normally used to verify, after the fact, that discharges to the storm drain system were below prescribed limits.

At 2:48 p.m. (PDT), the TSC was activated, and TSC personnel began discussions on pumping water from the stairwell and the ECCS pump rooms in order to minimize the amount of water in the LPCS pump room. (A report had been received that the water level in the LPCS pump room was approximately 2 feet and rising and that the RHR C pump and motor were completely submerged.) The TSC asked chemistry personnel for preliminary results on the sample taken from the stairwell, which was collected at 2:15 p.m. (PDT). The chemistry personnel indicated that based on their review of the counting, which was still in progress, they believed there would be no detectible radioactive contamination in the sample. At 3:00 p.m. (PDT), the TSC director, after consultation with other TSC personnel, gave the order to start pumping water from the stairwell to the storm drains and ultimately to Storm Drain Pond ST-101. The Storm Drain Pond ST-101 was located outside the protected area, but within the owner controlled area. Although not designated for radioactive discharges, the pond already contained very low levels of radioactive contamination. The augmented inspection team observed that the pond was fenced, locked, and posted as a radioactive material area.

At 3:04 p.m. (PDT), the emergency director function was transferred from the shift supervisor to the TSC director. The augmented inspection team noted that the function of emergency director was transferred to the TSC only after the order had been given (by the TSC) to commence discharging to the storm drain pond. Additionally, the augmented inspection team noted that the order to discharge to the pond had been given without the prior knowledge and consent of the operations shift supervisor, who was acting in the role of emergency director at the time.

At 3:20 p.m. (PDT), a second one liter grab sample was taken from the water being pumped from the stairwell to Storm Drain Pond ST-101. At 3:35 p.m. (PDT), the counting of the first sample was completed and indicated that no isotopic concentration values greater than the lower level of detection (LLD) for Co-60 were present in the sample. (The LLD for this sample was 1.5E-08 µCi/ml.) At 4:20 p.m. (PDT), chemistry personnel notified the TSC that the second grab sample was indicating a possibility of Co-60 activity greater than the calculated LLD limit. The licensee terminated pumping to the storm drain pond and routed the hoses to the T-4 sump. During the initial stages of dewatering, approximately 16,200 gallons had been pumped to Storm Drain Pond ST-101. All remaining water in the stairwell, RHR C pump room, and LPCS pump room was pumped to the T-4 sump. The augmented inspection team reviewed the available evidence and concluded that all the water from the fire protection system rupture was pumped either to Storm Drain Pond ST-101 or to the T-4 sump.

At 4:34 p.m. (PDT), a third one liter grab sample was obtained from the water in the stairwell, now being pumped to the T-4 sump. At 4:50 p.m. (PDT), the count of the second grab sample was completed and the sample indicated a Co-60 concentration of 5.21E-08 µCi/cc. No other isotopic concentrations greater than the LLD limits were noted.

At 5:25 p.m. (PDT), the composite sample of the discharge to Storm Drain Pond ST-101 was collected. Because chemistry personnel had set the sampling ratio of the composite sampler to a rate of 10 ml per 1000 gallons, only 171 ml were obtained. (Typically, a one liter sample is collected for analysis using the composite sampler.) After a 12-hour background count and a 12-hour sample count of the 171-ml composite sample, the licensee detected no activity above background. However, because of the small sample size, the minimum detectible activity (MDA) for this particular sample was only 5.0E-08 µCi/ml. The LLD for this sample was 1.50E-08 µCi/ml. Thus, based on the small sample size, the resolution of the licensee's counting process was not sufficient to ensure that the MDA was below the LLD limits. At 6 p.m. (PDT), the analysis of the third grab sample was completed. This sample indicated no isotopic concentrations greater than the LLD limits. After review of the above sequence, the licensee concluded in Problem Evaluation Report 298-0783 that based on their best available methodology, that there was no detectable radioactive discharge to Storm Drain Pond ST-101 and that no reporting was required. Subsequently, the licensee submitted the sample to an offsite laboratory with enhanced analytical capabilities. The results of the offsite analysis showed that the Co-60 concentration in the sample was less than the LLD limit of 1.50E-08 µCi/ml. The issues associated with the sampling methodology relative to the discharge to the storm drain pond will remain open pending further NRC review (50-397/9816-06).

The augmented inspection team determined that the direction to begin pumping the water from the stairwell to Storm Drain Pond ST-101 was given from the TSC before the TSC had properly assumed emergency control. Operations personnel in the control room and operations personnel directly overseeing activities in the stairwell indicated that they were surprised when the pumping to Storm Drain Pond ST-101 began. This operation countermanded their decision to not pump from the stairwell until the first grab

sample analysis was completed. The TSC director informed the augmented inspection team that the TSC was concerned with the rising water level in the LPCS pump room. TSC personnel indicated that they realized the potential for contamination from the pump rooms to migrate to the stairwell, however, they considered that since the doors were designed to seal from inside the pump rooms that the contamination would most likely be contained within the rooms. Further, the licensee indicated that to minimize pumping contaminated water to Storm Drain Pond ST-101, grab samples would be taken throughout the evolution. The TSC director stated that the TSC had considered the potential risk of pumping slightly contaminated water to Storm Drain Pond ST-101 against the need to lower the water level in the stairwell as soon as possible to prevent submergence of the LPCS pump motor and decided that the best risk-informed action was to begin pumping to the pond. The issues associated with the coordination and control associated with transferring authority to the TSC will remain open pending further NRC review (50-397/9816-07).

The licensee further stated that the second grab sample taken from the stairwell, which indicated 5.8E-08 µCi/ml was well below the 10 CFR Part 20 release of liquid effluent concentrations to unrestricted area's limit of 3.0E-06 µCi/ml for Co-60. The licensee acknowledged that discharge to Storm Drain Pond ST-101 was not covered by this limit, and that they were required to report any discharge of liquid effluent with isotopic concentrations greater than the LLD limit into Storm Drain Pond ST-101 to the state and, subsequently, the NRC. The licensee's instructions for radiological sampling of liquids from sumps and associated areas were contained in Chemistry Procedure 12.2.14, Revision 2, "Batch Release of Nonradioactive Liquid." This procedure indicated that release of effluent to the storm drains could only be accomplished if the effluent was not radioactive. For purposes of Co-60, Chemistry Procedure 12.5.28, Revision 7, "Sampling and Analysis for Unrestricted Release," indicated that the LLD limit was 1.5E-08 µCi/ml. The licensee's Offsite Dose Calculation Manual, Section 6.2, described LLD calculations. The licensee's procedures for discharge of liquid effluents to storm drains were based on the site's National Pollutant Discharge Elimination System Wastewater Discharge Permit. Section S1.F of this permit included permission to discharge nonradioactive plant leakage to the storm drain system.

The augmented inspection team reviewed the information associated with the discharge to Storm Drain Pond ST-101. The licensee stated that they were already maintaining a decommissioning file on Storm Drain Pond ST-101, and that the information related to the samples taken of this discharge would be added to the file. Based on this information and the fact that the pond was already controlled as a radioactive material area, the augmented inspection team determined that there was no risk of loss of control of contamination and that no new contaminated areas were created.

The augmented inspection team noted that the decision to set the composite counter at 10 ml per 1000 gallons did not appear to be conservative. This was based on the fact that TSC personnel knew that as the water level in the stairwell was lowered, back leakage from the RHR C pump room could flow into the stairwell, changing the radioactive isotropic concentrations in the water. (In fact, it was subsequently determined that the pump room door remained open throughout the event because of a

large toolbox obstructing the ability of the door to close.) Therefore, the water in the stairwell might not be a homogeneous batch, and grab samples might not be representative of the actual discharge. In addition, the licensee was concurrently pumping water from the RHR C pump room to the T-4 sump. Based on this information, the augmented inspection team concluded that the licensee did not have an accurate estimate of the volume of water to be pumped to Storm Drain Pond ST-101, and should have understood that the grab samples might not be representative of radioactive content. Therefore, in order to either limit or accurately reflect any radioactive effluent being discharged to the pond, the composite counter should have been used. Had the composite counter been set to a higher rate, in-process readings could have been taken and sufficient samples obtained to characterize the discharge with respect to the normal LLD limits. However, even though the resolution of the licensee's analysis was not sufficient to ensure that the discharge to Storm Drain Pond ST-101 was below the allowable limits, analysis at an offsite laboratory with a superior counting capability confirmed that the discharge was less than the LLD limits and that no detectable radioactive material was discharged to the pond. The issues associated with the decision making associated with discharging to the storm drains system will remain open pending further NRC review (50-397/9816-08).

2.4 Emergency Response and Event Classification

A review of the licensee's emergency response guidance revealed that there was no clear event classification that corresponded to this event. Classification of the flooding as an Unusual Event was made under the "Other" or judgment category which reads, "In the judgment of the Emergency Director, events are in process or have occurred which indicate a potential degradation of the level of safety of the plant" (Emergency Action Level 9.1.U.1). It was determined that the notification of state and local agencies occurred 7 minutes after declaration of the event and that notification of the NRC occurred 20 minutes after the declaration was made.

The control room manager, acting in the capacity of emergency director, indicated that the RHR pump room high level alarm in itself was not a sufficient degradation in the level of safety of the plant to declare an NOUE because operation in Mode 4 without one RHR pump is allowed by the technical specifications. Additionally, shutdown cooling had been verified to be adequate and was being provided by a non-affected pump. The control room manager indicated that securing of the fire suppression system, in conjunction with the losses of both RHR C and LPCS, were the events that resulted in a review of the emergency plan and declaration of the event. Based upon interviews and a review of operating logs, it was determined that an Unusual Event was declared in less than 15 minutes after stopping the fire pumps.

The emergency director indicated that his decision to not upgrade the event classification to the "Alert" category was based upon the fact that for the operating mode which was in effect (Mode 4), sufficient technical specification equipment had been verified to be available. Additionally, it had been verified that safe shutdown capabilities were being maintained and that the required shutdown system parameters did not indicate degraded performance. The train of the RHR system providing shutdown cooling was unaffected

and was performing its intended function. It was also known that the initiating event had been terminated when the fire protection system was secured and that a fire was not in progress. Further, all work that had the potential to pose as a fire hazard was terminated and offsite fire assistance was reporting to the site to provide standby fire protection coverage while the normal fire suppression system was out-of-service. The licensee reported that these decisions were made with input from operations and emergency preparedness management. The emergency director also reported that his initial decision was to not activate the site emergency response organization. However, due to the magnitude of incoming calls and the resulting anticipated response efforts, his decision was quickly revised and that the decision to activate the TSC and OSC occurred at the time of the declaration of the Unusual Event.

2.5 Plant and Equipment Response

General Plant Flood Design Characteristics

The team reviewed documentation, results of analyses, and presentations by the licensee to determine how the plant's design basis compared with the actual stairwell flooding event which occurred. It was reported by the licensee that the design basis flood for the reactor building northeast stairwell assumed a fire protection system pipe crack opening of ½ the pipe diameter in length and ½ the pipe wall thickness. This size break would result in a postulated flowrate of approximately 300 gpm. The flood event, which occurred on June 17, 1998, was a full circumferential break in Fire Protection System Valve FP-V-29D and resulted in an equivalent flowrate of between 8,000 and 10,000 gpm.

In the design basis case, the doors between the pump rooms and the stairwell are assumed to be closed and the double doors between adjacent pump rooms are assumed to be watertight. Additionally, the wall penetrations are also assumed to be watertight. The doors between the pump room and the stairwell are assumed to leak at a rate of 60-70 gpm, while the pump room doors are assumed to be watertight from leakage originating from within the rooms. During the flooding event, which occurred, the door to the RHR-C pump room was open and the double doors leaked slightly while the penetrations remained watertight. (A thorough discussion of the design and performance characteristics of the watertight doors is provided in Section 2.5.2.)

In addition to the above, the design basis assumes that Isolation Valve FDR-V-609 in the piping, which connects the RHR-C and LPCS pump room sumps, would either close automatically or fail closed. In the actual flood event, the cross-connect valve did not close and could not be closed from the control room. The water level rose to about 6 inches above the maximum safe operating limit in the LPCS pump room. Further, due to the magnitude of the flooding and the open pump room door, water rose well above the maximum safe operating limit for the RHR C pump room. The RHR and LPCS systems provide low pressure injection capability for the plant and are designed to mitigate the effects of a loss of coolant accident. Additionally, the RHR system provides the normal means of shutdown cooling as well as suppression pool cooling. The RCIC system is normally used for initial plant cooldown when the reactor is isolated from the

condenser. (A discussion of the design and performance characteristics of the reactor building floor drains system is provided in Section 2.5.3.)

The licensee's flooding design basis also assumes that no safety-related equipment susceptible to submergence effects, or that could have an adverse impact on the operability of the system while being submerged, is installed below the maximum safe flooding depth in the ECCS pump rooms. During the flooding event, the RHR C pump room fully submerged and the adjacent LPCS pump room flooded to a depth of 64 inches. Additionally, Divisions I and II of the Automatic Depressurization System (ADS) logic pressure switches, the RHR B/C keep fill pump, the RHR C pump, the LPCS/RHR A keepfill pump, and the LPCS mini-flow recirculation valve were submerged during the event. The analysis assumed no credit for operator action in the control room for 20 minutes and in the plant for 30 minutes following the initiation of the event. In the actual flood event, the operators took prompt actions to start the RHR-P-2B pump in the suppression pool cooling mode after identifying that the RHR-P-3 keepfill pump had tripped. It was determined that previous analyses suggested that unchecked flooding in the northeast stairwell of the reactor building could potentially affect all plant ECCS systems since all keepfill capability is within the northeast stairwell pump rooms. The issues associated with the review of previous analyses of stairwell flooding will remain open pending further NRC review (50-397/9816-09).

The augmented inspection team determined that the potential for a fire main rupture scenario was previously identified in Licensee Event Report 92-034-02 (June 24, 1993) and in plant Engineering Assessment SS2-PE-0591. Additionally, this scenario had been evaluated in other related plant engineering studies. The augmented inspection team noted that, based on the events that actually occurred, the corrective actions and assumptions contained in these analyses appeared to have been flawed. It was also noted by the team that the licensee's risk analysis had employed similar assumptions to those used in the design basis studies. Consequently, the licensee's risk analysis did not accurately model the likelihood or consequences of the scenario that occurred. The issues associated with the licensee's assumptions and corrective actions associated with LER 92-034-02 and SS2-PE-0591 will remain open pending further NRC review (50-397/9816-10).

The following sections provide a more detailed discussion of the design and performance characteristics of the plant systems, which were involved in the June 17, 1998, flooding event.

2.5.1 Fire Protection System

Fire Protection System Design Considerations

The fire protection system at WNP-2 consists of two diesel-driven and two electric-driven fire pumps. The Electric Pumps FP-P-2A and FP-P-2B and Diesel Pump FP-P-1 have a capacity of 2000 gpm and draw a supply from the circulating water basin. Diesel Driven Pump FP-P-110 has a capacity of 2500 gpm and is supplied by a 400,000 gallon embankment-supported fabritank (i.e., bladder). The fire pumps are normally in standby

and the system pressure is maintained at approximately 150 psig by a 220-gpm jockey pump. The system is arranged such that the pumps supply a main header, which in turn supplies various yard hydrant isolation valves and building standpipes. Additionally, the design also includes a number of preaction systems, which are not normally filled with water. Upon actuation of the associated detector(s) for a given preaction system, the preaction system valves open and allow water to flow from the main header into the associated piping. Some of the plant preaction systems are activated by ionization-type detectors (i.e., preaction System P66), whereas other preaction systems rely on thermal detectors (i.e., preaction System P81) for actuation. However, the sprinkler heads associated with the downstream piping are not actuated during a preaction unless the thermal-fusible links are melted on the individual heads, thereby, completing the flow path for the firewater.

The fire pumps are designed to start automatically on low system pressure in response to demands on the system. Pump FP-P-2A starts when pressure drops to 120 psig, Pump FP-P-2B autostarts at 110 psig, and Pump FP-P-110 will start if system pressure continues to fall to 100 psig. Pump FP-P-1 will start if pressure drops below 110 psig after a 30-second time delay. Additionally, the fire pumps can be started locally or from the control room. However, the fire pumps can only be stopped at the local controller. It was noted that Fire Pumps FP-P-2A, FP-P-2B, and FP-P-110 do not have a time-delay feature associated with the autostart circuit, but rather rely solely on the pressure signal. This particular aspect of the system design is being reviewed by the licensee.

Fire Protection System Performance During the Event

As noted earlier, on June 17, 1998, maintenance personnel were performing cutting and grinding activities within the Division II diesel generator room. This activity made use of a large acetylene cart which employed hoses that needed to be routed through the door leading into the space. A fire watch was established in accordance with plant procedures based on the need to maintain the fire door in an open configuration. Due to the fact that the grinding activities were generating large quantities of smoke, the detectors for preaction System 66 were activated. This resulted in the associated preaction valves opening to fill the previously empty piping associated with that system. Shortly thereafter, the preaction valves for System 81, which serves the diesel generator room, also actuated causing that system to fill. Since none of the thermal links on the sprinkler heads had been melted (i.e., no fire was present) no actual discharge of fire water occurred and no plant equipment was wetted via the sprinkler systems.

Upon the actuation of System 66, system pressure dropped rapidly (in the first second) due to the high demand caused by the filling of the downstream piping. This demand exceeded the capacity of the jockey pump and system pressure dropped to about 32 psig, which generated an autostart signal for all of the plant fire pumps. (It was determined that the actuation of System 81 occurred about 4 seconds into the transient.) Fire Pumps FP-P-2A, FP-P-2B and FP-P-110 started immediately since their autostart set point had been reached. Additionally, Pump FP-P-110 received an autostart signal and began its 30-second time delay sequence for starting. The pump started, as

designed, 30 seconds later due to the fact that system pressure had not risen above its 110 psig set point.

Shortly after the autostart of the first three fire pumps, several loud water hammer noises were heard throughout the plant and a rupture of Fire Protection Isolation Valve FP-V-29D occurred. This valve was located at the 441-foot elevation in the northeast reactor building stairwell riser. The rupture had the effect of increasing the fire protection system demand and kept system pressure from rising above the set point for Pump FP-P-1 before the 30-second time delay had expired. Thus, shortly after the rupture of Valve FP-V-29D, all four fire pumps were running and supplying full flow through the ruptured valve.

Subsequent investigations by the licensee indicated that the rupture of FP-V-29D was most likely the result of a significant water hammer event. Metallurgical examinations of the failed valve did not identify any preexisting defects which would be indicative of incipient valve failure. Additionally, a review of the maintenance history of the failed valve and other similar valves did not suggest that any structural deficiencies existed. (The valve was a 12-inch cast iron gate valve with a nominal thickness of approximately 3/4 inches.) Further, a structural analysis conducted by the licensee showed that the valve failure was almost certainly due to severe torsional stresses caused by the water hammer.

After the event, the licensee performed extensive engineering and hydraulic analyses in an attempt to model the event. It was determined that the actuation of System 66 resulted in a rapid drop in the water column in the fire protection riser in the northeast stairwell of the reactor building. (This riser is the high point in the plant fire protection system and rises approximately 180 feet from the lowest level of the reactor building.) The rapid drop of the water column resulted in significant voiding in the riser. The subsequent starting of three of the fire pumps served to rapidly repressurize the system and collapse the void. The net effect of these phenomena was the generation of large pressure waves resulting in a significant water hammer in the northeast stairwell riser. Due to various geometrical considerations associated with the system configuration, these forces were concentrated in the vicinity of Valve FP-V-29D and ruptured the valve.

The team found that a previous actuation of System 66 during maintenance activities had also resulted in a water hammer event at the facility and that corrective actions had not been implemented to address the issue. This event was documented by the licensee in a condition report (Problem Evaluation Report 298-0112) dated February 5, 1998, whereby cutting and grinding activities associated with System 66 had caused a similar preaction actuation and subsequent water hammer that had over-ranged the pressure gage at the top of the reactor building riser. (It should be noted that during the February 5, 1998 event, Pump FP-P-2A was running and Pump FP-P-110 auto started following the actuation of System 66.) The issues associated with the corrective actions associated with previous water hammer events will remain open pending further NRC review (50-397/9816-11).

The actuation of System 66 was an expected (although inadvertent) system response due to the large amount of smoke which activated the ionization detectors for that system. However, the actuation of System 81 was not an expected system response due to the fact that System 81 employed thermal detectors for actuation. These detectors were located at a significant distance from the cutting and grinding activities and a valid initiation was not a credible explanation for their actuation. Subsequent investigations by the licensee determined that deficiencies in the System 81 check valve trim assembly may have contributed to the sympathetic actuation of this system. The issues associated with the unexpected preaction of system 81 will remain open pending further NRC review (50-397/9816-12). The team noted that the licensee may have had a history with spurious or sympathetic preactions in that shortly after the event, a deficiency was written against System 81, which indicated that it had actuated concurrently with System 66 on three separate occasions in the past. However, the licensee stated that no formal documentation of these scenarios could be located. Additionally, the team found that many plant preaction stations have warning placards affixed, which indicate that the potential for multiple preactions exists during system testing. The issues associated with the history of multiple preaction scenarios at WNP-2 will remain open pending further NRC review (50-397/9816-13).

Following the event, the licensee analytically determined that with two fire pumps running continuously, any two simultaneous preaction actuations would not generate enough system voiding to induce another significant water hammer. On the basis of this analysis, the licensee reconfigured the system with both electric pumps running continuously and declared the system to be operable, but degraded. Subsequently, based on further analysis, the licensee established a nitrogen bubble at the top of the two highest stairwell risers. The nitrogen bubble was implemented to act as a "cushion" against potential water hammer events.

The team determined that the fire protection system had operated in accordance with the design description provided in the licensee's UFSAR, with the exception of the actuation of preaction System 81. The actuation of System 66 was in response to the smoke being generated from the maintenance activity and represented a valid demand. All four of the plant fire pumps started in response to valid initiation signals and in accordance with their prescribed setpoints. However, the augmented inspection team did not conduct a review of the system with respect to compliance with the National Fire Protection Association (NFPA) code.

After extensive review, the licensee concluded that the root cause of the event was due to a significant water hammer, which ruptured Valve FP-V-29D. Further, the licensee concluded that the design of the fire protection system was inadequate in that the system is configured such that destructive forces are generated during anticipated transients when the system is in a normal lineup. The issues associated with the design of the fire protection system will remain open pending further NRC review (50-397/9816-14).

2.5.2 Watertight Doors

Watertight Door Design Considerations

The watertight doors for the ECCS pump rooms were approximately 6 ½-feet high by 2 ½-feet wide. Once closed, the doors were secured by a mechanism, which inserted eight 1 1/4-inch diameter round metal rods into receptacles in the door frame. The receptacles were 1 ½-inches deep and the securing mechanism was controlled by a single handle on each side of the door. The licensee's design basis for the ECCS pump rooms (FSAR Sections 3.4.1.4.1.2 and 3.4.1.5.2) indicated that the rooms were "watertight," and that the effects of flooding in one room would not propagate to adjacent rooms. (Adjacent ECCS pump rooms at WNP-2 are separated by a vestibule, which incorporates a double watertight door arrangement. The arrangement addresses the fact that the doors are only watertight from one direction and that the double door arrangement is necessary to keep floods from propagating to adjacent rooms.) However, the doors, which lead into the RHR C and LPCS pump rooms in the northeast stairwell, as well as, those leading into the RHR A and RCIC rooms on the northwest stairwell, are not watertight from the stairwell direction; that is, these doors are not designed to prevent a flood in the stairwell from propagating into the associated pump rooms, and no double door arrangement exists in the stairwells.

The licensee's flooding analysis for stairwell floods acknowledged that the doors were not watertight from the stairwell, but that the door would limit the flow into the room to about 70 gpm. The licensee's flooding analysis for stairwell flooding relied significantly on timely operator actions to start either the RHR A or B pumps in the event of a flood in the northeast stairwell in order to maintain low pressure ECCS and shutdown cooling capabilities due to the fact that all low pressure ECCS keepfill capability could be lost in a stairwell flooding scenario. However, it was determined that discrepancies existed with respect to the as-built configuration and the licensing basis documents associated with the watertight doors. Specifically, Section 10.4.5.3 of the UFSAR stated that the doors would be watertight and designed to withstand a static head of 44 feet of water. Further, the UFSAR stated that the ECCS rooms are designed to be watertight for stairwell flooding. The issues associated with the design of the watertight doors will remain open pending further NRC review (50-397/9816-15).

Watertight Door Performance During the Flooding Event

The licensee postulated that the door to the RHR C pump room was either left in an unsecured condition sometime prior to the event or became unsecured during the event. A review of a door alarm printout from the common alarm station indicated that the door had changed state several minutes before the event. (The doors are monitored by limit switches, which indicate the position of the door. However, the licensee stated that this monitoring capability was not being routinely reviewed nor was the equipment used for monitoring being maintained.) Upon dewatering the stairwell, the door was found to be open with a tool box obstructing the ability of the door to close. It is believed that the toolbox (which is normally stored in the stairwell) floated into the doorway at some point during the event. With the door in an unsecured or opened condition, an unrestricted

pathway existed for flood water to flow from the stairwell into the RHR C pump room in that the door opens into the room. The issues associated with the position or securing of the door to the RHR 2C room during the event will remain open pending further NRC review (50-397/9816-16).

Additionally, it was noted that during the flooding event the watertight doors leading out of the RHR C and LPCS pump rooms leaked from the direction in which they are designed to be watertight. Water was found in the vestibule, which separates the LPCS and HPCS pump rooms, and water leaked into the RCIC room through the vestibule, which separates the RHR C and RCIC rooms. The issues associated with the performance of the watertight doors will remain open pending further NRC review (50-397/9816-17).

Early on June 18, 1998, the augmented inspection team observed that the R13 door (the door from the stairwell to RHR C) operated very freely, with no noticeable resistance when turning the handwheel. The full travel of the mechanism was a little more than one complete turn of the operating handle. The mechanism operated so freely that if the handle was spun and released, the securing mechanism would fully insert the rods into their receptacles with enough force to bounce the rods off the back of the receptacles and return the rods to the fully retracted position. Interviews with plant personnel suggested that it was common knowledge that this particular door had a somewhat freewheeling mechanism. However, the augmented inspection team noted that no formal documentation of this deficiency existed. Additionally, it was discovered that this particular door had an outstanding work order written against it (WO DJW2), which documented a deficiency in the sealing surface of the door. This deficiency had been noted in early 1996, but had not been dispositioned as of the event. However, the licensee stated that the work order had been assigned a low priority due to the fact that a determination had been made that the deficiency was of a minor cosmetic nature. The issues associated with the maintenance deficiency associated with the RHR 2C door will remain open pending further NRC review (50-397/9816-18).

During subsequent checks of the closing mechanisms on the watertight doors associated with other safety-related equipment rooms, the augmented inspection team observed:

- Many doors took only about 135 degrees of handle travel to allow fully inserted rods to clear the door frame, which would allow opening the door.
- Some door handles turned with no resistance to fully insert or retract the rods.
- Some door handles initially required no resistance to turn, but encountered sudden resistance when the operating mechanism rods entered their receptacles.
- On one occasion, a door was closed but the rods were not fully inserted.

Further, the augmented inspection team noted that although the ECCS pump room watertight doors were in the scope of the licensee's maintenance rule program, the monitoring and performance criteria, which were established did not appear to address

the function of the door from the perspective of mitigating the effects of stairwell flooding. The performance monitoring established by the licensee addressed only the function of the doors with respect to sealing from floods emanating from inside the pump rooms. The issues associated with the performance monitoring of the watertight doors will remain open pending further NRC review (50-397/9816-19).

The augmented inspection team determined that the watertight door position was recorded, but not alarmed or used to indicate the actual condition of the doors. Based on the operating features of the watertight doors, the augmented inspection team observed that the door securing mechanism operation, coupled with a failure to use existing alarms, made insuring proper door closure more difficult. Further, the augmented inspection team agreed with the licensee's determination that the door to the RHR C pump room had most likely been left in an unsecured or open condition and that this contributed directly to the extent of the flooding in the ECCS pump rooms.

2.5.3 Reactor Floor Drains System

Reactor Floor Drains Design Considerations

The design for the floor drains includes two separate systems. One system handles drainage from the drywell, whereas, the other collects and handles drainage from all floor drains located in the remaining portions of the reactor building.

The floor drain system associated with the ECCS pump rooms consists of four separate sumps. One of the sumps serves the RHR B pump room, while the other three serve two equipment rooms each. A single, nonsafety-related, air-operated valve (AOV) exists in the piping between the rooms associated with the sumps, which are cross connected. The following table provides a summary of the sump locations, rooms served, and the associated cross-connect valves:

Floor Drain Sump	Location	Rooms Served	Cross-Connect Valve
FD-R-1	RHR A Pump Room	RCIC and RHR A	FDR-V-607
FD-R-2	RHR B Pump Room	RHR B	n/a
FD-R-3	HPCS Pump Room	HPCS and CRD	FDR-V-608
FD-R-4	RHR C Pump Room	RHR C and LPCS	FDR-V-609

The isolation valves associated with the cross-connected sumps employ the use of a solenoid and a four-way shuttle valve. Additionally, each valve has a small accumulator, which is normally pressurized and provides the motive force for valve closure when the

normal air supply is unavailable. As designed, a loss-of-normal air pressure or loss-of-power to the solenoid would result in closure of the valve provided that the associated accumulator is sufficiently pressurized. Thus, even though the valve is designed to fail closed, it still requires a motive force (i.e., air from the accumulator) to achieve closure. The valves receive a signal from a limit switch in the sump and are designed to close on a high water level in the sump.

The augmented inspection team noted that significant discrepancies existed between the as-built configuration, design description, and licensing documents with respect to the sump isolation valves. In particular, Section 3.4.1.4.1.2 of the UFSAR indicated that common mode flooding between watertight rooms would be prevented in that the floor drain sumps serving more than one pump room have two isolation valves in the drain headers from adjacent pump rooms, which automatically close on a high sump level. Further, it was implied that the design would meet single-failure considerations. However, these assertions appear to be inconsistent with Section 9.3.3 of the facility's Safety Evaluation Report (SER), which implied that a single, nonsafety-related, fail-closed valve existed. Additional documentation on this issue was noted in the licensee's response to NRC Inspection and Enforcement Circular No. 78-06, dated May 30, 1978, which concluded that even though the design of the isolation valve is not Seismic Category 1 or Class 1E, the acceptance criterion of Standard Review Plan 3.6.1 was met. The issues associated with the design of the sump isolation valves will remain open pending further NRC review (50-397/9816-20).

Finally, it was noted that sump pumps are provided for the individual reactor building sumps. These pumps are nonsafety related and receive automatic start signals to pump the sumps to the radwaste system when the associated sump reaches a prescribed level. The capacity of the sump pump for the FD-R-4 sump located in the RHR C pump room is approximately 50 gpm. (It was noted that the sump pumps were not credited in the licensee's flooding analyses.)

Reactor Floor Drains Performance During the Event

Immediately following the rupture of Valve FP-V-29D, water began flowing from the fire protection system into the northeast stairwell and into the RHR C pump room through the open door. The pump room high level alarm was received within 1 minute of the initiation of the event. The room level indication is via a limit switch inside the room, which alerts operators that water is at the 6-inch level above the floor. At this point, the RHR C sump would have been completely filled and the limit switch in the sump, which controls the automatic closure of Valve FDR-V-609 (the isolation valve in the cross-connect piping between the RHR C and LPCS pump rooms) would have been actuated. Early in the event, operators realized that Valve FDR-V-609 had failed to close automatically and attempts to close the valve using the hand switch in the control room were unsuccessful. As a result of the failure of Valve FDR-V-609 to close, water flowed from the RHR C pump room through the 3-inch cross-connect piping into the LPCS pump room. Eventually, water in the LPCS pump room exceeded the maximum safe operating level and rose to a level just below the pump motor before recovery efforts stabilized the water level. Subsequent calculations performed by the licensee estimated that approximately

90 percent of the water that in the LPCS pump room entered through the failed isolation pathway. The system engineer was eventually able to close Valve FDR-V-609 (more than three hours later and after the flood source had been isolated) by bypassing the solenoid and repressurizing the accumulator associated with the valve.

The augmented inspection team noted that the hand switch for Valve FDR-V-609 in the control room had a deficiency tag that referenced Work Request WR 98000410, dated January 1998. The work request had been issued due to the fact that the valve had failed to close during the calibration of its associated level switch. However, it appeared that no effective corrective actions had been taken to ensure that the valve would operate as designed and that no compensatory measures had been established. The augmented inspection team determined that this preexisting condition significantly contributed to the consequences of the flooding event. The issues associated with the corrective actions associated with the degraded performance of FDR-V-609 will remain open pending further NRC review (50-397/9816-21). Further it was determined that Valve FDR-V-609 was not in the scope of the licensee's maintenance rule or inservice testing programs. Additionally, no periodic preventive maintenance or testing was being performed on the valve and that the last known systematic preventive maintenance was performed in 1995. The licensee stated that the valve had previously been subject to a 60-month maintenance task but that this had been discontinued during an effort to eliminate "unnecessary" periodic maintenance activities. The augmented inspection team noted that the other sump cross-connect valves in the high pressure core spray/control rod drive and RHR A/RCIC sumps were also not in the scope of the maintenance rule or inservice testing programs and did not receive periodic maintenance or testing. The issues associated with the periodic maintenance and testing of the sump isolation valves will remain open pending further NRC review (50-397/9816-22).

The operators indicated that the sump pump for the RHR C pump room operated for a significant duration during the event even though it had been completely submerged. However, due to the limited capacity of the pump, it was not significant in mitigating the consequences of the flooding.

2.5.4 Effects of Flooding on Plant Equipment

Equipment Design Considerations

The augmented inspection team reviewed the licensee's criteria for the qualification of safety-related equipment in the flooded areas. The licensee's UFSAR, Section 3.11.1.2.2, stated that flooding was not a required environmental service condition for equipment qualification in the secondary containment except for the safety-related pump rooms. However, the licensee provided information from Document EQES-2, Revision 6, "Technical Requirements for Electrical Equipment Environmental Qualification," Section 4.13, which stated that flooding was not a required environmental service condition outside containment above selected heights in the LPCS pump room and RHR C pump room. Section 4.1.3 of the UFSAR specified that the keepfill pumps for the LPCS and RHR pumps were below the selected levels in the LPCS and RHR C pump room, but concluded that these pump motors did not need to be

watertight because operator action would compensate for the loss of these pumps during flooding scenarios. UFSAR, Section 4.1.3, further indicated that the site flooding analysis showed that WNP-2 could be safely shut down using alternate safety-related equipment, which was not affected by flooding. Thus, the licensee had concluded that the electrical equipment in a potentially flooded room did not need to be qualified for flooding.

Major Equipment Affected by the Flooding

The RHR C and LPCS pump rooms along with the lower elevations of the reactor building northeast stairwell were the primary areas affected by the flooding event. Water in the stairwell rose to approximately the 441-foot elevation. This equated to about 19 feet of water above the 422-foot elevation of the pump room floors. The stairwell contained both normal and emergency lighting fixtures, but no other significant plant equipment. The weight of the water above the bottom of the stairwell destroyed the fire door at the base of the stairwell.

The water in the RHR C pump room rose to a level which completely submerged the pump motor. Thus, essentially all equipment, cables, conduits, switches, etc., in that room was affected by the flooding. As previously indicated, the RHR C keepfill pump (RHR-P-3), which also serves the B train of the RHR system was submerged by the flooding. The keepfill pump was running at the time of the event and tripped off shortly after becoming submerged. Additionally, the pump for the RHR C pump room sump (FDR-SUMP-R4) was submerged by the flooding and eventually tripped. In addition to the RHR C components, the room contains cable trays which serve various equipment loads throughout the plant, which were also submerged by the flooding event. During the flooding event a ground fault on Battery E-B1-2 (Division II) was received in the main control room. Since the design of the system is such that the safety-related battery buses are ungrounded, this would allow the supplied equipment to operate normally with a single ground. A review of the event did not indicate any Division II 125 V dc faults or failures of equipment to operate. The licensee determined that the only Division II 125 V dc equipment that was affected by the flooding was Pressure Switches RHR-PS-16C and 19C. These pressure switches provided safety-related indication of RHR C pump flow to the ADS circuits. The licensee found water inside the terminal box for these switches and determined that these circuits were the source of the ground.

The LPCS pump room was flooded up to an elevation just below the pump motor. The primary components, which were submerged, included the LPCS minimum flow valve and the LPCS keepfill pump (LPCS-P-2), which also serves the RHR A train. The keepfill pump was running at the time of the event, but was secured by the plant operators prior to being submerged. Additionally, the LPCS pump room contains ADS permissive pressure switches similar to those in the RHR C pump room.

Recovery of the Flood Affected Areas

The augmented inspection team reviewed the licensee's inspection results of the flooded rooms and noted that almost all of the electrical equipment in the rooms had been

internally wetted. The initial cleanup efforts began shortly after the affected areas had been dewatered and continued over the next several days. However, on June 20, 1998, the augmented inspection team observed that the licensee did not have a formal overall plan to ensure that all of the flooded equipment was restored and evaluated for continued operation. The augmented inspection team discussed this observation with licensee management who indicated that they had previously noted the lack of overall planning and were initiating a restoration plan with comprehensively defined responsibilities.

The licensee initially used their master equipment list (MEL) and room drawings to develop a list of all equipment in the flooded area. However, early in the cleanup, the licensee determined that this list was not of sufficient detail to ensure the proper restoration of all the flooded equipment. For example, numerous conduit covers, a source of potential leakage into electrical equipment, were not shown on the MEL or room electrical drawings. On June 21, 1998, the licensee initiated a process to control the restoration effort using a tagging system. As equipment was opened for inspection, the information would be documented on a work order. Equipment, which was found to be internally dry would receive a green tag while equipment found wet, or with insufficient access for proper inspection, would be red tagged. The tags would include the associated work order to ensure that proper followup activities would be conducted. When an item was satisfactorily restored or replaced, this information would be documented on the work order and a green tag would be added over the red tag, leaving the red tag visible for reference. After restoration was complete, the licensee planned to perform detailed walkdowns to ensure all potentially flooded equipment had been affixed with a green tag.

During the course of the inspection, the licensee was in the process of developing final restoration criteria for the affected equipment. Based on discussions with personnel, reviews of licensee documents, and observation of work in the flooded areas the augmented inspection team determined that the licensee was using the following guidelines for establishing system and component restoration:

Mechanical — All supports and hangers were being cleaned and green tagged. All five snubbers, which were wetted, were replaced. All safety-related piping was being cleaned. All lagging (pipe insulation) was removed, piping was cleaned, and new lagging was installed. All piping and valves were being externally cleaned.

Components — Most electrical components, including all the safety-related components, were being replaced. These components included pressure switches, circuit breakers, and position switches. Motors were being removed for refurbishment. The licensee stated that they intended to replace any wetted electrical components with internal moving parts or parts that could not be adequately cleaned.

Cables — All conduit pull and inspection boxes were being opened for inspection. The licensee's preliminary review indicated that cables in the wetted areas

had been purchased to specifications, which included submergence. The licensee stated that they planned to dry all of the affected conduit but not replace wetted cables except as warranted where cable ends were wetted within components or connection boxes.

Air Systems The licensee stated that air systems would be opened and inspected and any damaged components replaced.

The licensee provided the augmented inspection team with the results of chemical tests done on the water pumped from the flooded areas to the T-4 sump. These results indicated that there were acceptable levels of salts in the water. Additionally, it was determined that the water had a ph of 7.5. On the basis of these results, the licensee concluded that no adverse chemical effects would be experienced by piping or other mechanical components in the flood affected areas.

2.6 Licensee Root Cause Assessment Methodology

On June 19, 1998, the licensee initiated an incident review board (IRB) to preserve information for use in determining the cause of the event. After beginning the process of preserving information, the IRB was disbanded on June 20, 1998, and the licensee assembled a formal root-cause team in lieu of the IRB process. The interim results of the IRB were provided to the root-cause team for further review. The root-cause team consisted of licensee management, engineering, and quality services personnel and was supplemented by representatives from other industry and private organizations who were experienced in formal root-cause investigation techniques. A formal root-cause investigation charter was approved on June 21, 1998, and an investigation plan to implement the charter was approved on June 22, 1998.

The augmented inspection team reviewed the root-cause charter and associated investigation plan and observed the initial workings of the root-cause team. The licensee's root-cause charter stated that the scope of the effort was to provide cause(s) and contributing factors associated with flooding of the RHR C and LPCS pump rooms, provide recommendations to prevent recurrence; and explore the design adequacy of flooding controls, work practices, configuration management, human and equipment performance, and equipment maintenance. The charter indicated that a formal and systematic root-cause investigation would be conducted using state-of-the-art methods, which have been successfully used in the nuclear industry. Additionally, the investigative plan required establishing and validating event causal factors to prevent flooding of the RHR C and LPCS pump rooms and also for addressing the water hammer to the fire protection system piping. The plan listed the major items to be reviewed and addressed contingencies for addressing any new issues which might be identified. Additionally, the plan required a review of human and equipment performance, as well as, management oversight.

In an effort which was independent of the primary root-cause team, engineering personnel assembled two specialized teams to address engineering issues that were associated with the event. These teams were assigned to perform an event analysis, as well as, identify and review issues related to the design and operation of the fire protection system. The licensee's overall root-cause plan directed that the results from these two teams be provided as an input to the primary root-cause team.

The augmented inspection team determined that the licensee had assigned personnel with previous root-cause experience and had obtained a level of industry support which should be sufficient to address the issues identified in the root-cause charter and associated investigation plan. The augmented inspection team observed that the root-cause plan addressed all the significant findings identified by the licensee and issues raised by the team. Further, the augmented inspection team determined that the licensee's planned root-cause methodology was appropriate to support root-cause analysis and that the plans were sufficiently broad and considered management expectations, process variables, and human performance issues.

Preliminary Review of the Licensee's Final Root Cause Analysis Results

On June 30, 1998, the licensee provided the NRC with the final results of the root-cause investigation. As noted earlier, the licensee determined that the root cause of the flooding event was that the plant fire protection system was of an inadequate design. It was determined by the licensee that the system was configured such that destructive forces would be generated during anticipated challenges to the system when it was operating in a normal lineup. With respect to the flooding, the licensee determined that the plant staff's sensitivity to the operation of the watertight doors was insufficient and that the material condition of the doors and the sump cross-connect isolation valve were factors that contributed to the seriousness of the event.

While the licensee's analysis appeared to address the major root cause of the event, the augmented inspection team determined that certain important considerations appeared to have been omitted from the RCA report. In particular, the following specific observations were noted:

- The RCA did not address the issue of potential simultaneous preactions of System 66 and System 81, which may have occurred in the past. This is significant in that the deficiency tag which was posted in the control room for System 81 immediately after the event indicated that this may have occurred on three separate occasions. Additionally, the facility seems to have had some experience with sympathetic preactions in that many of the preaction stations have warning placards attached to the equipment that advises plant operators to bypass other specific preaction stations during testing to avoid inadvertent actuations.

- The RCA omitted the results of the metallurgical analysis, which was performed on the ruptured valve.

- The analysis did not give comprehensive consideration to the failure of the sump cross-connect valve (FDR-V-609). While it was acknowledged that adequate maintenance has not been performed on the valve, no failure modes and effects analysis (FMEA) was developed to explore whether the valve design itself is adequate. Additionally, the RCA did not address the adequacy of a single valve design in lieu of a double-valve, single failure proof arrangement. This is particularly significant in that discrepancies are known to exist in the licensing and design documents regarding this issue.

- The analysis gave only cursory treatment to the adequacy of the single door design leading into the ECCS pump rooms, which were affected by the flooding. The RCA indicated that the design was accepted by the NRC (as a result of an earlier 10 CFR 50.55e issue). However, as with the sump cross-connect valve, discrepancies exist with respect to the as built design versus the design documentation. Additionally, previous licensee engineering studies have also questioned the adequacy of the flood protection afforded by the doors.

- The issue of the watertight doors' performance with respect to sealing from inside the pump rooms was not addressed in the RCA. Even though the licensee stipulates that the doors will not seal from the stairwell side, the flooding analysis assumed that no leakage would occur from the "qualified" side of the door. This event illustrated that the doors do not seal completely from the watertight side. (This raises additional questions regarding the assumptions that leakage will be minimized from the non-watertight side.)

- Even though it was briefly discussed in the analysis, the possibility that the RHR C pump room door was somehow opened by floating debris during the event was basically discounted. Thus, no corrective actions appear to have been developed to address this possibility. The stairwell containing the doors had a significant amount of miscellaneous materials stored in the vicinity of the doors. In fact, the as-found condition of the door showed a large tool box, which had floated into the door frame and would have prevented the door from closing. Given that the toolbox was found obstructing the door, corrective actions regarding material storage in the area appear to be warranted.

- While the RCA concluded that the fire protection system was inadequately designed, it did not specifically address the issue of the simultaneous starting of the three main fire pumps. Discussions with licensee representatives on July 2, 1998, indicated that the magnitude of the water hammer may have been exacerbated by the fact that three pumps started concurrently as opposed to starting in a staggered, time-delay sequence.

3 Summary of Augmented Inspection Team Findings

As noted in earlier sections of this report, the augmented inspection team was chartered as a fact finding activity. Thus, the findings of the augmented inspection team have not been developed with the intent of providing a final regulatory position with respect to the

issues which were identified. However, certain augmented inspection team findings may have regulatory implications. The final dispositioning of these issues will be accomplished via detailed followup inspection activities, which will further develop the issues and determine whether significant weaknesses or instances of noncompliance caused or contributed to the event. This report section provides a brief summary of the more significant augmented inspection team issues which warrant further NRC inspection. Each of the individual issues discussed in the following paragraphs are considered to be NRC inspection followup items (IFI).

Design Issues

- The watertight doors on RHR C, LPCS, RCIC, RHR A pump rooms are not watertight from the stairwell side. The licensee's analysis for stairwell flooding assumed that the doors will leak at approximately 70 gpm. The UFSAR and NRC SER indicated that the ECCS pump rooms are watertight. Thus, discrepancies appear to exist between the as-built configuration and the design and licensing documents (50-397/9816-15, Sections 2.5, 2.5.2).

- Flood protection and isolation via the sumps between the RHR-C/LPCS, RHR-A/RCIC, and CRD/HPCS pump rooms are via a single, nonsafety-related, air-operated valve. Although some inconsistencies in the UFSAR exist, some design information suggests that two-valve isolation exists and that the design is single-failure proof. The NRC's SER provides a contradictory view of the configuration. Thus, discrepancies exist between the as-built configuration and the design and licensing documents (50-397/9816-20, Sections 2.5, 2.5.3).

- The fire protection system appears to be highly susceptible to water hammer (especially during preaction scenarios). The system may not tolerate multiple, simultaneous preactions. Many plant preaction stations have warning placards affixed which indicate that the potential for multiple preactions exists during system testing (50-397/9816-14, Sections 2.2.1, 2.5.1, 2.6).

- Previous licensee analyses suggest that unchecked flooding in the northeast stairwell could potentially affect all ECCS systems (50-397/9816-09, Sections 2.5, 2.5.2).

Maintenance Issues

- Valve FDR-V-609 (the RHR C to LPCS sump cross-connect valve) was known to be degraded at the time of the event (since January 1998). No corrective actions had been implemented to address the deficiency prior to the flooding event in which the valve failed to perform its intended function (50-397/9816-21, Section 2.5.3).

- RHR-2C door was known by plant personnel to have a very light, free-wheeling closing mechanism. This particular maintenance deficiency had not been

documented in the licensee's work control or corrective actions process (50-397/9816-18, Section 2.5.2).

- Valve FDR-V-609 (the RHR C to LPCS sump cross-connect valve) was not in the scope of the licensee's maintenance rule or inservice testing programs. No periodic preventive maintenance or testing had been performed on the valve in the past several years. The last known systematic preventive maintenance was performed in 1995. This same consideration applies to the other sump cross-connect valves in the HPCS/CRD and RHR-A/RCIC sumps (50-397/9816-22, Section 2.5.3).

- The ECCS pump room watertight doors were in the scope of the licensee's maintenance rule program. However, the monitoring and performance criteria, which have been established, do not appear to address the function of the door from the perspective of mitigating the effects of stairwell flooding. The performance monitoring established by the licensee only addressed the function of the doors with respect to sealing from floods emanating from inside the pump rooms (50-397/9816-19, Section 2.5.2).

Human Performance Issues

- Improper coordination and control appeared to exist between the TSC and the control room in the early stages of the flooding event. The TSC authorized the discharge of potentially contaminated floodwater to the storm drain's system without authority and without the knowledge and consent of the control room (50-397/9816-07, Section 2.3.2).

- The decision to discharge to the storm drain system did not appropriately consider all relevant information relative to sampling. The sampling methodology employed after the decision to discharge to the storm drain drains system did not ensure a sufficient sample size and did not ensure that a thoroughly representative sample was drawn for analysis (50-397/9816-08, Section 2.3.2).

- The RHR C door appears to have been left in an unsecured condition prior to the event. The secondary alarm station printout indicated that the door was opened (and left open) prior to the event (50-397/9816-16, Section 2.5.2).

- Water hammer events have previously occurred in the fire protection system at WNP-2. Corrective actions for these events have not prevented recurrence (50-397/9816-11, Section 2.5.1).

- The flooding scenario which occurred on June 17, 1998 was previously recognized and reported in LER 92-034-02 (June 24, 1993) and also evaluated in plant engineering assessment SS2-PE-0591. The corrective actions to address this issue and the assumptions contained in these analyses appear to be flawed (50-397/9816-10, Section 2.5).

- The operating crew's decision to start the LPCS pump following the receipt of the LPCS pump room flooding alarm was not in accordance with the abnormal conditions procedure which was in effect at the time of the event. However, plant management indicated that the operators' actions were in accordance with management expectations (50-397/9816-03, Section 2.3.1).

- All the required fire protection compensatory measures were not implemented following the disabling of the fire protection system. Most of the measures which were not implemented were of an administrative nature; however, fire watches were not posted in all the required areas (50-397/9816-04, Section 2.3.1).

Procedural Issues

- The procedure for the control of ignition sources, which was being used at the time of the event, may not provide adequate guidance to prevent unwanted fire protection system preactions (50-397/9816-05, Section 2.3.1).

- Abnormal Conditions Procedure 4.12.4.10, "RB 422 Area Flooding," provides weak guidance regarding flooding in the RHR C/LPCS pump rooms. The guidance suggests that operators "consider" starting the RHR A/RHR B pumps during flooding events in these rooms. However, the licensee's flood analysis assumed that definitive operator actions to start the pumps would be taken within a timely manner (50-397/9816-01, Section 2.3.1).

Radiological Sampling Issues

- Due to the small sample size taken during the discharge to the storm drain system, the analytical limits of the licensee's sampling methodology were not sufficient to ensure that the concentration of Co-60 was less than 1.5E-8 µCi/ml. However, it was noted that within the limits of resolution of the analysis which was performed, no detectable activity was discharged. Subsequently, the licensee submitted the sample to an offsite laboratory with enhanced analytical capabilities. The results of the offsite analysis showed that the sample contained less than 1.5E-8 µCi/ml (50-397/9816-06, Section 2.3.2).

Equipment Performance Issues

- The preaction of fire protection system 66 appeared to be a valid response to the smoke generated from the cutting and grinding activities which were being conducted in the diesel generator room. However, the preaction of fire protection System 81 appears to be an unexpected system response (50-397/9816-12, Section 2.5.1).

- The radio communications (using portable hand-held radios) between plant operators were degraded due to poor radio performance (50-397/9816-02, Section 2.3.1).

- Previous preactions of System 81 may have occurred coincident with System 66 in the past. (i.e., a control room deficiency tag suggested that three previous simultaneous preactions may have occurred) However, no definitive record of this deficiency or corrective actions to preclude recurrence had been identified (50-397/9816-13, Section 2.5.1).

- Leakage occurred through the ECCS pump room watertight doors in the RHR C and LPCS pump rooms in the direction from which the doors are designed to be watertight, (i.e., water leaked OUT of the watertight doors). The licensee's flooding analysis assumed zero leakage through the doors in the watertight direction (50-397/9816-17, Sections 2.5, 2.5.2).

4 Management Meetings

An entrance meeting was conducted with the licensee on June 19, 1998 in which the NRC discussed the purpose and scope of the inspection. Immediately following the meeting, the licensee was provided with a copy of the augmented inspection team's charter for the inspection. A detailed debriefing of the team's preliminary findings was conducted with licensee staff and management on June 23, 1998. Several conference calls were conducted between the NRC and licensee management throughout the event and subsequent recovery period. A public meeting between the NRC and the licensee was conducted in the NRC's Arlington, Texas office on July 2, 1998. The NRC conducted inspection activities associated with the licensee's immediate corrective actions and compensatory measures prior to restart. The results of these inspection activities will be documented in a separate NRC inspection report . At that meeting, the licensee presented the results of the final root-cause analysis and described the actions taken to recover from the event and the compensatory measures that had been taken prior to restarting the unit. An exit meeting and press conference were conducted on July 8, 1998. The licensee acknowledged the findings, which were presented and indicated that no proprietary material was provided to the NRC inspection team.

ATTACHMENT 1

AUGMENTED INSPECTION TEAM CHARTER