GROUP A

FOIA/PA NO: ______2012-0235_______

RECORDS BEING RELEASED IN THEIR ENTIRETY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

MEMORANDUM TO: Thomas B. Blount
Deputy Director
Division of Reactor Safety

Ray L. Kellar
Enforcement Officer

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Jeff Clark, Chief
Projects Branch F
Division of Reactor Projects

FROM: Elmo E. Collins
Regional Administrator

SUBJECT: CHARTER FOR BACKFIT PANEL ON POSTULATED FAILURE OF UPSTREAM DAMS AFFECTING FORT CALHOUN STATION

In response to a proposal from David Loveless, dated July 1, 2011, to evaluate a backfit exception for Fort Calhoun Station, a backfit panel is being convened in accordance with Regional Office Policy Guide 0901.6, "Facility-Specific Backfit and Information Collection Procedure. You are hereby designated as the Panel Chairman. Ray Kellar, Tom Farnholtz, and Jeff Clark are designated as panel members. Additionally, the technical liaison assigned to support the panel is David Loveless and the Office of General Counsel will support the panel from a backfit process perspective.

A. Basis

Management Directive 8.4, "Management of Facility-specific Backfitting and Information Collection," states that NRC staff shall be responsible for identifying proposed facility-specific backfits. By memorandum from David P. Loveless, Senior Reactor Analyst to me, dated July 1, 2011, Mr. Loveless identified a proposed adequate protection backfit exception for Fort Calhoun Station. The specific issue relates to the licensee's preparedness and capability to protect the reactor core and spent fuel onsite from the consequences of the total failure of any of the six upstream dams on the Missouri River.

A1

B. Scope

The panel is expected to address the following:

1. Review the analysis performed by Mr. Loveless to determine if an information request in accordance with 10 CFR 50.54(f) is justified and whether the burden imposed on the licensee by the information request is justified in view of the potential safety significance of the issue.

2. Determine if there is reason to believe that the public health and safety may not be adequately protected at the Fort Calhoun Station with respect to total failure of any of the six upstream dams.

3. If the panel determines under Item 1 that the public health and safety is adequately protected at the Fort Calhoun Station and under Item 2 that a request in accordance with 10 CFR 50.54(f) is not justified, prepare a memorandum to me discussing the reasons for your conclusions. This memorandum should include an evaluation of each of the pertinent assumptions from the analysis performed by Mr. Loveless.

4. If the panel makes either determination in Items 1 or 2 above, prepare a draft 50.54(f) letter to require Omaha Public Power District to provide additional safety information to enable the Commission to determine whether or not their Fort Calhoun Station operating license should be modified, suspended, or revoked.

5. Develop a final evaluation to demonstrate that the information request is necessary including at least the following elements:

   a. A statement of the problem describing the need for the requested information in terms of its potential benefit;
   b. The licensee actions required and an estimate of the burden on the licensee to develop a response to the information request; and
   c. An anticipated schedule for NRC to use the information.

6. The draft letter in Item 4 should be prepared for my signature and should request specific information related to OPPD's preparedness and capabilities to respond to a total failure of any of the six upstream dams in addition to the bases of the evaluation performed in Item 5.

7. Upon completion of Items 1–2, and 4-6, the panel shall provide me a briefing to discuss the panel's findings, the information to be requested from the licensee, and the perceived benefits of acquiring such information.

8. Following receipt of the licensee's response, the panel will reconvene to review the provided information and prepare a draft addendum to this charter recommending the subsequent actions to be taken by the Region.

C. Guidance

10 CFR 50.109, "Backfitting," describes the methods available to the Commission to require that licensees modify or add structures, components, or design of a facility. Section (a)(4)(ii) states that a backfit analysis is not required where the staff finds and declares with an appropriately documented evaluation, that regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public.

10 CFR 50.54(f) permits the Commission to request a licensee submit under oath or affirmation, to enable the Commission to determine whether or not the license should be modified, suspended, or revoked. If this information is not sought to verify licensee compliance with the current licensing basis for that facility, the NRC must prepare the reason for each information request.

Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection, states that information requests to power reactor licensees, made pursuant to 10 CFR 50.54(f), shall be justified in view of the potential safety significance of the issue for which the information is requested.

PG 0901.6, "Facility-Specific Backfit and Information Collection Procedure," states that NRC staff positions may be identified as potential backfits by the staff. When the staff invokes a backfit exception, the Regional Administrator must provide a documented evaluation that includes a statement of the objectives, reasons for the modification, and the basis for the backfit exception.

This charter may be modified should the panel develop significant new information that warrants review. If you have any questions concerning this guidance, contact Anton Vegel, Director, Division of Reactor Safety, at (817) 860-8180.

ADAMS	☒Yes ☐ No	☑ SUNSI Rev Complete	Reviewer Initials:	DPL
Publicly Avail	☐ Yes ☒ No	Sensitive Value:☒		

DRS/SRA	DRS/EB1/BC	DRP/DD	DRS/DD	DRS/D	OGC/RMR	RIV/RA
DLoveless	TFarnholtz	J.Clark	TBlount	AVegel	BJones	ECollins
/RA/	**/RA/**	**/RA/**	**/RA/**	**/RA/**		
9/15/11	9/15/11	9/23/11	9/23/11	9/23/11		

NEAR TERM STATION PRIORITIES

TASK	LEAD WORK GROUP	EXPECTED COMPLETION DATE	COMMENTS
Evaluate spare breaker in the warehouse for use in MCC-4A2	DEN	July 20	Need to determine timely success path for MCC-4A2.
Confirm date of delivery of breaker for MCC-4A2 temp mod.	Procurement Engineering	July 20	The breaker did not arrive and a delivery date needs to be determined and temp mod schedule confirmed.
Install Temp. Mod. for MCC-4A2	EM	July 19	*Date is in jeopardy. EC 53319 Parts hold. Breaker WO 417046
Transport 16 cases of paper to Admin Building.	Maint	July 19	Paper is to arrive Tuesday and needs to be put in Admin Building. Contact Cathy Geren x7241.
Maintenance Bldg Support Column	DEN	July 20-PRC Approval of TM July 29 Installed	Temp Mod 53436 to install 6 shoring columns. EC 50986 for permanent repair during site recovery.
Perform Operability evaluation of Stainless Steel sample vessel that dropped into FO-1	DEN	July 21	CR 2011-6050-8
VA-89/90: Sandbag/berm protection will be installed for immediate protection (2-3 Days); Remove, restore and elevate contingency (10-14 days after EC issued).	SFM	July 22	EC 53400 being processed by DEN for the elevated stands. Issue EC as contingency action.
Jim Shores needs EM support for pulling new phone cable into the Security Building	Comm/EM	July 27	WR 166809/WO 417997; Cable has arrived and staged in security bldg.

FLOOD BARRIER [illegible]

Flood barriers installed per PE-RR-AE-1001, with the following deviations:
AE-22- 1007-1 door to Radwaste Building (power door) is removed. NOTE: NO POWER from MCC-4A2.
AE-23- 1007-9 door from RCA to Chemistry is removed.
AE-24- 1007-19 door – normal RP access point is removed.
AE-25- 1011-1 door from Turbine Mezzanine to Coordinator 52 is removed.
AE-28-1011-4 south door switchgear room for emergency switchgear exit
AE-21- Intake structure "trash rack" outlet is removed for cell cleaning work.
SD-127, 128 – Maintenance Shop drains to Lift Station #1 - open
VD-681, 682 – Switchgear Room HVAC drains to TB Sump - open
Other 1007' barriers can be removed to allow work access with authorization from SM. All 1007' barriers will be re-installed when river level reaches 1006' 9" and rising, or when directed by the SM.

ACTIVE HITS

- Level A CR 2011-5414: Loss of 1B4A – Owner: Steve Clayton. RCA in progress.
- Intake Screen sanding in issues – HIT formed under CR 2011-5750 – Ken Erdman lead.

A

NOTES: (Topics to Discuss, Actions to Complete, Kudos to Share)

- Spare B.5.b Fire Truck from Fort Calhoun Volunteer Fire Dept located onsite. Supply hose in rack along walkway to King Tut blocks
- Bottled water is stored in a sealand container at the top of the hill. Bring into PA as needed.
- Shift Manager and IC will continue to monitor river levels and utilize AOP-1, EPIP-TSC-2 and PE-RR-AE-1001 as the principle governing documents. NOTE: New Revision to EPIP-TSC-2 issued on 7-01-11 for alternate filling of D/G fuel oil.
- If communications or ERDS is lost in the Control Room or TSC; Incident Commander is to call NRC Senior Resident Inspector and the NRC operations center (per AOP).
- Use controls in place for boat safety – Need supervisor brief before getting boat keys from Incident Commander.
- Spare gas generators are staged on the Turbine Deck. Spare gas pumps are in the Maintenance Shop.
- The current National Weather Service prediction for Blair river level is a rise of about 5" between now and Thursday night (7/21). Based on trending as the river went up, we may not see this full rise at Fort Calhoun but should expect some river level rise during the week.

Mr. David J. Bannister
Vice President and CNO
Omaha Public Power District
Fort Calhoun Station
444 South 16th St. Mall
Omaha, NE 68102-2247

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 – REQUEST FOR ADDITIONAL INFORMATION RE: PROPOSED CHANGE TO ESTABLISH THE ULTIMATE HEAT (UHS) SINK LIMITING CONDITION FOR OPERATION AND ADDITION OF UHS LEVEL AND TEMPERATURE SURVEILLANCE REQUIREMENTS (TAC NO. ME5830)

Dear Mr. Bannister:

By letter dated March 4, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110680093), Omaha Public Power District (the licensee) requested an amendment to Renewed Facility Operating License No. DPR-40 for Fort Calhoun Station, Unit 1. The proposed amendment would establish the limiting condition for operation (LCO) requirements for the ultimate heat sink (UHS) in Technical Specification (TS) 2.16, "River Level," and adds two new surveillance requirements (SRs) for UHS level and temperature to TS 3.2, "Equipment and Sampling Tests," Table 3-5, "Minimum Frequencies for Equipment Tests." Specifically, this proposed change revises the title of LCO 2.16 from "River Level" to "Ultimate Heat Sink (UHS)"; provides more explicit applicability for the LCO 2.16; removes the existing LCOs for river level in TS 2.16, items 1) and (2); and reformats TS LCO 2.16 to provide required actions for an inoperable UHS. In addition, two new SRs, items 25 and 26, will be added to TS 3.2, Table 3-5, to test the river level and temperature on a daily frequency and for consistency, the columns will be reformatted to allow adding the column header to Table 3-5 for items 17 through 24. The Table of Contents is also revised to reflect the title change of LCO 2.16.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in your application and determined that additional information is required in order to complete its review. A draft copy of the enclosed request for additional information was provided to Mr. Bill Hansher of your staff via e-mail on September 16, 2011. A public teleconference will be scheduled to discuss your responses to the NRC staff's questions.

If you have any questions, please contact me at 301-415-1377 or via e-mail at lynnea.wilkins@nrc.gov.

Sincerely,

Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure:
As stated

cc w/encl: Distribution via Listserv

If you have any questions, please contact me at 301-415-1377 or via e-mail at lynnea.wilkins@nrc.gov.

Sincerely,

Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure:
As stated

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ADAMS Accession No. ML112770314

OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DSS/SBPB/BC	NRR/DE/EMCB/BC
NAME	LWilkins	JBurkhardt	GCasto	MMurphy
DATE		10/5/11		

OFFICE	NRR/DSS/SCVB/BC	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM
NAME	RDennig	MMarkley	LWilkins
DATE			

OFFICIAL RECORD COPY

REQUEST FOR ADDITIONAL INFORMATION

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

By letter dated March 4, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110680093), Omaha Public Power District (OPPD, the licensee) requested an amendment to Renewed Facility Operating License No. DPR-40 for Fort Calhoun Station, Unit 1 (FCS). The proposed amendment would establish the limiting condition for operation (LCO) requirements for the ultimate heat sink (UHS) in Technical Specification (TS) 2.16, "River Level," and adds two new surveillance requirements (SRs) for UHS level and temperature to TS 3.2, "Equipment and Sampling Tests," Table 3-5, "Minimum Frequencies for Equipment Tests." Specifically, this proposed change revises the title of LCO 2.16 from "River Level" to "Ultimate Heat Sink (UHS)"; provides more explicit applicability for the LCO 2.16; removes the existing LCOs for river level in TS 2.16, items 1) and (2); and reformats TS LCO 2.16 to provide required actions for an inoperable UHS. In addition, two new SRs, items 25 and 26, will be added to TS 3.2, Table 3-5, to test the river level and temperature on a daily frequency and for consistency, the columns will be reformatted to allow adding the column header to Table 3-5 for items 17 through 24. The Table of Contents is also revised to reflect the title change of LCO 2.16.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information provided in the application and determined that the following additional information is required in order to complete its review.

1. FCS is committed to the draft General Design Criteria (GDC), which are contained in Appendix G, "Response to 70 Criteria," of the FCS Updated Safety Analysis Report (USAR), and are similar to the GDCs in Appendix A, "General Design Criteria for Nuclear Power Plants," in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. FCS Design Criterion 2, "Performance Standards," states, in part, that

> Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice and other local site effects.

NRC Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," requires, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena such as floods without loss of capability to perform their safety functions.

In its letter dated March 4, 2011, the licensee stated:

> The proposed amendment would establish the limiting condition for operation (LCO) requirements for the ultimate heat sink (UHS) in the Technical Specification (TS) 2.16, *River Level*, and adds two new surveillance requirements (SRs) for UHS level and temperature to TS 3.2...

Please provide the reason and justification for removing the high end (flood) of the Missouri River level from TS 2.16, and the new location of these requirements.

2. In its letter dated March 4, 2011, the licensee stated that flooding at FCS is highly unlikely and, therefore, requested to remove the provision in the TS requiring a plant shutdown if the Missouri River level reaches 1009 feet. The licensee stated that a high river level TS does not meet the requirements of 10 CFR 50.36, "Technical specifications."

   The regulations in 10 CFR 50.36(c)(2)(ii)(D) Criterion 4 states "a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." In addition, 10 CFR 50.36(c)(2)(ii)(B) requires the licensee to establish a TS limiting condition for a "operating restriction that is a initial condition of a design basis accident or transient analysis." Based upon recent operating experience, please describe flooding as a risk to the plant's ability to maintain safe shutdown capabilities.

   In addition, please provide the latest data on river level peaking for the Missouri River at the FCS location and explain if the latest data on river level supports the current high-level setpoint. Also, please describe how removing the high river level limit still provides assurance that the plant can safely shut down prior to losing equipment at the intake structure.

3. In the FCS Design Criterion 12, the licensee states that instrumentation conforms to the applicable Institute of Electrical and Electronics Engineers (IEEE) standards. The licensee is proposing to add a new TS surveillance to limit the UHS temperature to 90 °F. The licensee credits existing instrumentation to monitor river temperature. The licensee states that it will encompass the loop uncertainty for UHS temperature in the surveillance test. Also, the licensee proposes to add a TS surveillance to limit the UHS to a river level of 976 feet 9 inches. However, the licensee states its current river level instrumentation is not very accurate and intends on replacing the river level instrumentation with a more reliable system. The licensee also designates a TS surveillance limit for the river water maximum temperature and minimum level at the assumed limit used in the design basis calculations which shows sufficient heat removal in order to mitigate a design-basis accident.

Please provide the methods used for monitoring river water temperature and level to comply with the current licensing basis. Also, please explain how OPPD plans to capture the uncertainty in the instrumentation used for monitoring river temperature and level.

4. In its letter dated March 4, 2011, the licensee states:

> Missouri River level is currently monitored by non-safety related loop L-1900. The measurement technology is based on a bubbler system where air pressure is directly proportional to the depth of the bubbler outlet. The accuracy of this loop is very poor with an As Found/As Left loop tolerance of +/- 2 feet as documented in calibration procedure IC-CP-01-1900. This loop is in the process of being replaced with loop L-2000 via engineering change EC 35741. This instrumentation is expected to be placed in service during 2011.

Please provide the test data (accuracy) and documentation (industry experience) to support the EC 35741.

5. In its letter dated March 4, 2011, the licensee proposed a TS SR for the low level of 976 feet 9 inches of the UHS – Missouri River. The licensee states that a river level of 976 feet 9 inches is required for raw water pumps minimum submergence level.

The regulations in 10 CFR 50.36 require the licensee to establish LCOs, and when they are not met, the licensee shall shut down the reactor. The licensee proposes a TS limit on low river level at the same level where the raw water pumps become unavailable for use; however, the raw water pumps are required to shut down the reactor. Please describe how this TS change will affect the availability of raw water pumps to bring the plant to cold shutdown conditions.

6. Please verify that no change is being made to any containment safety analysis or any parameter that could affect the containment safety analysis as part of this license amendment request. If this license amendment request is resulting in a change to any containment safety analysis or any parameter that could affect the containment safety analysis, please describe and justify these changes.

7. Please state if the calculation for the component cooling water heat exchanger's heat removal capability to supply 160 degree return water with 90 degree river water has been reviewed by the NRC staff.

Balance of Plant Branch input into the Safety Evaluation for
Fort Calhoun Station Request for Changing Technical Specification 2.16

1.0 Introduction

By letter dated March 4, 2011, Omaha Public Power District, the licensee, submitted a request for changes to the Fort Calhoun Station (FCS) Technical Specifications (TS). The requested change establishes the limiting condition for operation (LCO) requirements for the ultimate heat sink (UHS). The licensee proposes to remove the current TS 2.16 titled "River Water", and add a TS 2.16 titled "Ultimate Heat Sink" with provisions that are more in line with NUREG 1432, Standard Technical Specification Combustion Engineering Plants. The licensee proposes to add a surveillance requirement to monitor the river water temperature not to exceed 90 degrees, and surveillance requirement to monitor the river level not to exceed a low level of 976 feet 9 inches. The staff notes that in the processes of replacing the current specification, the licensee proposes to delete the existing requirement to implement an emergency plan to protect the plant when the river rises to 1004.2 feet, and shut down the reactor when the river level rises above 1009 feet.

2.0 Background

Fort Calhoun Station is a single unit site, pressurized water reactor, designed by Combustion Engineering. The Corps of Engineers manages the Missouri River water level and flow rates using massive earthen upstream dams. The Missouri River serves as the UHS, supplying the cooling water for Fort Calhoun Station. Fort Calhoun Station uses river water for both its nonsafety-related circulating water system and its safety-related raw water system. The raw water system has four pumps installed in three separate bays in the intake structure. Two bays have one raw water pump, and one bay has two raw water pumps. Water level instrumentation in the intake structure provides the control room with indication alarms if the raw water pumps are in danger of flooding.

2.0 Regulatory Evaluation

The FCS received its construction permit on June 7, 1968. Since this is prior to the issuance of the General Design Criteria (GDC) of 10 CFR Part 50 Appendix A, these GDC are not applicable to the FCS. Appendix G to the FCS Updated Safety Analysis Report contains the applicable criteria for the FCS.

The FCS Design Criterion 52, "Containment Heat Removal Systems," states: Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided. The containment spray and the containment fan coolers would remain in operation.

CEOG STS describes the UHS as a complex of water sources and any necessary retaining structures, e.g. river and its associated dam. In addition, the UUS includes any canals that connect the source with the cooling water system intake structure. However, the UHS does not include the intake structure.

Standard Review Plan (SRP), NUREG-0800 Section 9.2.5, describes the UHS as an assured supply of water credited for dissipating reactor decay heat and essential station heat loads after a normal reactor shutdown, or shutdown following an accident or transient. The water source must be adequate to supply the required flow of cooling water at the appropriate temperatures for normal, accident, and shutdown conditions for at least 30 days.

Regulator Guide 1.27 describes methods acceptable for ensuring UHS can withstand the effects of natural phenomena

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions,

10 CFR 50 App A, GDC 19 as it relates to the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear plant safely under normal conditions and to maintain it in a safe condition under accident conditions,

10 CFR 50 App A, GDC 44 as it relates to cooling water systems transferring heat loads from safety-related structures, and systems to the heat sink under normal operating and accident conditions,

3.0 Technical Evaluation

3.1 Proposed Technical Specification Changes

3.1.1. TS 2.16 Applicability

Currently, TS 2.16 states the applicability

> Applied to Missouri River level as measured at the intake structure at the Fort Calhoun Station.

The licensee proposes changing the applicability of TS 2.16 to read:

> Applied to operational status of the Missouri River at the intake structure at the Fort Calhoun Station for reactor coolant temperature Tcold -> 210°F.

3.1.2 TS 2.16 River Low Level

Currently the TS for river level states,

> If the Missouri River level is less than 976 feet 9 inches the reactor will be placed in a cold shutdown condition using normal operating procedures. At river levels less than 980 feet a continuous watch will be maintained to assure no sudden loss of water supply occurs.

The licensee is proposing to define the operability of the UHS to read:

> The OPERABILITY of the UHS is based on having a minimum Missouri River water level of 976 feet 9 inches mean sea level and a maximum river water temperature of 90°F.

In the event one of the conditions cannot be satisfied, the licensee proposes the following actions:

If the Missouri River level is less than 976 feet 9 inches or river temperature is greater than 90°F, then the reactor will be placed in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 36 hours using normal operating procedures. At river levels less than 980 feet, a continuous watch will be maintained to assure no sudden loss of water supply occurs.

3.1.5 TS 2.16 River High Level

The licensee is proposing to delete the shutdown requirement upon high river level that read:

Specifications

1) If the Missouri River level exceeds 1009 feet the reactor will be placed in a cold shutdown condition using normal operating procedures. When the river level reaches elevation 1004.2 feet and rising, the emergency plan to protect the plant will be instituted.

3.1.6 TS 2.16 Surveillance Requirements

The licensee is proposing a TS surveillance performed daily to monitor the river temperature and level. The proposed surveillance will state

25. UHS-Missouri River Level	Verify water level of UHS is > 976 feet 9 inches above mean sea level.
26. UHS-Missouri River Temperature	Verify water temperature is < 90 F.

The licensee has proposed the TS changes to become more in line with CEOG STS in NUREG 1432. However, the licensee currently does not have the verbiage in CEOG STS surveillance requirement 3.0.1 to account for monitor the variable between surveillance periods. Therefore, the licensee must provide assurance that the limit will not be exceeded between surveillance periods. CEOG STS states, "Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO." The staff requests additional information from the licensee to meet the monitoring requirements.

3.1.7 Additional Applicable TS Requirement for UHS

Currently the licensee has operability requirements for the raw water pumps associated with river temperature in TS specification 2.4, Containment Cooling. Within TS 2.4, when the river temperature is below 60 degrees, one raw water pump may be inoperable indefinitely. The licensee proposes no changes to this provision.

4.0 Staff Evaluation

4.1 UHS Description

The Missouri River provides the ultimate heat sink (UHS). Corps of Engineers control releases from six upstream dams to manage the river water level normally at 983 feet during winter months, and at 992 feet during navigation periods. The nearest dam is Gavins Point, approximately three miles upstream of the FCS. The Corps of Engineers' annual operating plan states normal release from Gavins Point is 12,000 feet per second (cfs), depending largely on availability of water. Flows during the non-navigation season will range from 12,000 cfs to 18,000 cfs. In years when an extended period of drought has depleted storage reserves, release flows may run as low as 6,000 cubic cfs.

The UHS supplies water to the raw water system, an engineered safeguard system. In order for the raw water system to meet its safety function during a design basis accident, the UHS must supply water to the raw water pumps at a temperature 90 degrees or below, and at a river level of 979 feet 9 inches in order to maintain adequate net positive suction.

4.2.1 River High Level

Studies show the Missouri River water level would increase to 998 feet only 1% of the time. The momentary yearly flood peak with an expected 1% probability is set at 1001.3 feet. The design flood river level is conservatively set at 1006 feet, based upon a 0.1% probability. The Corps of Engineers considers a total failure of a large upstream dam not credible. However, in the event of an upstream credible fault of either the Fort Randal or Oahe dams coincident with runoff from maximum probable rain storms, the Corps of Engineers expect a peak of 1009.3 feet. From a study perform in 1995 by Federal Emergency Management Agency (FEMA), the 100-year flood level at the site is 1006.1 feet and the 500-year flood level is 1008.5 feet.

The original design basis from 1960's data, assumes credit for passive protection above the design flood peak stage of 1,006 feet. The concrete intake structure and elevation of openings provides permanent protection for the raw water pumps up to a river water level of 1007.5 feet. Without special provisions, the plant can accommodate flood levels of up to 1,007 feet.

The licensee can install flood gates to offer protection of the safety related SSC's against the maximum probable flood peak stage of 1009.3 feet. Without special provisions, the plant can accommodate flood levels of up to 1,007 feet. Steel flood gates are permanently mounted above and adjacent to openings in structures containing equipment required for a safe and orderly plant shutdown. In the event of high water levels, these flood gates and sandbags around the travel screens can be installed to provide protection to a level of 1,009.5 feet. Protection can be provided up to 1014.5 using sandbags around the traveling screens and supplementing intake structure perimeter walls with sandbags

In the USAR, Section XXX, the licensee states, "Because of the temporary nature of the emergency measures, the plant would normally be shut down when the flood water level exceeds the permanently installed protection provisions to minimize the likelihood of an accident."

The licensee proposes to delete the current TS requirement to place the reactor in a cold shutdown condition if the Missouri River level exceeds 1009 feet. When the river level reaches elevation 1004.2 feet and rising, the emergency plan to protect the plant will be instituted. In RAI XXXX, the staff requested the licensee provide justification for removing this requirement from TS.

4.2.2 River Low Level

The Corps of Engineers manages the river water level normally at 983 feet during winter months and at 992 feet during navigation periods. River level may run low during extended periods of drought when the Corps of Engineers reduce flow from the upstream dams. At low river levels, debris and/or ice on the traveling screens and/or trash racks can cause significant head loss potentially reducing intake cell levels below the raw water pump minimum submergence levels. The elevation of raw water pump suction bells is 973'-9"; however, the pumps require 3 feet of submergence above the bottom of the suction bell, elevation 976'-9". The circulating water pumps for the condenser have a much higher minimum submergence level (MSL) requirement (983 feet 0 inches) and would become unstable and trip or be manually shutdown well before intake cell levels decrease to the RW pump MSL.

As a precaution, whenever the river level goes below 980 feet, the licensee's procedures "AOP-1" establishes monitoring of the level of the river and the intake cells to ensure satisfactory pump submergence. In the event all raw water pumps become unavailable, operators can manual align the diesel or motor driven fire pumps XXXX. Or the reactor can be maintained at hot shutdown using auxiliary feedwater to the steam generators.

4.2.3 River Temperature

The current TS do not have any restriction on the river water temperature. The licensee proposes adding a TS surveillance requiring the river temperature remain below 90°F. In USAR, the licensee references calculations that show the river water system can meet the design requirements to mitigate a DBA if the river temperature is maintain below 90°F. The raw water is used as a cooling medium for the CCW heat exchangers. The licensee calculations show that 90°F river water to the CCW heat exchangers can remove sufficient heat from the CCW water in order to supply a 160°F maximum return temperature. With a maximum 160°F return temperature, the CCW system can provide its safety functions in the event of a DBA.

4.2.4 Instrumentation for Monitoring River Level and Temperature

The licensee proposes a TS limit on the river temperature of 90 degrees in order to satisfy the assumptions used in calculation for safety-related components to perform their design function. The licensee proposes to use temperature element TE-6251 and instrument loop T-6251 to monitor the river water temperature. The licensee proposes to encompass the uncertainty in the T-6251 instrument loop so that the temperature limit of 90°F is not exceeded.

The licensee proposes a TS limit on the low river level of 976 feet 9 inches in order to adequate suction for the raw water pumps to perform their safety function. The licensee is currently using a bubbler instrument with very poor accuracy measurement. Calibration procedures indicate an uncertainty of +/- two (2) feet. However, the licensee indicates that new level instrument will be installed in 2011 with greater accuracy.
[The staff request additional information on how the licensee will monitor the river level with current instrumentation and how they will incorporate the uncertainty of the new instrument.]

The licensee is requesting to remove from TS the requirement to shut the reactor down if the river exceeds a high level setpoint. The staff objects to the removal and requesting further justification. If the level stay in TS, the same instrument that measures low level will be used for monitoring high level??

Instrument channel uncertainties in these analyses are based upon the characteristics of installed instrumentation, the environmental conditions present at the instruments' installed locations, and process conditions. A properly established setpoint will initiate a plant protective action before the process parameter exceeds its analytical limit. This, in turn, ensures that the transient will be avoided and/or terminated before the process parameters exceed the established safety limits.

10 CFR 50 Appendix A, General Design Criterion (GDC) 13, "Instrumentation and Control," requires in part that instrumentation be provided to monitor variables and systems, and that controls be provided to maintain these variables and systems within prescribed operating ranges.

USAR Appendix G, CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges. This criterion is met. Instrumentation is provided for continuous measurement of all significant process variables. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation. The instrumentation conforms to applicable Institute of Electrical and Electronics Engineers (IEEE) standards.

4.5 ECCS Systems Interfacing with UHS

The following engineering safeguard systems interface with the UHS to provide a safety function; component cooling water system, containment spray system, containment air cooling and filtering system.

4.5.1 Raw Water System

The raw water system provides a cooling medium for the component cooling water system. In the event CCW pumps or heat exchangers are unavailable, raw water can circulated directly to vital engineered safeguards components. Raw water is credited as backup to CCW for fire events as described in Appendix R safe shutdown analysis. Raw water is on the seismic safe shutdown analysis equipment list for makeup to the emergency feedwater storage tank.

The intake structure houses four raw water pumps. Originally, the plant was designed with three raw water pumps, but later upgraded to four pumps to allow for continued operations with one pump out of service for maintenance. The elevation of raw water pump suction bells is 973 feet 9 inches. The pumps require 3 feet of submergence above the bottom of the suction bell, elevation 976 feet 9 inches. The design rating for the raw water pumps is 5325 gallons per minute (gpm) with 118 feet total dynamic head (TDH), at a design pressure of 150 psig and temperature of 150 degrees F. The pumps discharge into two separate 20-inch headers up into the auxiliary building. Each header provides sufficient flow to the component cooling heat exchangers to support normal modes of plant operation, and support design basis accident mitigation.

In the event of a complete failure of the component cooling system, raw water can used directly to cool certain engineered safeguards equipment:

- shutdown cooling heat exchangers for long-term decay heat removal after a large LOCA.

- control room A/C waterside economizers
- cooling of the safety injection and containment spray pumps, if locally accessible. They may be inaccessible post-RAS, however the pumps can perform their post-accident function without cooling water.
- Raw water to the containment air coolers is not required for containment peak pressure suppression.

Raw Water supplied directly to the containment cooling coils should be carefully implemented. It is not a postulated occurrence with any other event in progress. Calculation FC05662 concludes that the water will flash at the outlet of the cooling coils because the pressure will be less than 3.718 psia (saturation pressure of 150°F). Raw Water should not be used to cool the containment if the river level is less than 976 feet 9 inches; and may apply to any river level less than 983.5 feet. This condition will allow boiling to occur in the cooling coils causing water flashing, which may damage coils.

4.5.2 Containment Spray System.

In the event of a main steam line break, the containment spray system limits the containment pressure rise by spraying cool borated water throughout the containment atmosphere. The containment spray system can assist with long term cooling after a LOCA by recirculating water from the containment sump through the two shutdown cooling heat exchangers, transferring heat to the component cooling system. The cool water can be directed to safety injection pump suctions, or directly to the containment atmosphere. The containment spray system has three pumps, but only one pump is required to meet the design requirements for a DBA.

The containment spray system is capable of removing 280 x 10^6 Btu/hr (2 pumps) from the containment atmosphere at 288°F by spraying cool borated water from the 314,000 gallon SIRW tank. It also has the ability to recirculate spray water from the containment sump and through the shutdown cooling heat exchangers into the containment atmosphere. Under this mode of operation, the heat removal capability per heat exchanger is 87.5 x 10^6 Btu/hr based upon 4,000 gpm of cooling water at 114°F inlet temperature

4.5.3 Containment Air Cooling and Filtering System.

The containment air cooling and filtering system provides long term cooling of the containment atmosphere, and filters airborne activity. This system circulates the post-accident containment atmosphere through cooling coils, transferring heat to the component cooling water system. The system has four air handling units, but only two have a filtering capability. The two units with filtering capability are designed to remove 140 x 10^6 BTU/hr at DBA conditions. The other two units are designed to remove 70 x 10^6 BTU/hr at DBA conditions. A cooling water flow of 4,680 gpm with an inlet temperature of 120°F will remove 280 x 10^6 Btu/hr maximum with instrument air available, from a saturated containment atmosphere at 60 psig and 288°F, in order prevent containment from exceeding 60 psig design limit. In the event of the loss of normal power sources and the loss of an emergency diesel generator only two of the four units will be available, with only one having filtering capabilities.
[Hence, provide 140 +70 =210 x 106 BTU, which is less than required design 280]

4.5.4 Component Cooling Water System.

The CCW system provides cooling water to the shutdown heat exchangers, the containment air coolers, the spent fuel pool heat exchanger, the letdown heat exchanger, the blowdown sampling coolers, control room economizer coils and various pump coolers. The CCW system has three motor driven pumps. Each pump is rated at 3425 gpm. The CCW system has four heat exchangers, which transfer heat from the CCW system to the raw water system. Each heat exchanger is rated at 134 BTU/hr at DBA conditions. The four heat exchangers and two or the three pumps are sufficient to provide requirements for DBA loads. Even with a single failure and loss of IA, CCW can satisfy DBA load requirements.

"A contribution from the containment air cooler (cooled by CCW) is credited in the mitigation of containment peak pressure for a main steam line break."

"For a LOCA, the containment air coolers are credited for both mitigation of peak containment pressure and long term containment cooling."

The licensee calculates the CCW return temperature can be maintained below 160°F under DBA loads and conditions with the river water at 90°F. Normally CCW temperature is maintained between 55 to 110°F using a combination of the four CCW heat exchangers, the three CCW pumps and the four river water pumps. CCW is used during normal plant shutdowns to cool the reactor coolant water from 350°F to below 200°F for refueling operations. In the event of a safety injection actuation signal (SIAS) the CCW system components automatic start and align to deliver design basis flow to engineered safeguards components, and isolate non-critical components. In the event instrument air is not available during the DBA, those CCW components without accumulators will fail to their safe position, which may prevent isolation of CCW flow to non-essential components. In the event of a loss of normal power and a failure of one EDG to start, only one CCW pump will start automatically. A second CCW pump will be available for start on the swing bus. After the SIRW tank is depleted and a recirculation actuation signal (RAS) is generated, the valves supplying CCW to the shutdown heat exchangers will remain closed. This alignment ensures optimal flow to the containment air coolers, which is credited for long term containment cooling.

In the event of a CCW component failure rendering the CCW system inoperable, the raw water system can be manually aligned to flow through the CCW system piping to critical engineered safeguard components.

4.5.5 HVAC System

The control room HVAC is an essential auxiliary support system and designated an engineered safeguards system. The main control room HVAC system consists of two refrigeration and air handling units. The air conditioning units have hermetic compressors and air cooled condensers located on the auxiliary building roof. CCW is only used for the water side of the economizer, which is required for control room cooling when the ambient temperature is below 0°F. A ventilation isolation actuation signal (VIAS) will close the CCW inlet and outlet valves on the economizers.

The switch gear room also has two HVAC systems. The containment HVAC system maintains the temperature inside containment is maintained below 120°F during normal operations.

4.5.6 Shutdown Cooling Heat Exchanger

The shutdown cooling heat exchangers remove decay and sensible heat during normal and accident cooldowns. Once the reactor coolant system decreases below 300 psia, operator use the two low-pressure safety injection pumps to circulate the reactor coolant through two shutdown cooling heat exchangers, to reduce the temperature of the reactor coolant at a controlled rate from 350°F to a refueling temperature. The heat exchangers can remove 37.1 x 10^6 BTU/hr, based on 4500 gpm of CCW cooling water at 93°F inlet temperature and 3000 gpm of reactor coolant at 140°F inlet temperature.

In the event one diesel-generator fails, cooldown at a reduced but acceptable rate is possible with one heat exchanger and one pump. The shutdown cooling system also provides an emergency backup for the spent fuel pool cooling system in the event of failure of that system.

Following RAS, the containment spray pumps can cool the water from the containment sump by circulating the water through the shutdown cooling heat exchangers. Each of the shutdown heat exchangers can remove 58.9 x 10^6 BTUs based upon 2,937 gpm of cooling water at 95 F inlet, and 2,250 gpm spray flow at 212 F. Once the water exits the heat exchangers, the operator can direct the cooled water to the suction of the high-pressure safety injection pumps, which inject the subcooled water into the core. Under this mode of operation, the heat removal capability per heat exchanger is 87.5 x 10^6 Btu/hr based upon 4,000 gpm of cooling water at 114°F inlet temperature. The low-pressure safety injection pumps may also be used for long term cooling, if the reactor coolant pressure is sufficiently low.

4.5.7 Spent Fuel Pool

The system is designed to cool the pool water by recirculating the contents through the storage pool heat exchanger once every two hours with both pumps operating. Cooling water to the heat exchanger is provided by the CCW system. While the plant is shutdown, and the core is fully off loaded, the shutdown cooling system provides an emergency backup for the spent fuel pool cooling system in case of a failure.

4.6 Design Basis Accidents

In the event of a Design Basis Accidents (DBA), a safety injection signal (SIS) will start all 4 raw water pump. In the event only one emergency diesel generator (EDG) is available, only 2 raw water pumps will start. During a DBA, the containment air cooling system provides containment cooling by transferring heat to CCW system. The containment air cooling system includes four separate self-contained units. CCW flow of 4,680 gpm with an inlet temperature of 120°F is credited for removing up to 280 x 10^6 Btu/hr. The containment air cooling rate is sufficient to cool the containment air during normal operation and is credited to limit the pressure rise in the event of a design basis accident. The containment spray system can be aligned to take a suction on the containment sump and direct flow through the shutdown heat exchangers in order to provide an additional method to remove heat from containment.

Raw water system provides the cooling medium for the CCW system. The raw water system is credited to remove sufficient heat from CCW heat exchangers, so that the maximum return temperature does not exceed 160 degrees. Licensee's calculations support the raw water system's ability to remove sufficient heat up to a river temperature of 90°F. The raw water system is design to satisfy it design function with a single failure and loss of IA. Air accumulators are provided at the intake structure to operate valves upon loss of normal IA. When the river water temperature is below 60 degrees, only one pump is required to meet design basis heat removal.

The two DBAs that are most challenging for the UHS system are the main steam line break, and the loss of coolant accident, In the event of a main steam line break, the licensee states, "A contribution from the containment air cooler (cooled by CCW) is credited in the mitigation of containment peak pressure for a main steam line break." In the event of a loss of coolant accident, the licensee states, "For a LOCA, the containment air coolers are credited for both mitigation of peak containment pressure and long term containment cooling."

The river temperature limit exists to ensure that there is sufficient transfer of heat from the CCW system to the RW system during an event which is associated with a valid safety injection actuation signal (SIAS). Licensee Calculation FC07259, *FCS RW/CCW GOTHIC Model-Additional Cases,* used the integrated RW/CCW model developed in the **FCS** RSG Gothic Integrated Model calculation, FC06979, *FCS RSG - GOTHIC Integrated Model."* Calculation FC07259 determined the maximum RW inlet temperature achievable without exceeding a peak CCW temperature of 160°F at the inlet to the containment fan coolers. This maximum RW temperature is provided in calculation FC07259, Section 4.4.3, as 90°F.

[The NRC staff has not approved for general use the Gothic code used by the licensee. Therefore the Containment and Ventilation Branch (SCVB) needs to review the licensee's individual calculations.]

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 259 TO
RENEWED FACILITY OPERATING LICENSE NO. DPR-32
AND

1.0 INTRODUCTION

By letter dated March 4, 2011, Omaha Public Power District, the licensee, submitted a request for changes to the Fort Calhoun Station Technical Specifications. The requested change establishes the limiting condition for operation (LCO) requirements for the ultimate heat sink. The licensee proposes to remove the current TS 2.16 titled River Water, and add a TS 2.16 titled Ultimate Heat Sink with provision that are more in line with Standard Technical Specification Combustion Engineering Plants, NUREG 1432. The licensee proposes to add 1) a surveillance requirement to monitor the river water temperature not to exceed 90 degrees, 2) and surveillance requirement to monitor the river not to exceed a low level of 976 feet 9 inches.

The staff notes that in the processes of deleting the current River Water specification and replacing it with a Ultimate Heat Sink specification, the licensee is deleting the requirement to implement an emergency plan to protect the plant when the river rises to 1004.2 feet, and shut down the reactor when the river level rises above 1009 feet.

2.0 BACKGROUND

The Missouri River is the source of cooling water that services Fort Calhoun Nuclear Power Station. The Corps of Engineers manage the Missouri River water level and flow rates using massive earthen upstream dams. Fort Calhoun Nuclear Power station uses river water for its nonsafety-related circulating water system, and its safety-related raw water system. The river water intake structure is configured in three bays. Two bays have one raw water pump, and one bay has two raw water pumps. Water level instrumentation in the intake structure provides the control room with indication alarms if the raw water pumps are in danger of flooding.

Originally, the plant was designed with three raw water pumps, but later upgraded to four pumps to allow for continued operations with one pump out of service for maintenance. The design rating for the raw water pumps is 5325 gallons per minute (gpm) with 118 feet total dynamic(discharge, developed) head (TDH), at a design pressure of 150 psig and temperature of 150 degrees F. The pumps discharge into two separate 20" headers, which flow from the intake structure into the auxiliary building. Each header provides sufficient flow to the component cooling heat exchangers to support normal modes of plant operation, and support design basis accident mitigation.

The raw water system provides a cooling medium for the component cooling water system. In the event of a complete failure of the component cooling system, raw water can used directly to cool certain engineered safeguards equipment.

Raw water is also credited as backup to CCW for fire events as described in Appendix R, Safe Shutdown Analysis; and the Seismic Safe Shutdown Analysis equipment list for makeup to the Emergency Feedwater Storage Tank.

In the unlikely event of a complete failure of the component cooling water system, raw water direct cooling capability exists for the following equipment:

- Raw water direct cooling of the shutdown cooling heat exchangers would be needed for long-term decay heat removal after a large LOCA.
- Raw water direct cooling may be used for the control room A/C waterside economizers
- Raw water may be utilized for direct cooling of the safety injection and containment spray pumps, if locally accessible. They may be inaccessible post-RAS, however the pumps can perform their post-accident function without cooling water.
- Raw water direct cooling to the containment air coolers is not required for containment peak pressure suppression. Raw Water (RW) backup of the Component Cooling Water (CCW) being supplied to the Containment Cooling Coils should be carefully implemented. It is not a postulated occurrence with any other event in progress. Calculation FC05662 concludes that the water will flash at the outlet of the Cooling Coils because the pressure will be less than 3.718 PSIA (saturation pressure of 150°F), and Raw Water should not be used in any condition to cool the containment if the river level is less than 976'9" this may apply to any river level less than 983.5'. This condition will allow boiling to occur in the cooling coils causing water flashing, which may damage coils.

DBA

In the event of a DBA, upon a safety injection signal,SIS, all 4 pumps start automatically, all eight heat exchanger isolation valves open. In the event only one EDG is available, only 2 pumps will start.

When the river water temperature is below 60 degrees, only one pump is required to meet design basis heat removal.

Raw water system is design to remove sufficient heat from CCW so that the maximum return temperature does not exceed 160 degrees. The raw water system is design to satisfy it design function with a single failure and loss of IA. Air accumulators are provided at the intake structure to operate valves upon loss of normal IA.

3.0 REGULATORY EVALUATION

The regulatory requirements and the guidance used by the NRC staff for its review are:

Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, in particular

General Design Criterion (GDC 1) as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed,

GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions,

GDC 5 as it relates to the sharing of structures, systems, and components should not significantly impair the ability to perform their safety functions, GDC 16 as it relates to the containment and associated systems establishing a leak-tight barrier against the uncontrolled release of radioactivity to the environment and assuring that the containment design conditions important to safety are not exceeded for as long as the postulated accident requires,

GDC 19 as it relates to the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear plant safely under normal conditions and to maintain it in a safe condition under accident conditions,

GDC 38 as it relates to the containment heat removal system safety function which shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels,

GDC 44 as it relates to cooling water systems transferring heat from structures, systems, and components important to safety to an ultimate heat sink under normal operating and accident conditions,

GDC 50 as it relates to the containment heat removal system which shall be designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Standard Review Plan (SRP), NUREG-0800,

Section 6.2.1 "Containment Functional Design",
Section 6.2.1.1.A "PWR [Pressurized Water Reactors] Dry Containments, Including Subatmospheric Containments,"
Section 6.2.2 "Containment Heat Removal Systems,"
Section 9.4.1 "Control Room Area Ventilation System," and
Section 9.2.1 "Station Service Water System."

The NRC staff's review of the proposed change focused on the impact on the ability of the SR structures, systems, and components (SSCs) to perform their safety functions. Acceptability of the proposed change is judged based upon continued conformance with the plant licensing basis as reflected primarily in Updated Final Safety Analysis Report (UFSAR).

4.0 TECHNICAL EVALUATION

4.1 Proposed Technical Specification

4.1.1.

Currently, TS 2.16 states the applicability

> Applied to Missouri River level as measured at the intake structure at the Fort Calhoun Station.

The licensee is proposing to change the applicability of TS 2.16 to read:

> Applied to operational status of the Missouri River at the intake structure at the Fort Calhoun Station for reactor coolant temperature T_{cold} -> 210°F.

4.1.2

Currently the TS for river level state

> If the Missouri River level is less than 976 feet 9 inches the reactor will be placed in a cold shutdown condition using normal operating procedures. At river levels less than 980 feet a continuous watch will be maintained to assure no sudden loss of water supply occurs.

The licensee is proposing to define the operability of the UHS to read:

> The OPERABILITY of the UHS is based on having a minimum Missouri River water level of 976 feet 9 inches mean sea level and a maximum river water temperature of 90oF.

In the event one of the conditions cannot be satisfied, the licensee proposes the following actions:

> If the Missouri River level is less than 976 feet 9 inches or river temperature is greater than 90oF, then the reactor will be placed in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 36 hours using normal operating procedures. At river levels less than 980 feet, a continuous watch will be maintained to assure no sudden loss of water supply occurs.

4.1.3

The licensee is proposing to delete the shutdown requirement upon high river level that read:

Specifications

> 1) If the Missouri River level exceeds 1009 feet the reactor will be placed in a cold shutdown condition using normal operating procedures. When the river level reaches elevation 1004.2 feet and rising, the emergency plan to protect the plant will be instituted.

4.1.4

Currently the licensee has operability requirements for the raw water pumps associated with river temperature in TS specification 2.4, Containment Cooling. Within TS 2.4, when the river temperature is below 60 degrees, one raw water pump may be inoperable indefinitely. The licensee proposes no changes to this provision.

4.2 Staff Evaluation

4.2.1

Fort Calhoun Nuclear Power Plant is a single pressurized water reactor with two steam generators, designed by Combustion Engineering. The plant is located on the Missouri River. The condenser cooling water pumps and the raw water pumps are located in the intake structure.

The containment structure completely encloses the reactor coolant system. The prestressed, post tensioned concrete structure is designed for low leakage at a design pressure of 60 psig and 305°F.

The containment spray and the containment air cooling and filtering systems are part of the engineered safeguards systems.

The containment cooling system includes four separate self-contained units. The cooling coils and fans provide containment cooling at DBA conditions. The heat is transferred to the component cooling system. Cooling water flow of 4,680 gpm with an inlet temperature of 120°F will remove 280 x 10^6 Btu/hr to cool the containment air during normal operation and limit the pressure rise in the event of a design accident.

Following a LOCA, during the recirculation mode, the containment spray system can be aligned to the suction of the high pressure safety injection pumps to provide a source of subcooled water to supplement long term core cooling. The containment spray system has the ability to recirculate spray water from the containment sump and through the shutdown cooling heat exchangers into the containment atmosphere. Under this mode of operation, the heat removal capability per heat exchanger is 87.5 x 10^6 Btu/hr based upon 4,000 gpm of cooling water at 114°F inlet temperature..

Shutdown Cooling System The shutdown cooling system is used to reduce the temperature of the reactor coolant at a controlled rate from 350°F to a refueling temperature. It also provides an emergency backup for the spent fuel pool cooling system in the event of failure of that system. it also provides an emergency backup for the spent fuel pool cooling system in the event of failure of that system

Component Cooling and Raw Water Systems

The component cooling system provides cooling water to the shutdown heat exchangers, the containment air cooling system, the spent fuel pool heat exchanger, the letdown heat exchanger, the sampling heat exchanger, ventilation equipment and various pumps. Heat removed by the component cooling water system is transferred to the raw water system by the component cooling heat exchangers. The raw water system is a once through system operating with screened river water. Further, the system arrangement permits the raw water to be circulated through portions of the component cooling system piping to provide direct cooling of vital engineered safeguards components in the unlikely event of all of the component cooling pumps and heat exchangers being unavailable to fulfill their design function.

4.2.2 **Flood Protection**

UFSAR 2.7 states The original design basis, based on 1960s data, includes passive protection above the design flood peak stage of 1,006 feet msl due to the elevation of openings to safety related Structures, Systems and Components (SSC's) being 1007'.

In addition, installation of flood gates protects the safety related SSC's against the maximum probable flood peak stage of 1009.3'.

A conservative design flood elevation of 1,006 feet based on a 0.1 percent probability flood.

Without special provisions, the plant can accommodate flood levels of up to 1,007 msl.

Steel flood gates are permanently mounted above and adjacent to openings in structures containing equipment required for a safe and orderly plant shutdown. In the event of high water levels, these flood gates can be installed to provide protection to a level of 1,009.5 msl.

In the Intake Structure, protection to 1009.5 msl is accomplished with flood gates and sandbagging. The plant can be protected by sandbags, temporary earth levees and other methods to allow a safe shutdown with a flood elevation of 1,013 msl.
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Procedure Precautions

AOP-1 requires that a continuous watch be established at river level 980'-0". At this level, hourly cell level measurements will be taken to ensure satisfactory pump submergence. At low river levels, debris and/or ice on the traveling screens and/or trash racks can cause significant head loss potentially reducing intake cell levels below the raw water pump minimum submergence level (MSL) of 976 feet 9 inches. As a precaution, the level of the river and the intake cells is continuously monitored when river level is below 980 feet.

circulating water pumps

The elevation of the circulating water pump suction bells is 974' and 1" msl and the center line of discharge is 979'-0" msl. The circulating water system incorporates three pumps which circulate
screened river water through the turbine condensers and the turbine plant cooling water system heat exchangers.

For the pumps to operate properly the required submergence above the center line of discharge **MUST BE** 4'-0" or an elevation of 983'-0" msl, otherwise pumps would become unstable and trip or be manually shutdown well before intake cell levels decrease to the RW pump MSL.

raw water pumps

The elevation of raw water pump suction bells is 973'-9" msl and the required submergence above the bottom of the suction bell is 3'-0", or an elevation of 976'-9" msl.

Flood protection for raw water pumps at intake

Corps of Engineers,attempt to control the river normal water level at 983 feet during einter months, 992 during navigation periods by controlled releases from dam, Gavins Point.

Estimates exceed 998 feet 1%. 1001.3 peak level of a 1 % probability flood. And 1004.2 is peak level of design flood. 1009.3 feet is compute worse maximum rain runoff from Gavins Point dam upstream.

Worse condition postulated as a credible fault of the Fort Randal Dam simultaneously with 1009.3 peak from maximum runoff.

Permanent protection is provided for the river water pumps up to a river water level of 1007.5 feet by concrete intake structure.

Protection can be provided using sandbags around the travel screens up to 1009.5 feet, and gasketed steel closures at exterior doorway openings in the intake structure walls.

Protection can be provided up to 1014.5 using
Sandbags around the traveling screens and supplementing intake structure perimeter walls with sandbags.

Low river water concern

River water level is normally maintained by controlled releases from Gavins Point dam at 983 feet. The minimum submergence required level for raw water pumps is 976 feet 9 inches. Therefore, there is 6 feet of water above the minimum level

Threats: at low river levels, accumulation of ice on the traveling screens or trash racks can cuase significant head loss, potentially reducing intake cell levels below raw water pump MSL. Below 980 requires continuous watch.

Circulating water pump require 983 feet MSL.

3 separate cells at intake structure, 1 raw water pump in 2 cells, and 2 pumps in the third cell.

Alternate
In the event all raw water pumps become unavailable, water can be obtained from the diesel or motor driven fire pumps. Requires operator actions to make up a interconnection

Otherwise the reactors can be maintained at hot shutdown using steam generators.

Spent Fuel Pool
The
system is designed to cool the pool water by recirculating the contents through
the cooling loop once every two hours with both pumps operating. The tie to the
shutdown cooling system provides a redundant fuel pool cooling loop.The storage pool pumps
circulate borated water through the storage pool heat

exchanger and return it to the pool. Cooling water to the heat exchanger is provided by the component cooling water system (see Section 9.7).While the plant is shutdown, and the core is fully off loaded, the shutdown cooling system provides an emergency backup for the spent fuel pool cooling system in case of failure of that system.
This emergency backup cooling capability of the shutdown cooling system is not available when the CCW or RW systems are out of service for maintenance. This condition is acceptable due to the short duration of the system outages, close attention to the fuel pool heatup rate, and the availability of makeup water sources.

AFW

The emergency feedwater storage tank can be filled with water from the condensate pump discharge, the demineralized water header, the diesel driven auxiliary feedwater pump discharge or, in the case of an emergency, the raw water system and is maintained at or above the Technical Specification minimum volume and ready for operation.
Water sources are the condensate system, demineralized water and the outside condensate storage tank. In an emergency such that these alternate sources are not available, makeup water can be obtained from the raw water system.

Shutdown cooling system
The system utilizes the low-pressure safety injection pumps to circulate the reactor coolant through the two shutdown cooling heat exchangers, returning it to the reactor coolant system through the low-pressure safety injection header.

Cooling water for the shutdown cooling heat exchangers is supplied by the component cooling system (see Section 9.7).
Two shutdown heat exchangers and two low-pressure pumps are provided. If one diesel-generator fails, cooldown at a reduced but acceptable rate is possible with one heat exchanger and one pump in operation.
Cooldown rate is controlled by adjusting the flow rate through the heat exchanger. Either heat exchanger can achieve cooldown at the design rateThe system is operated in the shutdown mode for further cooling of the reactor coolant system when the coolant temperature falls below 350°F and the coolant pressure falls below 300 psia (Ref. 9.3-5).the shutdown cooling system, reactor coolant is circulated using the low-pressure safety injection pumps (see Section 6.2.3). The flow path from the pump discharge runs through normally locked closed valve HCV-335, through the shutdown cooling heat exchangers, and through normally closed valves SI-173 or SI-174, to the normally locked closed valve HCV-341, to the low-pressure safety injection header, and enters the reactor coolant system through the four safety injection nozzles.
The shutdown cooling heat exchangers are described in Section 6.2.3.4 and in Table 6.2-3.

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AND

1.0 INTRODUCTION

By letter dated March 4, 2011, Omaha Public Power District, the licensee, submitted a request for changes to the Fort Calhoun Station Technical Specifications. The requested change would add a surveillance requirement to monitor the river water temperature and level.

2.0 BACKGROUND

UFSAR 2.7 states

The original design basis, based on 1960s data, includes passive protection above the design flood peak stage of 1,006 feet msl due to the elevation of openings to safety related Structures, Systems and Components (SSC's) being 1007'.

In addition, installation of flood gates protects the safety related SSC's against the maximum probable flood peak stage of 1009.3'.

A conservative design flood elevation of 1,006 feet based on a 0.1 percent probability flood.

Without special provisions, the plant can accommodate flood levels of up to 1,007 msl.

Steel flood gates are permanently mounted above and adjacent to openings in structures containing equipment required for a safe and orderly plant shutdown. In the event of high water levels, these flood gates can be installed to provide protection to a level of 1,009.5 msl.

In the Intake Structure, protection to 1009.5 msl is accomplished with flood gates and sandbagging. The plant can be protected by sandbags, temporary earth levees and other methods to allow a safe shutdown with a flood elevation of 1,013 msl.
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Procedure Precautions
AOP-1 requires that a continuous watch be established at river level 980'-0". At this level, hourly cell level measurements will be taken to ensure satisfactory pump submergence. At low river levels, debris and/or ice on the traveling screens and/or trash racks can cause significant head loss potentially reducing intake cell levels below the raw water pump minimum submergence level (MSL) of 976 feet 9 inches. As a precaution, the level of the river and the intake cells is continuously monitored when river level is below 980 feet.

circulating water pumps
The elevation of the circulating water pump suction bells is 974' and 1" msl and the center line of discharge is 979'-0" msl.

For the pumps to operate properly the required submergence above the center line of discharge **MUST BE** 4'-0" or an elevation of 983'-0" msl, otherwise pumps would become unstable and trip or be manually shutdown well before intake cell levels decrease to the RW pump MSL.

raw water pumps
The elevation of raw water pump suction bells is 973'-9" msl and the required submergence above the bottom of the suction bell is 3'-0", or an elevation of 976'-9" msl.

RAW WATER SYSTEM
The raw water system was designed to provide a cooling medium for the component cooling water system. The heat transferred to the raw water is discharged to the river.

For protection against a complete failure of the component cooling system, raw water can be diverted to cool certain engineered safeguards equipment. Raw water direct cooling is utilized via normally hand-jacked, locked-closed valves. Raw water is credited as backup to CCW for fire events as described in Appendix R, Safe Shutdown Analysis. Raw water is also credited in the Fire Safe Shutdown Analysis and the Seismic Safe Shutdown Analysis equipment list for makeup to the Emergency Feedwater Storage Tank.

Four raw water pumps are installed in the intake structure pump house to provide screened river water to the component cooling heat exchangers. The pump discharge piping is arranged as two headers, two raw water lines between the intake structure and the auxiliary building are buried in separate trenches. Each header was designed to accommodate sufficient flow to the component cooling heat exchangers to support normal modes of plant operation.

Water level instrumentation in the intake structure will alarm in the control room if water from any source should endanger the raw water pumps.

In the unlikely event of a complete failure of the component cooling water system, raw water direct cooling capability exists for the following equipment:

- Raw water direct cooling of the shutdown cooling heat exchangers would be needed for long-term decay heat removal after a large LOCA.
- Raw water direct cooling may be used for the control room A/C waterside economizers

- Raw water may be utilized for direct cooling of the safety injection and containment spray pumps, if locally accessible. They may be inaccessible post-RAS, however the pumps can perform their post-accident function without cooling water.
- Raw water direct cooling to the containment air coolers is not required for containment peak pressure suppression. Raw Water (RW) backup of the Component Cooling Water (CCW) being supplied to the Containment Cooling Coils should be carefully implemented. It is not a postulated occurrence with any other event in progress. Calculation FC05662 concludes that the water will flash at the outlet of the Cooling Coils because the pressure will be less than 3.718 PSIA (saturation pressure of 150°F), and Raw Water should not be used in any condition to cool the containment if the river level is less than 976'9" this may apply to any river level less than 983.5'. This condition will allow boiling to occur in the cooling coils causing water flashing, which may damage coils.

3.0 REGULATORY EVALUATION

4.0 TECHNICAL EVALUATION

Balance of Plant Branch input into the Safety Evaluation for
Fort Calhoun Station Request for Changing Technical Specification 2.16

1.0 Introduction

By letter dated March 4, 2011, Omaha Public Power District, the licensee, submitted a request for changes to the Fort Calhoun Station Technical Specifications. The requested change establishes the limiting condition for operation (LCO) requirements for the ultimate heat sink. The licensee proposes to remove the current TS 2.16 titled River Water, and add a TS 2.16 titled Ultimate Heat Sink with provision that are more in line with Standard Technical Specification Combustion Engineering Plants, NUREG 1432. The licensee proposes to add 1) a surveillance requirement to monitor the river water temperature not to exceed 90 degrees, 2) and surveillance requirement to monitor the river not to exceed a low level of 976 feet 9 inches.

The staff notes that in the processes of deleting the current River Water specification and replacing it with a Ultimate Heat Sink specification, the licensee is deleting the requirement to implement an emergency plan to protect the plant when the river rises to 1004.2 feet, and shut down the reactor when the river level rises above 1009 feet.

2.0 Background

The Missouri River is the source of cooling water that services Fort Calhoun Nuclear Power Station. The Corps of Engineers manage the Missouri River water level and flow rates using massive earthen upstream dams. Fort Calhoun Nuclear Power station uses river water for its nonsafety-related circulating water system, and its safety-related raw water system. The river water intake structure is configured in three bays. Two bays have one raw water pump, and one bay has two raw water pumps. Water level instrumentation in the intake structure provides the control room with indication alarms if the raw water pumps are in danger of flooding.

2.0 Regulatory Evaluation

The regulatory requirements and the guidance used by the NRC staff for its review are:

Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, in particular

GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions,

GDC 19 as it relates to the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear plant safely under normal conditions and to maintain it in a safe condition under accident conditions,

GDC 44 as it relates to cooling water systems transferring heat loads from safety-related structures, and systems to the heat sink under normal operating and accident conditions,

Standard Review Plan (SRP), NUREG-0800,
Section 6.2.1 "Containment Functional Design",

Section 6.2.1.1.A "PWR [Pressurized Water Reactors] Dry Containments, Including Subatmospheric Containments,"
Section 6.2.2 "Containment Heat Removal Systems,"
Section 9.4.1 "Control Room Area Ventilation System," and
Section 9.2.1 "Station Service Water System."

Section 9.2.5 Ultimate Heat Sink

The NRC staff's review of the proposed change focused on the impact on the ability of the SR structures, systems, and components (SSCs) to perform their safety functions. Acceptability of the proposed change is judged based upon continued conformance with the plant licensing basis as reflected primarily in Updated Final Safety Analysis Report (UFSAR).

3.0 Technical Evaluation

3.1 Proposed Technical Specification

3.1.1. Currently, TS 2.16 states the applicability

Applied to Missouri River level as measured at the intake structure at the Fort Calhoun Station.

The licensee is proposing to change the applicability of TS 2.16 to read:

Applied to operational status of the Missouri River at the intake structure at the Fort Calhoun Station for reactor coolant temperature Tcold -> 210°F.

3.1.2 Currently the TS for river level state

If the Missouri River level is less than 976 feet 9 inches the reactor will be placed in a cold shutdown condition using normal operating procedures. At river levels less than 980 feet a continuous watch will be maintained to assure no sudden loss of water supply occurs.

3.1.3 The licensee is proposing to define the operability of the UHS to read:

The OPERABILITY of the UHS is based on having a minimum Missouri River water level of 976 feet 9 inches mean sea level and a maximum river water temperature of 900F.

3.1.4 In the event one of the conditions cannot be satisfied, the licensee proposes the following actions:

If the Missouri River level is less than 976 feet 9 inches or river temperature is greater than 900F, then the reactor will be placed in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 36 hours using normal operating procedures. At river levels less than 980 feet, a continuous watch will be maintained to assure no sudden loss of water supply occurs.

3.1.5 The licensee is proposing to delete the shutdown requirement upon high river level that read:

Specifications

1) If the Missouri River level exceeds 1009 feet the reactor will be placed in a cold shutdown condition using normal operating procedures. When the river level reaches

elevation 1004.2 feet and rising, the emergency plan to protect the plant will be instituted.

Currently the licensee has operability requirements for the raw water pumps associated with river temperature in TS specification 2.4, Containment Cooling. Within TS 2.4, when the river temperature is below 60 degrees, one raw water pump may be inoperable indefinitely. The licensee proposes no changes to this provision.

4.0 Staff Evaluation

4.1 Raw Water System

Fort Calhoun Station is a single unit site, pressurized water reactor, designed by Combustion Engineering. The containment is a prestressed, post tensioned concrete structure is designed for low leakage at a design pressure of 60 psig and 305°F. The Missouri River provides the ultimate heat sink (UHS). Corps of Engineers control releases from Gavins Point dam to manage the river water level normally at 983 feet during winter months, and at 992 feet during navigation periods.

The raw water system provides a cooling medium for the component cooling water system. Raw water is also credited as backup to CCW for fire events as described in Appendix R, Safe Shutdown Analysis; and the Seismic Safe Shutdown Analysis equipment list for makeup to the Emergency Feedwater Storage Tank. Originally, the plant was designed with three raw water pumps, but later upgraded to four pumps to allow for continued operations with one pump out of service for maintenance. The design rating for the raw water pumps is 5325 gallons per minute (gpm) with 118 feet total dynamic head (TDH), at a design pressure of 150 psig and temperature of 150 degrees F. The pumps discharge into two separate 20" headers, which flow from the intake structure into the auxiliary building. Each header provides sufficient flow to the component cooling heat exchangers to support normal modes of plant operation, and support design basis accident mitigation.

The intake structure houses three condenser cooling water pumps and four raw water pumps. The elevation of the condenser cooling water pumps' suction bells is 974'-1" msl and the center line of discharge is 979'-0" msl. In order to operate properly, the circulating water pumps require a four foot (4') submergence above the center line of discharge, equal to river elevation of 983'-0" msl. Otherwise pumps become unstable and may trip. The elevation of raw water pump suction bells is 973'-9" and the required submergence above the bottom of the suction bell is 3'-0", or an elevation of 976'-9". Therefore, the condenser cooling water pumps will shutdown before intake river level decreases to the RW pump minimum level requirement.

In the event of a complete failure of the component cooling system, raw water can used directly to cool certain engineered safeguards equipment. In the unlikely event of a complete failure of the component cooling water system, raw water direct cooling capability exists for the following equipment:

- Raw water direct cooling of the shutdown cooling heat exchangers would be needed for long-term decay heat removal after a large LOCA.
- Raw water direct cooling may be used for the control room A/C waterside economizers

- Raw water may be utilized for direct cooling of the safety injection and containment spray pumps, if locally accessible. They may be inaccessible post-RAS, however the pumps can perform their post-accident function without cooling water.
- Raw water direct cooling to the containment air coolers is not required for containment peak pressure suppression. Raw Water (RW) backup of the Component Cooling Water (CCW) being supplied to the Containment Cooling Coils should be carefully implemented. It is not a postulated occurrence with any other event in progress. Calculation FC05662 concludes that the water will flash at the outlet of the Cooling Coils because the pressure will be less than 3.718 PSIA (saturation pressure of 150°F), and Raw Water should not be used in any condition to cool the containment if the river level is less than 976'9" this may apply to any river level less than 983.5'. This condition will allow boiling to occur in the cooling coils causing water flashing, which may damage coils.

4.1 Design Basis Accidents

In the event of a Design Basis Accidents (DBA), a safety injection signal (SIS) will start all 4 raw water pump. In the event only one emergency diesel generator (EDG) is available, only 2 raw water pumps will start.

In the event of a DBA, the CCW system provides containment cooling by removing the heat from the containment spray and the containment air cooling system. The containment air cooling system includes four separate self-contained units. CCW flow of 4,680 gpm with an inlet temperature of 120°F will remove 280 x 10^6 Btu/hr. The containment air cooling rate is sufficient to cool the containment air during normal operation and limit the pressure rise in the event of a design accident.

Raw water system is design to remove sufficient heat from CCW so that the maximum return temperature does not exceed 160 degrees. The raw water system is design to satisfy it design function with a single failure and loss of IA. Air accumulators are provided at the intake structure to operate valves upon loss of normal IA. When the river water temperature is below 60 degrees, only one pump is required to meet design basis heat removal.

4.1.4 ECCS Systems

The following are Engineering Safeguard Systems that interface with the UHS:

- Containment Spray System. The containment spray system removes heat by spraying cool borated water through the containment atmosphere. Heat is transferred to the component cooling system through the shutdown heat exchangers. The containment spray system has the ability to recirculate water from the containment sump through the shutdown cooling heat exchangers into the containment atmosphere.
- Containment Air Cooling and Filtering System. This system removes heat by circulating the post-accident containment atmosphere over coils cooled by the component cooling water system and removes particulates by filtration.
- Component Cooling Water System. Portions of the component cooling water system provide cooling water necessary for operation of engineered safeguards equipment. Component Cooling and Raw Water Systems The component cooling

system provides cooling water to the shutdown heat exchangers, the containment air cooling system, the spent fuel pool heat exchanger, the letdown heat exchanger, the sampling heat exchanger, ventilation equipment and various pumps. Heat removed by the component cooling water system is transferred to the raw water system by the component cooling heat exchangers.

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- Raw Water System. The raw water system provides a cooling medium for the component cooling water system. The raw water system is a once through system operating with screened river water. Further, the system arrangement permits the raw water to be circulated through portions of the component cooling system piping to provide direct cooling of vital engineered safeguards components in the unlikely event of all of the component cooling pumps and heat exchangers being unavailable to fulfill their design function.
- Control Room HVAC System. HVAC systems which are required for operation of engineered safeguards are considered to be essential auxiliary support systems.

Shutdown Cooling Heat Exchangers

The shutdown cooling heat exchangers are used to remove decay and sensible heat during plant cooldowns and cold shutdowns. The units were sized to remove 37,100,000 BTU/hr, based on 4500 gpm of cooling water at 93°F inlet temperature and 3000 gpm of reactor coolant at 140°F inlet temperature. The units were further specified to accept a 70°F to 300°F transient when the containment spray pump suction is switched to the containment recirculation line inlet at elevation 994'-0".

Water from the containment sump can also be circulated by the containment spray pumps and is cooled by the shutdown cooling heat exchangers. The operator would direct this cooled water to the suction of the high-pressure safety injection pumps. This would provide subcooled water to the core. The low-pressure safety injection pumps may also be used for long term cooling, if the reactor coolant pressure is sufficiently low.

Following RAS, the operators may manually align the containment spray system to provide cooled water from the shutdown cooling heat exchangers to the suction of the high pressure safety injection pumps.

Following a LOCA, during the recirculation mode, the containment spray system can be aligned to the suction of the high pressure safety injection pumps to provide a source of subcooled water to supplement long term core cooling. Under this mode of operation, the heat removal capability per heat exchanger is 87.5 x 10^6 Btu/hr based upon 4,000 gpm of cooling water at 114°F inlet temperature..

Shutdown Cooling System

The shutdown cooling system is used to reduce the temperature of the reactor coolant at a controlled rate from 350°F to a refueling temperature. It also provides an emergency backup for the spent fuel pool cooling system in the event of failure of that system. it also provides an emergency backup for the spent fuel pool cooling system in the event of failure of that system

4.2.2 **Flood Protection**

UFSAR 2.7 states The original design basis, based on 1960s data, includes passive protection above the design flood peak stage of 1,006 feet msl due to the elevation of openings to safety related Structures, Systems and Components (SSC's) being 1007'.

In addition, installation of flood gates protects the safety related SSC's against the maximum probable flood peak stage of 1009.3'.

A conservative design flood elevation of 1,006 feet based on a 0.1 percent probability flood.

Without special provisions, the plant can accommodate flood levels of up to 1,007 msl.

Steel flood gates are permanently mounted above and adjacent to openings in structures containing equipment required for a safe and orderly plant shutdown. In the event of high water levels, these flood gates can be installed to provide protection to a level of 1,009.5 msl.

In the Intake Structure, protection to 1009.5 msl is accomplished with flood gates and sandbagging. The plant can be protected by sandbags, temporary earth levees and other methods to allow a safe shutdown with a flood elevation of 1,013 msl.
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Procedure Precautions
AOP-1 requires that a continuous watch be established at river level 980'-0". At this level, hourly cell level measurements will be taken to ensure satisfactory pump submergence. At low river levels, debris and/or ice on the traveling screens and/or trash racks can cause significant head loss potentially reducing intake cell levels below the raw water pump minimum submergence level (MSL) of 976 feet 9 inches. As a precaution, the level of the river and the intake cells is continuously monitored when river level is below 980 feet.

Flood protection for raw water pumps at intake

Corps of Engineers,attempt to control the river normal water level at 983 feet during einter months, 992 during navigation periods by controlled releases from dam, Gavins Point.

Estimates exceed 998 feet 1%. 1001.3 peak level of a 1 % probability flood. And 1004.2 is peak level of design flood. 1009.3 feet is compute worse maximum rain runoff from Gavins Point dam upstream.

Worse condition postulated as a credible fault of the Fort Randal Dam simultaneously with 1009.3 peak from maximum runoff.

Permanent protection is provided for the river water pumps up to a river water level of 1007.5 feet by concrete intake structure.

Protection can be provided using sandbags around the travel screens up to 1009.5 feet, and gasketed steel closures at exterior doorway openings in the intake structure walls.

Protection can be provided up to 1014.5 using
Sandbags around the traveling screens and supplementing intake structure perimeter walls with sandbags.

Low river water concern

River water level is normally maintained by controlled releases from Gavins Point dam at 983 feet. The minimum submergence required level for raw water pumps is 976 feet 9 inches. Therefore, there is 6 feet of water above the minimum level

Threats: at low river levels, accumulation of ice on the traveling screens or trash racks can cuase significant head loss, potentially reducing intake cell levels below raw water pump MSL. Below 980 requires continuous watch.

Circulating water pump require 983 feet MSL.

3 separate cells at intake structure, 1 raw water pump in 2 cells, and 2 pumps in the third cell.

Alternate
In the event all raw water pumps become unavailable, water can be obtained from the diesel or motor driven fire pumps. Requires operator actions to make up a interconnection

Otherwise the reactors can be maintained at hot shutdown using steam generators.

Spent Fuel Pool
The
system is designed to cool the pool water by recirculating the contents through the cooling loop once every two hours with both pumps operating. The tie to the shutdown cooling system provides a redundant fuel pool cooling loop.The storage pool pumps circulate borated water through the storage pool heat exchanger and return it to the pool. Cooling water to the heat exchanger is provided by the component cooling water system (see Section 9.7).While the plant is shutdown, and the core is fully off loaded, the shutdown cooling system provides an emergency backup for the spent fuel pool cooling system in case of failure of that system.
This emergency backup cooling capability of the shutdown cooling system is not available when the CCW or RW systems are out of service for maintenance. This condition is acceptable due to the short duration of the system outages, close attention to the fuel pool heatup rate, and the availability of makeup water sources.

AFW

The emergency feedwater storage tank can be filled with water from the condensate pump discharge, the demineralized water header, the diesel driven auxiliary feedwater pump discharge or, in the case of an emergency, the raw water system and is maintained at or above the Technical Specification minimum volume and ready for operation.
Water sources are the condensate system, demineralized water and the outside condensate storage tank. In an emergency such that these alternate sources are not available, makeup water can be obtained from the raw water system.

Shutdown cooling system

The system utilizes the low-pressure safety injection pumps to circulate the reactor coolant through the two shutdown cooling heat exchangers, returning it to the reactor coolant system through the low-pressure safety injection header.

Cooling water for the shutdown cooling heat exchangers is supplied by the component cooling system (see Section 9.7).
Two shutdown heat exchangers and two low-pressure pumps are provided. If one diesel-generator fails, cooldown at a reduced but acceptable rate is possible with one heat exchanger and one pump in operation.
Cooldown rate is controlled by adjusting the flow rate through the heat exchanger. Either heat exchanger can achieve cooldown at the design rateThe system is operated in the shutdown mode for further cooling of the reactor coolant system when the coolant temperature falls below 350°F and the coolant pressure falls below 300 psia (Ref. 9.3-5).the shutdown cooling system, reactor coolant is circulated using the low-pressure safety injection pumps (see Section 6.2.3). The flow path from the pump discharge runs through normally locked closed valve HCV-335, through the shutdown cooling heat exchangers, and through normally closed valves SI-173 or SI-174, to the normally locked closed valve HCV-341, to the low-pressure safety injection header, and enters the reactor coolant system through the four safety injection nozzles.
The shutdown cooling heat exchangers are described in Section 6.2.3.4 and in Table 6.2-3.

Balance of Plant Branch input into the Safety Evaluation for
Fort Calhoun Station Request for Changing Technical Specification 2.16

1.0 Introduction

By letter dated March 4, 2011, Omaha Public Power District, the licensee, submitted a request for changes to the Fort Calhoun Station (FCS) Technical Specifications (TS). The requested change establishes the limiting condition for operation (LCO) requirements for the ultimate heat sink (UHS). The licensee proposes to remove the current TS 2.16 titled "River Water", and add a TS 2.16 titled "Ultimate Heat Sink" with provisions that are more in line with NUREG 1432, Standard Technical Specification Combustion Engineering Plants. The licensee proposes to add a surveillance requirement to monitor the river water temperature not to exceed 90 degrees, and surveillance requirement to monitor the river level not to exceed a low level of 976 feet 9 inches. The staff notes that in the processes of replacing the current specification, the licensee proposes to delete the existing requirement to implement an emergency plan to protect the plant when the river rises to 1004.2 feet, and shut down the reactor when the river level rises above 1009 feet.

2.0 Background

Fort Calhoun Station is a single unit site, pressurized water reactor, designed by Combustion Engineering. The Corps of Engineers manages the Missouri River water level and flow rates using massive earthen upstream dams. The Missouri River serves as the UHS, supplying the cooling water for Fort Calhoun Station. Fort Calhoun Station uses river water for both its nonsafety-related circulating water system and its safety-related raw water system. The raw water system has four pumps installed in three separate bays in the intake structure. Two bays have one raw water pump, and one bay has two raw water pumps. Water level instrumentation in the intake structure provides the control room with indication alarms if the raw water pumps are in danger of flooding.

2.0 Regulatory Evaluation

CEOG STS describes the UHS as a complex of water sources and any necessary retaining structures, e.g. river and its associated dam. In addition, the UUS includes any canals that connect the source with the cooling water system intake structure. However, the UHS does not include the intake structure.

Standard Review Plan (SRP), NUREG-0800 Section 9.2.5, describes the UHS as an assured supply of water credited for dissipating reactor decay heat and essential station heat loads after a normal reactor shutdown, or shutdown following an accident or transient. The water source must be adequate to supply the required flow of cooling water at the appropriate temperatures for normal, accident, and shutdown conditions for at least 30 days.

Regulator Guide 1.27 describes methods acceptable for ensuring UHS can withstand the effects of natural phenomena

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions, 10 CFR 50 App A, GDC 19 as it relates to the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear plant safely under normal conditions and to maintain it in a safe condition under accident conditions,

10 CFR 50 App A, GDC 44 as it relates to cooling water systems transferring heat loads from safety-related structures, and systems to the heat sink under normal operating and accident conditions,

3.0 Technical Evaluation

3.1 Proposed Technical Specification Changes

3.1.1. Currently, TS 2.16 states the applicability

> Applied to Missouri River level as measured at the intake structure at the Fort Calhoun Station.

The licensee proposes changing the applicability of TS 2.16 to read:

> Applied to operational status of the Missouri River at the intake structure at the Fort Calhoun Station for reactor coolant temperature Tcold -> 210°F.

3.1.2 Currently the TS for river level states,

> If the Missouri River level is less than 976 feet 9 inches the reactor will be placed in a cold shutdown condition using normal operating procedures. At river levels less than 980 feet a continuous watch will be maintained to assure no sudden loss of water supply occurs.

3.1.3 The licensee is proposing to define the operability of the UHS to read:

> The OPERABILITY of the UHS is based on having a minimum Missouri River water level of 976 feet 9 inches mean sea level and a maximum river water temperature of 900F.

3.1.4 In the event one of the conditions cannot be satisfied, the licensee proposes the following actions:

> If the Missouri River level is less than 976 feet 9 inches or river temperature is greater than 900F, then the reactor will be placed in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 36 hours using normal operating procedures. At river levels less than 980 feet, a continuous watch will be maintained to assure no sudden loss of water supply occurs.

3.1.5 The licensee is proposing to delete the shutdown requirement upon high river level that read:

> Specifications
> 1) If the Missouri River level exceeds 1009 feet the reactor will be placed in a cold shutdown condition using normal operating procedures. When the river level reaches elevation 1004.2 feet and rising, the emergency plan to protect the plant will be instituted.

Currently the licensee has operability requirements for the raw water pumps associated with river temperature in TS specification 2.4, Containment Cooling. Within TS 2.4, when the river temperature is below 60 degrees, one raw water pump may be inoperable indefinitely. The licensee proposes no changes to this provision.

4.0 Staff Evaluation

4.1 UHS Description

The Missouri River provides the ultimate heat sink (UHS). Corps of Engineers control releases from six upstream dams to manage the river water level normally at 983 feet during winter months, and at 992 feet during navigation periods. The nearest dam is Gavins Point, approximately three miles upstream of the FCS. The Corps of Engineers' annual operating plan states normal release from Gavins Point is 12,000 cfs, depending largely on availability of water. Flows during the non-navigation season will range from 12,000 cfs to 18,000 cfs. In years when an extended period of drought has depleted storage reserves, release flows may run as low as 6,000 cfs.

4.2 River High Level

Studies show the Missouri River water level would increase to 998 feet only 1% of the time. The momentary yearly flood peak with an expected 1% probability is set at 1001.3 feet. The design flood river level is conservatively set at 1006 feet, based upon a 0.1% probability. The Corps of Engineers considers a total failure of a large upstream dam not credible. However, in the event of an upstream credible fault of either the Fort Randal or Oahe dams coincident with runoff from maximum probable rain storms, the Corps of Engineers expect a peak of 1009.3 feet. From a study perform in 1995 by Federal Emergency Management Agency (FEMA), the 100-year flood level at the site is 1006.1' and the 500-year flood level is 1008.5'.

The original design basis from 1960's data, assumes credit for passive protection above the design flood peak stage of 1,006 feet. The concrete intake structure and elevation of openings provides permanent protection for the raw water pumps up to a river water level of 1007.5 feet. Without special provisions, the plant can accommodate flood levels of up to 1,007 msl.

The licensee can install flood gates to offer protection of the safety related SSC's against the maximum probable flood peak stage of 1009.3'. Without special provisions, the plant can accommodate flood levels of up to 1,007. Steel flood gates are permanently mounted above and adjacent to openings in structures containing equipment required for a safe and orderly plant shutdown. In the event of high water levels, these flood gates and sandbags around the travel screens can be installed to provide protection to a level of 1,009.5. Protection can be provided up to 1014.5 using sandbags around the traveling screens and supplementing intake structure perimeter walls with sandbags

In the USAR, Section XXX, the licensee states, "Because of the temporary nature of the emergency measures, the plant would normally be shut down when the flood water level exceeds the permanently installed protection provisions to minimize the likelihood of an accident."

The licensee proposes to delete the current TS requirement to place the reactor in a cold shutdown condition if the Missouri River level exceeds 1009 feet. When the river level reaches elevation 1004.2 feet and rising, the emergency plan to protect the plant will be instituted.
In RAI XXXX, the staff requested the licensee provide justification for removing this requirement from TS.

4.3 River Low Level

The Corps of Engineers manages the river water level normally at 983 feet during winter months and at 992 feet during navigation periods. River level may run low during extended periods of drought when the Corps of Engineers reduce flow from the upstream dams. At low river levels, debris and/or ice on the traveling screens and/or trash racks can cause significant head loss potentially reducing intake cell levels below the raw water pump minimum submergence levels. The elevation of raw water pump suction bells is 973'-9"; however, the pumps require 3 feet of submergence above the bottom of the suction bell, elevation 976'-9". The circulating water pumps for the condenser have a much higher MSL requirement (983 feet 0 inches) and would become unstable and trip or be manually shutdown well before intake cell levels decrease to the RW pump MSL.

As a precaution, whenever the river level goes below 980 feet, the licensee's procedures "AOP-1" establishes monitoring of the level of the river and the intake cells to ensure satisfactory pump submergence. In the event all raw water pumps become unavailable, water can be obtained from the diesel or motor driven fire pumps. Operator actions is required to make up a interconnection. Otherwise the reactors can be maintained at hot shutdown using auxiliary feedwater to the steam generators.

4.4 River Temperature

The current TS do not have any restriction on the river water temperature. The licensee proposes adding a TS surveillance requiring the river temperature remain below 90 F. In USAR, the licensee references calculations that show the river water system can meet the design requirements to mitigate a DBA if the river temperature is maintain below 90 F. The raw water is used as a cooling medium for the CCW heat exchangers. The licensee calculations show that 90 F river water to the CCW heat exchangers can remove sufficient heat from the CCW water in order to supply a 160 F maximum return temperature. With a maximum 160 F return temperature, the CCW system can provide its safety functions in the event of a DBA.

4.5 ECCS Systems Interfacing with UHS

The following engineering safeguard systems interface with the UHS to provide a safety function; component cooling water system, containment spray system, containment air cooling and filtering system.

4.5.1 Raw Water System

The raw water system provides a cooling medium for the component cooling water system. In the event CCW pumps or heat exchangers are unavailable, raw water can circulated directly to vital engineered safeguards components. Raw water is credited as backup to CCW for fire events as described in Appendix R safe shutdown analysis. Raw water is on the seismic safe shutdown analysis equipment list for makeup to the emergency feedwater storage tank.

The intake structure houses four raw water pumps. Originally, the plant was designed with three raw water pumps, but later upgraded to four pumps to allow for continued operations with one pump out of service for maintenance. The elevation of raw water pump suction bells is 973'-9". The pumps require 3 feet of submergence above the bottom of the suction bell, elevation 976'-9". The design rating for the raw water pumps is 5325 gallons per minute (gpm) with 118 feet total dynamic head (TDH), at a design pressure of 150 psig and temperature of 150 degrees F. The pumps discharge into two separate 20" headers up into the auxiliary building. Each header provides sufficient flow to the component cooling heat exchangers to support normal modes of plant operation, and support design basis accident mitigation.

In the event of a complete failure of the component cooling system, raw water can used directly to cool certain engineered safeguards equipment:

- shutdown cooling heat exchangers for long-term decay heat removal after a large LOCA.
- control room A/C waterside economizers
- cooling of the safety injection and containment spray pumps, if locally accessible. They may be inaccessible post-RAS, however the pumps can perform their post-accident function without cooling water.
- Raw water to the containment air coolers is not required for containment peak pressure suppression.

Raw Water supplied directly to the containment cooling coils should be carefully implemented. It is not a postulated occurrence with any other event in progress. Calculation FC05662 concludes that the water will flash at the outlet of the cooling coils because the pressure will be less than 3.718 PSIA (saturation pressure of 150°F). Raw Water should not be used to cool the containment if the river level is less than 976'9"; and may apply to any river level less than 983.5'. This condition will allow boiling to occur in the cooling coils causing water flashing, which may damage coils.

4.5.2 Containment Spray System.

In the event of a main steam line break, the containment spray system limits the containment pressure rise by spraying cool borated water throughout the containment atmosphere. The containment spray system can assist with long term cooling after a LOCA by recirculating water from the containment sump through the two shutdown cooling heat exchangers, transfering heat to the component cooling system. The cool water can be directed to safety injection pump suctions, or directly to the containment atmosphere. The containment spray system has three pumps, but only one pump is required to meet the design requirements for a DBA.

4.5.3 Containment Air Cooling and Filtering System.

The containment air cooling and filtering system provides long term cooling of the containment atmosphere, and filters airborne activity. This system circulates the post-accident containment atmosphere through cooling coils, transferring heat to the component cooling water system. The system has four air handling units, but only two have a filtering capability. The two units with filtering capability are designed to remove 140 x 10^6 BTU/hr at DBA conditions. The other two units are designed to remove 70 x 10^6 BTU/hr at DBA conditions. The system is designed to remove 280 x 10^6 BTU maximum with instrument air available, from a saturated containment atmosphere at 60 psig and 288 F, in order prevent containment from exceeding 60 psig design limit. In the event of the loss of normal power sources and the loss of an emergency diesel generator only two of the four units will be available, with only one having filtering capabilities. [Hence, provide 140 +70 =210 x 106 BTU, which is less than required design 280]

4.5.4 Component Cooling Water System.

The CCW system provides cooling water to the shutdown heat exchangers, the containment air coolers, the spent fuel pool heat exchanger, the letdown heat exchanger, the blowdown sampling coolers, control room economizer coils and various pump coolers. The CCW system has three motor driven pumps. Each pump is rated at 3425 gpm. The CCW system has four heat exchangers, which transfer heat from the CCW system to the raw water system. Each heat exchanger is rated at 134 BTU/hr at DBA conditions. The four heat exchangers and two or the three pumps is sufficient to provide requirements for DBA loads. Even with a single failure and loss of IA, CCW can satisfy DBA load requirements.

"A contribution from the containment air cooler (cooled by CCW) is credited in the mitigation of containment peak pressure for a main steam line break."

"For a LOCA, the containment air coolers are credited for both mitigation of peak containment pressure and long term containment cooling."

The licensee calculates the CCW return temperature can be maintained below 160 F under DBA loads and conditions with the river water at 90 F.
Normally CCW temperature is maintained between 55 to 110 F using a combination of the four CCW heat exchangers, the three CCW pumps and the four river water pumps. CCW is used during normal plant shutdowns to cool the reactor coolant water from 350 F to below 200 F for refueling operations. In the event of a safety injection actuation signal (SIAS) the CCW system components automatic start and align to deliver design basis flow to engineered safeguards components, and isolate non-critical components. In the event instrument air is not available during the DBA, those CCW component without accumulators will fail to their safe position, which may prevent isolation of CCW flow to non-essential components. In the event of a loss of normal power and a failure of one EDG to start, only one CCW pump will start automatically. A second CCW pump will be available for start on the swing bus. After the SIRW tank is depleted and a recirculation actuation signal (RAS) is generated, the valves supplying CCW to the shutdown heat exchangers will remain closed. This alignment ensures optimal flow to the containment air coolers, which is credited for long term containment cooling.

In the event of a CCW component failure rendering the CCW system inoperable, the raw water system can be manually aligned to flow through the CCW system piping to critical engineered safeguard components.

4.5.5 HVAC System

The control room HVAC is an essential auxiliary support system and designated a engineered safeguards system. The main control room HVAC system consists of two refrigeration and air handling units. The air conditioning units have hermetic compressors and air cooled condensers located on the auxiliary building roof. CCW is only used for the water side of the economizer, which is required for control room cooling when the ambient temperature is below 0°F. A ventilation isolation actuation signal (VIAS) will close the CCW inlet and outlet valves on the economizers.

Switch gear room also has two HVAC systems. The containment HVAC system maintains the temperature inside containment is maintained below 120 F during normal operations.

4.5.6 Shutdown Cooling Heat Exchanger

The shutdown cooling heat exchangers remove decay and sensible heat during normal and accident cooldowns. Once the reactor coolant system decreases below 300 psia, operator use the two low-pressure safety injection pumps to circulate the reactor coolant through two shutdown cooling heat exchangers, to reduce the temperature of the reactor coolant at a controlled rate from 350°F to a refueling temperature. The heat exchangers can remove 37.1 x 10^6 BTU/hr, based on 4500 gpm of CCW cooling water at 93°F inlet temperature and 3000 gpm of reactor coolant at 140°F inlet temperature.

In the event one diesel-generator fails, cooldown at a reduced but acceptable rate is possible with one heat exchanger and one pump. The shutdown cooling system also provides an emergency backup for the spent fuel pool cooling system in the event of failure of that system.

Following RAS, the containment spray pumps can cool the water from the containment sump by circulating the water through the shutdown cooling heat exchangers. Each of the shutdown heat exchangers can remove 58.9 x 10^6 BTUs based upon 2,937 gpm of cooling water at 95 F inlet, and 2,250 gpm spray flow at 212 F. Once the water exits the heat exchangers, the operator can direct the cooled water to the suction of the high-pressure safety injection pumps, which inject the subcooled water into the core. Under this mode of operation, the heat removal capability per heat exchanger is 87.5 x 10^6 Btu/hr based upon 4,000 gpm of cooling water at 114°F inlet temperature. The low-pressure safety injection pumps may also be used for long term cooling, if the reactor coolant pressure is sufficiently low.

4.5.7 Spent Fuel Pool

The system is designed to cool the pool water by recirculating the contents through the storage pool heat exchanger once every two hours with both pumps operating. Cooling water to the heat exchanger is provided by the CCW system. While the plant is shutdown, and the core is fully off loaded, the shutdown cooling system provides an emergency backup for the spent fuel pool cooling system in case of a failure.

4.6 Design Basis Accidents

In the event of a Design Basis Accidents (DBA), a safety injection signal (SIS) will start all 4 raw water pump. In the event only one emergency diesel generator (EDG) is available, only 2 raw water pumps will start.

The containment air cooling and ventilating system can be used to reduce the high containment atmospheric pressure and to remove decay heat from the building if it becomes necessary to stop shutdown cooling.

In the event of a DBA, the CCW system provides containment cooling by removing the heat from the containment spray and the containment air cooling system. The containment air cooling system includes four separate self-contained units. CCW flow of 4,680 gpm with an inlet temperature of 120°F will remove 280 x 10^6 Btu/hr. The containment air cooling rate is sufficient to cool the containment air during normal operation and limit the pressure rise in the event of a design accident.

Raw water system is design to remove sufficient heat from CCW so that the maximum return temperature does not exceed 160 degrees. The raw water system is design to satisfy it design function with a single failure and loss of IA. Air accumulators are provided at the intake structure to operate valves upon loss of normal IA. When the river water temperature is below 60 degrees, only one pump is required to meet design basis heat removal.

4.6.1 Main Steam Line Break

"A contribution from the containment air cooler (cooled by CCW) is credited in the mitigation of containment peak pressure for a main steam line break."

4.6.2 Loss of Coolant Accident

"For a LOCA, the containment air coolers are credited for both mitigation of peak containment pressure and long term containment cooling."

Some of the issues involved with this amendment:

Issues for Fort Calhoun Technical Specification Change

<u>Issue 1- low river level does not assure availability of raw water pumps to bring plant to cold shutdown conditions</u>

The licensee is proposing a TS surveillance requirement for low level of 976 feet 9 inches of the UHS – Missouri River. The licensee states a river level of 976 feet 9 inches is required for raw water pumps minimum submergence level. The raw water pumps provide the cooling medium for the component cooling water heat exchangers, which are used to cool the plant down to cold shutdown. If the pumps are lost, then the plant will not have a normal means to achieve cold shutdown, nor will it have a safety-related means to obtain cold shutdown. In addition, raw water is credit as an emergency makeup source to the emergency condensate storage tank as a source of water for auxiliary feedwater pumps that maintain the plant in a hot shutdown condition. 10 CFR 50.36 requires the licensee establish limiting conditions for operation, and when they are not met, the licensee shall shut down the reactor. The licensee is proposes a TS limit on low river level at the point the raw water pumps become unavailable for use; however, the raw water pumps are required to shut down the reactor. Therefore, the licensee is proposing an action that could not be accomplished.

<u>Issue 2- removing high river level limit no longer gives assurance the plant can safely shut down prior to losing equipment at intake structure</u>

The licensee states that flooding at Fort Calhoun is highly unlikely, therefore, the licensee requests to remove the provision in TS requiring a plant shut down if the Missouri River level reach 1009 feet. The licensee states that a high river level technical specification does not meet the requirements of 10 CFR 50.63. However, 10 CFR 50.63(c)(2)(ii)(D) *Criterion 4* states "a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." In addition, 10 CFR 50.36(c)(2)(ii)(B) requires the licensee to establish a TS limiting condition for a operating restriction that is a initial condition of a design basis accident or transient analysis. Based upon recent operating experience, the staff regards susceptibility to flooding as risk to the plant's ability to maintain safe shutdown capabilities. A TS provision for high river level needs to be established in which the plant can safely shutdown using normal operating equipment and procedures prior to the loss of this equipment from flooding conditions.
What is the latest data on river level peaking for the Missouri River at the Ft Calhoun location? Does the latest data on river level support current high level setpoint?

<u>Issue 3 – how is the licensee capturing the uncertainty in the instrumentation used for monitoring river temperature and level</u>

In the FCS USAR, Appendix G, Criterion 12, the licensee states instrumentation conforms to the applicable Institute of Electrical and Electronics Engineers (IEEE) standards. The licensee is proposing to add a new TS surveillance to limit the UHS temperature to 90°F. The licensee credits existing instrumentation to monitor river temperature. The licensee states that it will encompass the loop uncertainty for UHS temperature in the surveillance test. Also, licensee proposes to add a TS surveillance to limit the UHS to a river level of 976 feet 9 inches. However, the license states its current river level instrumentation is not very accurate and intends on replacing the river level instrumentation with a more reliable system. The licensee designates a TS surveillance limit for the river water maximum temperature and minimum level at the assumed limit used in their design basis calculations showing sufficient heat removal in order to mitigate a design basis accident. The licensee needs provide the methods used for monitoring river water temperature and level to comply with their current licensing basis.

<u>Issue 3 – is there an adequate basis for 90 degree limit for the containment analysis</u>

Details can be supplied by Containment and Ventilation Branch

<u>Issue 4 – Has the calculation for the CCW heat exchanger's heat removal capability to supply 160 degree return water with 90 degree river water, been reviewed by the NRC staff?</u>

Some of the issues involved with this amendment:

Issues for Fort Calhoun Technical Specification Change

Issue 1- low river level does not assure availability of raw water pumps to bring plant to cold shutdown conditions

The licensee is proposing a TS surveillance requirement for low level of 976 feet 9 inches of the UHS – Missouri River. The licensee states a river level of 976 feet 9 inches is required for raw water pumps minimum submergence level. The raw water pumps provide the cooling medium for the component cooling water heat exchangers, which are used to cool the plant down to cold shutdown. If the pumps are lost, then the plant will not have a normal means to achieve cold shutdown, nor will it have a safety-related means to obtain cold shutdown. In addition, raw water is credit as an emergency makeup source to the emergency condensate storage tank as a source of water for auxiliary feedwater pumps that maintain the plant in a hot shutdown condition. 10 CFR 50.36 requires the licensee establish limiting conditions for operation, and when they are not met, the licensee shall shut down the reactor. The licensee is proposes a TS limit on low river level at the point the raw water pumps become unavailable for use; however, the raw water pumps are required to shut down the reactor. Therefore, the licensee is proposing an action that could not be accomplished.

Issue 2- removing high river level limit no longer gives assurance the plant can safely shut down prior to losing equipment at intake structure

The licensee states that flooding at Fort Calhoun is highly unlikely, therefore, the licensee requests to remove the provision in TS requiring a plant shut down if the Missouri River level reach 1009 feet. The licensee states that a high river level technical specification does not meet the requirements of 10 CFR 50.63. However, 10 CFR 50.63(c)(2)(ii)(D) *Criterion 4* states "a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." In addition, 10 CFR 50.36(c)(2)(ii)(B) requires the licensee to establish a TS limiting condition for a operating restriction that is a initial condition of a design basis accident or transient analysis. Based upon recent operating experience, the staff regards susceptibility to flooding as risk to the plant's ability to maintain safe shutdown capabilities. A TS provision for high river level needs to be established in which the plant can safely shutdown using normal operating equipment and procedures prior to the loss of this equipment from flooding conditions.
What is the latest data on river level peaking for the Missouri River at the Ft Calhoun location? Does the latest data on river level support current high level setpoint?

Issue 3 – how is the licensee capturing the uncertainty in the instrumentation used for monitoring river temperature and level

In the FCS USAR, Appendix G, Criterion 12, the licensee states instrumentation conforms to the applicable Institute of Electrical and Electronics Engineers (IEEE) standards. The licensee is proposing to add a new TS surveillance to limit the UHS temperature to 90°F. The licensee credits existing instrumentation to monitor river temperature. The licensee states that it will encompass the loop uncertainty for UHS temperature in the surveillance test. Also, licensee proposes to add a TS surveillance to limit the UHS to a river level of 976 feet 9 inches. However, the license states its current river level instrumentation is not very accurate and intends on replacing the river level instrumentation with a more reliable system. The licensee designates a TS surveillance limit for the river water maximum temperature and minimum level at the assumed limit used in their design basis calculations showing sufficient heat removal in order to mitigate a design basis accident. The licensee needs provide the methods used for monitoring river water temperature and level to comply with their current licensing basis.

<u>Issue 3 – is there an adequate basis for 90 degree limit for the containment analysis</u>

Details can be supplied by Containment and Ventilation Branch

<u>Issue 4 – Has the calculation for the CCW heat exchanger's heat removal capability to supply 160 degree return water with 90 degree river water, been reviewed by the NRC staff?</u>

George, Gerond

From: Clark, Jeff
Sent: Thursday, January 13, 2011 8:05 AM
To: Famholtz, Thomas; George, Gerond
Subject: FW: Op Eval
Attachments: Operability Evaluation 2010-6635.pdf

Please see the attached Op Eval from FCS. I think they are pushing heavy on that "Easy Button."

Jeff

From: Kirkland, John
Sent: Thursday, January 13, 2011 7:56 AM
To: Clark, Jeff
Subject: Op Eval

This is the op eval we talked about yesterday.

John Kirkland
Senior Resident Inspector, Fort Calhoun Station
9610 Power Lane
Blair, NE 68008
402-426-9611
402-426-9613 (fax)

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Operability Evaluation Form

CR No: 2010-6635	
Initiating Event: (Maint., Surveillance, etc.) NSRG Review	Date: 12/15/10

I. Affected Item (SSC): Equipment Tag No: AE-AUX, AE-AUX-RCA (Equipment in the Aux. Building)

II. Describe the **OPERABILITY CONCERN** associated with the affected item, including a discussion of the requirement or commitment established for the item and why the requirement or commitment may not be met:

Leakage through penetrations in the Auxiliary Building envelope during the design basis flood (1014') may affect the operability of safety related equipment in the Auxiliary Building.

III. Describe the **INTENDED SAFETY FUNCTION(S)** of the affected item (See Section 6.3.3):

The purpose of the Auxiliary Building exterior walls with respect to external flooding is to protect safety related equipment housed within from the design basis flood of 1014' [1].

IV. Determine relevant industry and plant information (CRs, Work Management maintenance information, NRC Information Notices, Generic Letters, Operating Experience)(See Section 6.3.4):

- CR 2009-4166: This CR identified the unsealed fire piping penetrations below the design basis flood level in the Intake Structure.
- CR 2010-6485: Identified a fuel oil piping penetration in the West wall of the Intake Structure that was supposedly not sealed. The associated QP-31 determined that a seal was in place and was of sufficient quality to perform its design function.
- CR 2010-5211: This CR documents that the fire line test penetrations AE-13A and AE-13B in the West wall of the Intake Structure were not sealed with a L-CQE sealant. The associated QP-31 determined that the seals were of sufficient quality to perform their design function.

V. List references used (See Section 6.3.5):

[1] USAR – 2.7 Rev. 9, *Hydrology*

[2] SO-G-103 Rev. 25, *Fire Protection Operability and Surveillance Requirements*

[3] Drawing A-5992 Rev. 66 (File 41970), *Fire Barrier Penetration Schedule*

[4] PED-MWP-21 Rev. 3, *Fire Penetration Seals Installation and Repair*

[5] EA0 -010 Rev. 0, *Internal Flooding*

[6] EA9 -005 Rev. 5, *Fire Barrier Evaluation for 86-10 Miscellaneous Penetrations*

[7] EA9 -002 Rev. 5, *Fire Barrier Evaluation for Cable Tray*

[8] EA98-004 Rev. 2, *Fire Barrier Evaluation for Miscellaneous*

[9] Drawing D-4720 Rev. 1

[10] Drawing B-4341 Rev. 2

[11] Drawing B-4340 Rev. 5

[12] Drawing B-4339 Rev. 8

[13] Drawing B-4347 Rev. 6

[14] Drawing B-4345 Rev. 7

[15] Drawing B-4344 Rev. 9

[16] Drawing B-4342 Rev. 8

[17] Drawing B-4348 Rev. 3

[18] Drawing B-4337 Rev. 4

[19] Drawing D-4728 Rev. 1

VI. Are the Technical Specifications being met (See Section 6.3.6):

There are no Technical Specifications directly associated with these seals; however, as discussed below in Section VII, the seals are capable of protecting safety related equipment in the Auxiliary Building. The Technical Specifications are being met.

CR No: 2010-6635

VII. Justification of Decision (See Section 6.3.7 - Include aggregate effect to existing degraded conditions) (Attach additional material if needed):

The first step in this operability determination is to identify all of the relevant penetrations in the envelope of the Auxiliary Building. Penetrations will be considered if they are above grade and below 1014'. Penetrations below grade are not expected to leak significant amounts of water during the flooding event. Larger openings in the building such as doorways are not being considered in this Operability Evaluation because qualified protections are already in place to ensure that these openings are protected.

SO-G-103 [2] identifies that all of the exterior walls of the Auxiliary Building relied upon for flood protection are also fire barriers. Therefore, any penetrations through these walls are sealed for fire protection purposes, and the method/type of seal used is cataloged and controlled. A spreadsheet maintained by design drafting [3] is used for configuration control of the fire barrier penetrations. A review of the spreadsheet revealed that the relevant penetrations conform to one of 9 types of standard penetration seals or are directly addressed in one of three Engineering Analyses. Each seal type has a typical drawing detail which includes typical materials used; furthermore, the installation of these seals is controlled by PED-MWP-21 [4] which contains numerous QC hold points to ensure that the seals are installed per the typical drawings. It is recognized that these penetration seals have not installed to be L-CQE for the purpose of flood protection; however, the quality control used to ensure that the penetrations meet fire protection requirements is sufficient to ensure that the seals were installed in accordance with their associated drawing and that the correct materials were used. This Operability Evaluation will assess the adequacy of the sealing materials and configurations for the purpose of flood protection.

The layout drawings for the penetrations were reviewed to determine seal elevations. The lowest relevant barrier is above 1007'; therefore, the maximum hydrostatic pressure applied to these penetrations is about 3 psi.

The design basis flood resulting in a river elevation at the plant of 1014' requires the failure of an upstream dam. It is expected that the river elevation resulting from the dam failure would be transient in nature and the 1014' peak would only last a short time. A flood up to 1009.3' could result without dam failure; however, this flood scenario requires a probable maximum precipitation (PMP) event occur below Gavin's Point Dam. A PMP event drops massive amount of rain over a short period of time; therefore, the river level is expected to rise and recede quickly from this event also. For the reasons discussed above, the fire penetration seals are not required to be watertight for an extended period of time. Furthermore, it has been shown that the Auxiliary Building can accommodate a significant amount of water without affecting the operation of safety related equipment [5]. It has been shown that flood waters in the Auxiliary Building will generally flow to Room 23 which contains no redundant safety related equipment which could be affected by the flood water and can accommodate a volume of approximately 200,000 gallons [5].

What follows is a discussion of each type of relevant fire penetration seal type and why that seal type is expected to withstand the effects of the design basis flood. The seal types are identified by their typical penetration drawing. The discussion that follows is to be used in conjunction with the information given above on submergence time and applied hydrostatic pressure.

B-4337 & B-4338: These penetration types use grout as the main sealant. The grout is to be installed such that the annular space is completely filled and the minimum axial thickness of the grout barrier is to be 6". Grout is essentially the same as concrete except that it uses much finer aggregate; therefore, the grout will not degrade under submerged conditions. Penetrations of these types are expected to be watertight.

B-4347, B-4345, B-4344, B-4342, & 86-10/EA98-002: These penetration types use Dow Corning 3-6548 silicone RTV foam as the main sealant. The self expanding capability of this product ensures that the annular space of a penetration is completely filled and that no uninhibited leak paths exist. The silicone RTV itself is a well known and widely used water proofing material that is not expected to degrade when submerged. The Dow Corning published product information indicates that not only does the RTV foam produce a smoke and gas proof seal but it also can be used to "Seal building against damaging contaminants such as dirt, dust and water." Finally, a physical inspection of several barriers containing RTV foam was conducted. The foam had expanded to completely fill each of these penetrations; furthermore, when force was applied to the foam it remained bonded at the outer diameter of the penetration which indicates that the foam forms an acceptably strong bond with the penetration. Penetrations of these types are expected to be water resistant and capable of performing their design function with respect to external flooding.

B-4340, B-4339: Penetrations of this type must have and average annular gap no greater than 9/16". The annular gap is filled with a fire proof fiber and then sealed on both ends with either Dow Corning 96-081 RTV adhesive/sealant or Dow Corning 732 Adhesive/Sealant. Both of these materials are widely used and sealants and are well suited for this water proofing application. Seals of this type are expected to be watertight.

86-10/EA98-004: This is a series of penetration drawings representing each penetration on the south wall of the east switchgear room. These penetrations are physically plugged, sealed with grout, sealed with RTV foam or sealed with a combination of grout and RTV foam. For the reasons cited in the discussions above, these penetrations are expected to be water resistant and capable of performing their design function with respect to external flooding.

B-4341 & 86-10/EA98-005: Penetration No. 29-W-9 is the only relevant penetration that conforms to typical drawing 86-10/EA98-005. Limited information is represented on the typical drawing for this penetration. EA98-005 states, "Penetration 29-W-9 is a sleeved pipe penetration through the Auxiliary Building wall. The penetration is sealed with a boot on the Room 29 side. The penetration does not penetrate the Radwaste Building wall, but ends inside the 6" annular gap between the Radwaste and Auxiliary Buildings. The pipe is capped within the annular gap." Penetration No. 29-W-9 is directly adjacent to Penetration No. 29-W-10 (Drawing B-4341) and appears to be of similar construction. Both penetrations consist of a 1-1/2" pipe running through a 2" pipe sleeve. This leaves less than a 3/8" average annular gap. The annular gap of Penetration 29-W-10 is filled with a fibrous fireproof blanket material and capped with a fabric boot which is hose clamped to the pipe and sleeve. Penetration 29-W-9 has the same fabric boot on the interior wall; however, the typical drawing of the penetration does not provide any detail on how the annular gap is filled. Given the similarity and close proximity of the penetrations, it is reasonable to assume that Penetration 29-W-9 is also filled with a fibrous fireproof blanket. These penetrations are not expected to be watertight; however, given that the penetration will be under at most 3' of water, have a small annular gap and that the gap is filled, the leakage is expected to be insignificant. Furthermore, both of these penetrations are contained in Room 29, the VCT room, where there is no equipment required to maintain the plant in a safe shutdown condition. Water that may leak into Room 29 will then pass under the door and into Corridor 26, down the lift path from Corridor 26 to Corridor 4 and finally into Room 23. As discussed above, Room 23 has a significant volume that can safely retain flood waters. Safety related equipment is not expected to be affected by leakage through Penetrations 29-W-9 and 29-W-10.

For the reasons discussed above, the fire barrier penetrations in the exterior envelope of the Auxiliary Building are expected to perform their design function of protecting safety related equipment from flood waters to an elevation of 1014' without modification to plant operation.

VIII. Based on a review of the intended safety function(s) of the affected item and operability concern (See Section 6.3.2):

X	The item is **OPERABLE** without any modifications to plant operation, or
	The item is **OPERABLE, BUT**, restrictions must be placed on plant operation, to maintain operability (State restrictions in Section IX and finish this form then continue with an SAO)
	The item is **INOPERABLE (IMMEDIATELY NOTIFY THE Shift Manager)**

OPTIONAL

	Preliminary determination is as above. Further determination/evaluation must be made by NE. CR AI# has been issued (See Section 6.3.9).

IX. Restriction to maintain validity of this Operability Evaluation (See Section 6.3.8)

None

Prepared by: Anthony Filips (10 CFR 50.59 and/or 72.48 Screener Qualified)	Date: 1/3/11	Time: 12:30
Reviewed by: Travis Dendinger	Date: 1/3/11	Time: 12:31
(Independent Reviewer[1]) NA	Date:	Time:
Yes No	Date: 1/3/11	Time: 12:45
[1] (Department Supervisor) Interdisciplinary Review		
[2] ISI Coordinator: NA	Date: 1/3/11	Time:
[3] Approved:	Date: 1-3-11	Time: 1520

(Shift Manager) A 50.59/72.48 review **SHALL** be completed before an SSC is declared operable if Compensatory measures or restrictions are needed.

[1] Department Supervisor **SHALL** determine that an interdisciplinary review is performed by appropriate personnel (i.e., design engineering mechanical, design engineering electrical, system engineering, programs engineering, etc.). An interdisciplinary review is not always appropriate. If one is not needed the Department supervisor may so indicate. The independent reviewer **MAY** perform the interdisciplinary review.

[2] The ISI Coordinator's shall be the Reviewer **ONLY** for ISI-related SSC.

[3] The Shift Manager shall ensure that the on-shift crew is briefed on the Operability evaluation.

NOTE: THIS OPERABILITY EVALUATION MUST BE REVIEWED BY THE PRC AFTER IT IS APPROVED BY THE SHIFT MANAGER. PROVIDE ORIGINAL OPERABILITY EVALUATION TO PRC TECHNICAL SECRETARY.

Elliott, Robert

From: Brown, Frederick
Sent: Friday, June 10, 2011 9:59 AM
To: Andersen, James; Ashley, MaryAnn; Cartwright, William; Elliott, Robert; Franovich, Rani; Kobetz, Timothy; McHale, John; Pruett, Troy; Shoop, Undine; Thorp, John; Westreich, Barry
Cc: Giitter, Joseph; Howe, Allen; Salgado, Nancy
Subject: FW: POSTPONEMENT OF FORT CALHOUN 95002 SUPPLEMENTAL INSPECTION

FYI

From: Collins, Elmo
Sent: Friday, June 10, 2011 9:10 AM
To: Ruland, William
Cc: Boger, Bruce; Leeds, Eric; Brown, Frederick; Kennedy, Kriss
Subject: FW: POSTPONEMENT OF FORT CALHOUN 95002 SUPPLEMENTAL INSPECTION

Bill
As a follow up to our discussion – we decided to postpone the FCS 95002 – see below – we have informed the licensee

From: Virgilio, Martin
To: Howell, Art
Sent: Thu Jun 09 17:43:06 2011
Subject: RE: POSTPONEMENT OF FORT CALHOUN 95002 SUPPLEMENTAL INSPECTION

Thanks, Art

I support your proposal.

Marty

From: Howell, Art
Sent: Thursday, June 09, 2011 12:36 PM
To: Virgilio, Martin
Subject: POSTPONEMENT OF FORT CALHOUN 95002 SUPPLEMENTAL INSPECTION

Marty,

I have a call into you on this subject if you would like to discuss further. I wanted to give you heads up on a decision we made this morning to postpone the conduct of the Fort Calhoun 95002 supplemental inspection from this month until next month. The basis for the postponement is that our on-going onsite incident response activities for the rising river level and the switchgear fire are our priority focus areas. These areas will remain our principal priority at Fort Calhoun for the next few weeks during the period in which there is some uncertainty associated with the effectiveness of the licensee's flood protection measures as the Missouri River level continues to rise. We believe that we'll be in a better position to determine the efficacy of conducting a major team inspection activity concurrent with onsite incident response activities after the river crests and we have had an opportunity to gauge the overall conditions at the site and effectiveness of the licensee's actions. Based on that, we think we could be in a position to conduct the inspection about the middle of July.

Licensee senior management believes they are ready for the inspection and has indicated a strong desire (to Elmo yesterday while he was on site) for the Agency to perform the supplemental inspection this month as scheduled. I understand that earlier this year the timing of this inspection was raised by Senator Nelson of Nebraska when he met with the Chairman.

We plan to inform the licensee of our decision today.

Art

Smith, Chris

From: Melfi, Jim
Sent: Sunday, June 12, 2011 4:53 AM
To: Clark, Jeff; Smith, Chris; Sifre, Wayne
Cc: Kirkland, John; Wingebach, Jacob
Subject: RE: NRC "Concerns" and Monitoring

Jeff, et. al.

So far, touring the auxiliary building, the groundwater inleakage into the auxiliary building seems minor. One new leak identified in room 10 (989 foot elevation), but very minor seepage.

Basement of turbine building you seem more wall seepage, but it is still very manageable.

Licensee identified that manhole 3 in the radwaste building has water in it. Apparently, it contains electrical cables from offsite. Water level seems to be about 2 feet down from the 1007 foot elevation. Water level in the manhole appears to be about the same as flood level. They were going to try another pump in the manhole. The flood barriers between the auxiliary building and the radwaste building seem to be in place. Licensee actions, so far seem to be appropriate.

Some more seepage seen around the aquaberm, but the licensee is monitoring and taking actions.

JIM

From: Clark, Jeff
Sent: Saturday, June 11, 2011 8:53 PM
To: Smith, Chris; Melfi, Jim; Sifre, Wayne
Cc: Kirkland, John
Subject: ACT: NRC "Concerns" and Monitoring

Guys,

For the brief I am giving Elmo tomorrow evening, I have been asked to give the "NRC's concerns." Pleas provide a list to John of what you see as either concerns or questions about what the licensee is doing. Please give some context (e.g., this is someting I think could turn into a big problem, they are missing something, or this is an enhancement).

Also, I want to emphasize we are not in the inspection mode, we are in incident response. We are monitoring what the licensee is doing. We should be making comments, but not as if they are doing something wrong (like in a violation), but more to assess what they are doing and possibly help them to think through things. However, do not take your comments lightly. Make sure they get to the appropriate managers, such as the incident commander (Mike Firm). Don't be afraid to come back around and ask what they specifically did.

And finally, please spend a good deal of time out in the plant. You should be walking key items (e.g. aquadam, switchgear rooms, etc) multiple times on your shift. Just like having a plant issue during operations, don't be afraid to give me a call if you see something, or learn of something, you don't understand or you are not getting traction on.

The Region and I both feel you all are doing a great job at something that is highly important. Keep up the good work.

Jeff

Azua, Ray

From: Clark, Jeff
Sent: Saturday, June 18, 2011 9:04 AM
To: Kennedy, Kriss; Howell, Linda
Cc: Azua, Ray; Kirkland, John; Alferink, Beth; Alexander, Ryan; Hay, Michael
Subject: FYI: FCS Site Update <Sat - 6/18/11>

Kriss/Linda,

Conditions at the site degraded somewhat Friday evening and into today. We have experienced severe thunderstorms overnight, and showers/thunderstorms are in the forecast through Monday. Some of the issues we are currently monitoring:

1) Communications (both phone lines and PC connections) have been degraded. Service is intermittent. The 402-426-9611 is the worst; it has a lot of noise. We are attempting to have x9612 rerouted to give us a clear line. FCS telecommunications folks have been in this morning and are attempting to repair. However, they stated they are having a lot of problems around the site, since many of their junction boxes have been flooded. Leakage/flooding **could affect TSC, control room, and ERDS**. We do have cell phone coverage and both we and the licensee have satellite phones.
2) The licensee informed us they <u>will not</u> be repairing the Aquadam tear (NW side). The vendor indicated the repair could cause more damage if not properly installed. For some reason, the vendor is not available for some time. The licensee plans to continually refill this section as needed, and they have constructed a temporary sandbag "berm" to cover the tear.
3) The licensee contacted Cargill (north of plant) previously about flooding provisions and hazardous chemicals. They stated Cargill had taken provisions and only had sulfuric acid as a concern for hazardous materials. We asked about anhydrous ammonia (recalled this was stored in tanks there). The licensee responded that was not stored by Cargill anymore. Leonard Willoughby had nights last evening. He brought this up again to the licensee, with the same response. However, Leonard told them it was <u>Terra</u> (another company) that owned the anhydrous ammonia tanks. The licensee is now checking on this.
4) The licensee found a "soft spot" in the ground, just to the west of the Unit/Service transformers. Engineering attempted to probe this with a 2x4, and it sunk 3 feet into the ground. They are classifying this as a "sinkhole" or "sand boil." They are looking at what they could do to firm up or repair. We are also seeing several other potential soft spots as well, as a good bit of the PA is not paved.
5) The licensee is still battling several aspects involving security. The water flow through the underground cable vaults near the PAP continues. It has undercut some of the pavement, and is going under the Hasco barriers. They worked all yesterday to shore this up again. They also are exploring new pathways to get to some of their positions, as they would lose normal access at 1007 feet.
6) The licensee is still working on a number of issues we raised with them, including: how to get water into the reactor vessel if level exceeds 1014; the fuel oil transfer procedure; and procedures for use of the temporary emergency diesel generator.

Jeff

Elliott, Robert

From: Elliott, Robert
Sent: Monday, June 27, 2011 10:02 AM
To: Anderson, Shaun; Bucholtz, Kristy; Grover, Ravinder; Hamm, Matthew; Hemphill, Khadijah; Richards, Karen; Schulten, Carl; Singletary, Melana; Waig, Gerald
Subject: FW: Resend to include ADAMS # ML111770003, PNO-IV-11-003B Update: Fort Calhoun Station Devlaration of a Notification of Unusual Event Due to High River Level
Attachments: PNO-IV-11-003B ADAMS.docx

FYI, Fort Calhoun's difficulties are continuing....

From: Cunningham, Liza
Sent: Monday, June 27, 2011 7:11 AM
To: Auluck, Rajender; Boyce, Tom (RES); Brock, Kathryn; Campbell, Stephen; Carlson, Robert; Casto, Greg; Chernoff, Harold; Cranston, Gregory; Dennig, Robert; Dozier, Jerry; Eads, Johnny; Elliott, Robert; Franovich, Rani; Gavrilas, Mirela; Harrison, Donnie; Helton, Shana; Howe, Allen; Imboden, Andy; James, Lois; Kemper, William; Khanna, Meena; Klein, Alex; Kobetz, Timothy; Kulesa, Gloria; Lupold, Timothy; Manoly, Kamal; Markley, Michael; McHale, John; McMurtray, Anthony; Mendiola, Anthony; Mitchell, Matthew; Murphy, Martin; Pascarelli, Robert; Pelton, David; Pham, Bo; Raghavan, Rags; Rosenberg, Stacey; Salgado, Nancy; Scott, Michael; Shoop, Undine; Simms, Sophonia; Tate, Travis; Taylor, Robert; Thatcher, Dale; Thorp, John; Wilson, George; Wrona, David; Zimmerman, Jacob; Boger, Bruce; Givvines, Mary; Grobe, Jack; Leeds, Eric; Bahadur, Sher; Blount, Tom; Brown, Frederick; Cheok, Michael; Cunningham, Mark; Evans, Michele; Ficks, Ben; Galloway, Melanie; Giitter, Joseph; Hiland, Patrick; Holian, Brian; Lee, Samson; Lubinski, John; Lund, Louise; McGinty, Tim; Nelson, Robert; Quay, Theodore; Ruland, William; Skeen, David
Cc: NRR_DIRS_IOEB Distribution
Subject: FW: Resend to include ADAMS # ML111770003, PNO-IV-11-003B Update: Fort Calhoun Station Devlaration of a Notification of Unusual Event Due to High River Level

Attached in the PNO-IV-11-003B: **(UPDATE)** Fort Calhoun Station Declaration of a Notification of Unusual Event Due to High River Level.

Thanks,
Liza Cunningham

From: Tannenbaum, Anita
Sent: Sunday, June 26, 2011 3:42 PM
To: R4; PN_Distribution
Subject: Resend to include ADAMS # ML111770003, PNO-IV-11-003B Update: Fort Calhoun Station Devlaration of a Notification of Unusual Event Due to High River Level

Tracking:

Recipient	Read
Anderson, Shaun	
Bucholtz, Kristy	Read: 6/27/2011 10:04 AM
Grover, Ravinder	
Hamm, Matthew	
Hemphill, Khadijah	Read: 6/27/2011 11:59 AM
Richards, Karen	Read: 6/27/2011 10:47 AM
Schulten, Carl	Read: 6/27/2011 10:58 AM
Singletary, Melana	Read: 6/27/2011 10:13 AM
Waig, Gerald	Read: 6/27/2011 12:06 PM

June 26, 2011

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO-IV-11-003B

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region IV, Arlington, Texas, staff on this date.

Facility	**Licensee Emergency Classification**
	_X_ Notification of Unusual Event
Fort Calhoun Station	___ Alert
	___ Site Area Emergency
Docket: 50-285	___ General Emergency
License No. DPR-40	___ Not Applicable

SUBJECT: **(UPDATE)** Fort Calhoun Station Declaration of a Notification of Unusual Event Due to High River Level

DESCRIPTION: This Preliminary Notification (PN) updates information discussed in PNO-IV-003 and PNO-IV-003A. On June 26, 2011, at approximately 1:25 a.m., the water berm that had been providing supplemental floodwater protection to Fort Calhoun Station failed as a result of site activities. As a result, floodwaters have reached an elevation of 1006 feet 4 inches (mean sea level, MSL) around the auxiliary and containment buildings. Those buildings are protected by design to a floodwater elevation of 1014 feet MSL.

The failure of the water berm allowed floodwaters to surround the main electrical transformers, which prompted the licensee to take the precautionary measure of transferring from offsite power to onsite emergency diesel generators (EDGs). Reactor shutdown cooling and spent fuel pool cooling remain unaffected. The licensee is developing plans to restore the site to offsite power.

In response to this event, the NRC entered the Monitoring mode of Incident Response at 1:53 a.m. (CDT) and staffed the Region IV Incident Response Center.

Fort Calhoun Station remains in the Notification of Unusual Event (NOUE) that was declared on June 6, 2011.

Region IV received notification of this occurrence by phone from the augmented resident inspector staff at the site.

The States of Nebraska and Iowa have been notified.

The information presented herein has been discussed with the licensee and is current as of 10:00 a.m., June 26, 2011.

ADAMS ACCESSION NUMBER: ML111770003

CONTACTS: Wayne Walker, at (817) 320-3675

Smith, Chris

From: Haire, Mark
Sent: Monday, July 18, 2011 5:14 PM
To: Howell, Linda; Drake, James; Azua, Ray; Smith, Chris; Alexander, Ryan; Alferink, Beth; Smith, Rich; Baca, Bernadette; Rosebrook, Andrew; Denissen, Christie; Presby, Peter; OHara, Timothy; Newport, Christopher
Subject: FYI: FW: IC Turnover Sheet 7/18/11 Day Shift
Attachments: 7-18-11 Days.docx

From: PIER, DONALD M [mailto:dpier@oppd.com]
Sent: Monday, July 18, 2011 5:08 PM
To: BRANDEAU, JOHN F; BANNISTER, DAVID J; BAKER, RUSSELL J; BCP Finance - Administration; BCP Incident Commanders; BCP Logistics; BCP Operations; BCP Planning; BCP Public Information; FANSLAU, STEVEN A; MCCAW, GERALD D; MCCORMICK, KEVIN S; ZAVADIL, CLINTON J; doremuskr@inpo.org; FC CRS; FC POD; FC SM All; FCS LEADERS; GODFREY, MICHAEL E; KRAMER, BRIAN D; NISSEN, TIMOTHY J; R4 IRC; REINHART, JEFFREY A; R4 IRC; 'mark.haire@nrc.gov'
Cc: KREIFELS, BRADLEY E; MERRICK, DAVID E; MCMULLEN, CARLA L; MOELLER, CHRISTOPHER J
Subject: IC Turnover Sheet 7/18/11 Day Shift

Attached is the Incident Commander 7/18/11 dayshift report.

The Incident Commander is stationed M-F day shift.

Thanks

Don Pier

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2011 FLOODING INCIDENT COMMANDER'S REPORT

DATE: 7-18-2011	TIME: 1700	IC: Donald Pier	SHIFT: Day Shift

EMERGENCY PLAN CLASSIFICATION

Notification of Unusual Event (NOUE): Declared 6/6/11 0800; HU-1, EAL-5

SAFETY STATUS and ISSUES

<u>Nuclear Safety:</u>

SOPP Shutdown Condition: 1	ORAM Color: Yellow Due to bus outages	Time to Boil - Cavity: 38 hours Time to Boil - SFP: 80 hours

<u>Industrial Safety:</u>

1) Monitor work areas for heat stress and stay times per FCSG-15-8.
2) The two person rule is in effect for working or traversing in high flood waters.
3) Contact Mike Godfrey and Brian Kramer for all non-routine dayshift switchyard entries.
4) Maintenance Shop cross path secured with red tape due to column sinking. Requires bypassing through the Machine Shop. Safety Glasses required for passage. Safety glasses staged at both doors to the Machine Shop.
5) Do not park in or block reserved parking locations in the parking lot. Also, only on shift staff and personnel directly supporting the flooding efforts/priorities are to park in the new parking lot at the top of the hill. All others are to park at Blair High School and use the shuttle bus.
6) Additional opportunities for TDAP shots (tetanus, diphtheria and pertussis) are available at several Alegent facilities. Contact Fitness for Duty Specialist Michelle Bentley at ext. 3089 or Wellness Specialist Dave Williams at ext. 3374.

<u>Radiological Safety:</u> The portal radiation monitor at the south security exit point has been restored to service. Please ensure you stop and pause in the monitor on your way out of the Protected Area.

PLANT STATUS

345 kV Status: Available	161 kV Status: Available
DG1 Status: Available	DG2 Status: Available
Significant Bus / MCC Outages: Buses 1B4A, 1B3A, 1B3A-4A,	
Containment Closure Status: Still set for heavy loads	Containment Temp: 80 °F
Reactor/Cavity Water Level: 1035' 10.5"	River Level: 1006' 0" (AOP-01)

Significant Equipment Unavailable: AC-3B CCW pump, VA-64B Control Room Filter Fan and Inverter D bypass transformer, AC-5A, Spent Fuel Pool Pump. 'C' cell, AC-10D, FP-1B.

SIGNIFICANT PROTECTED EQUIPMENT

<u>Areas and Equipment protected for Spent Fuel Pool, Shutdown Cooling, and Containment Closure:</u>

- Room 5 – Spent Fuel Pool Cooling components
- Room 15A - Shutdown Cooling Components
- Room 69 – Component Cooling Water Pump AC-3C
- DG-1 Room 63 – Emergency Generator DG-1
- Room 21 - Shutdown Cooling
- Room 13 & 14 – Shutdown Cooling Components
- Room 18 – Raw Water/CCW Heat exchanger AC-1D
- East & West Switchgear Room – Bus 1A3
- VA-3A, Containment Filtering Unit
- Intake Structure – Raw Water Pump AC-10D, "E" gate/screen, AC-12B RW Strainer, FP-1B, Diesel Fire Pump, CW-16B
- Corridor 26 by Boric Acid Tanks – Boric Acid and Makeup Flow paths (includes MCCs)
- Corridor 4 - Boric Acid and Makeup Flow paths (includes MCCs)
- Containment - Boric Acid and Makeup Flow paths
- Upper Electrical Penetration Room - Boric Acid and Makeup Flow paths (includes MCCs)
- Switchyard 345 KV Breakers 5 and 6
- Transformers T1A-1 and T1A-2
- Aqua Dam Protected to 6'- Spotter required within 6'. Pump crews and inspections are exempt. IC and SM approval required.

- Cleaned Turbine building portable toilets, sealed for future use.
- Continued cleanup inside PA (debris removal, sandbag consolidation, flood equipment storage).
- Filled and walked down Aqua berms.
- Started work on 'C' cell to remove sand

- Monitor water intrusion inside the security building area communications room. (Maint/NPS)
- Coach new pump and aqua dam monitoring team members on their duties and safety precautions. Ensure adequate monitoring crew resources as they transition to new personnel this next week. (Support Coordinator)
- The breaker for MCC-4A2 temp mod did not arrive. Engineering needs to evaluate the use of a breaker in the warehouse or other timely solution to complete the temp mod. The replacement breaker ordered is not ready for shipment. Proc. Engineering needs to confirm delivery date of replacement breaker. (Engr)
- Perform safety observations. (Support Coordinator/IC)
- Check WBGT during the hot parts of the day. Specifically check in direct sunlight to ensure correct stay times. (Support Coordinator)
- Aqua dam team evaluated the air bubble on piece 1 in the PA new information, direction on venting Tuesday. The aqua dams need filling as required (Support Coordinator, Kevin Smith/David Ortiz),
- Cell work de-sanding continues.

NEAR TERM STATION PRIORITIES

TASK	LEAD WORK GROUP	EXPECTED COMPLETION DATE	COMMENTS
Evaluate spare breaker in the warehouse for use in MCC-4A2	DEN	July 20	Need to determine timely success path for MCC-4A2.
Confirm date of delivery of breaker for MCC-4A2 temp mod.	Procurement Engineering	July 20	The breaker did not arrive and a delivery date needs to be determined and temp mod schedule confirmed.
Install Temp. Mod. for MCC-4A2	EM	July 19	*Date is in jeopardy. EC 53319 Parts hold. Breaker WO 417046
Transport 16 cases of paper to Admin Building.	Maint	July 19	Paper is to arrive Tuesday and needs to be put in Admin Building. Contact Cathy Geren x7241.
Maintenance Bldg Support Column	DEN	July 20-PRC Approval of TM July 29 Installed	Temp Mod 53436 to install 6 shoring columns. EC 50986 for permanent repair during site recovery.
Perform Operability evaluation of Stainless Steel sample vessel that dropped into FO-1	DEN	July 21	CR 2011-6050-8
VA-89/90: Sandbag/berm protection will be installed for immediate protection (2-3 Days); Remove, restore and elevate contingency (10-14 days after EC issued).	SFM	July 22	EC 53400 being processed by DEN for the elevated stands. Issue EC as contingency action.
Jim Shores needs EM support for pulling new phone cable into the Security Building	Comm/EM	July 27	WR 166809/WO 417997; Cable has arrived and staged in security bldg.

FLOOD BARRIERS STILL PENDING

Flood barriers installed per PE-RR-AE-1001, with the following deviations:
AE-22- 1007-1 door to Radwaste Building (power door) is removed. NOTE: NO POWER from MCC-4A2.
AE-23- 1007-9 door from RCA to Chemistry is removed.
AE-24- 1007-19 door – normal RP access point is removed.
AE-25- 1011-1 door from Turbine Mezzanine to Coordinator 52 is removed.
AE-28-1011-4 south door switchgear room for emergency switchgear exit
AE-21- Intake structure "trash rack" outlet is removed for cell cleaning work.
SD-127, 128 – Maintenance Shop drains to Lift Station #1 - open
VD-681, 682 – Switchgear Room HVAC drains to TB Sump - open
Other 1007' barriers can be removed to allow work access with authorization from SM. All 1007' barriers will be re-installed when river level reaches 1006' 9" and rising, or when directed by the SM.

ACTIVE HITS

- Level A CR 2011-5414: Loss of 1B4A – Owner: Steve Clayton. RCA in progress.
- Intake Screen sanding in issues – HIT formed under CR 2011-5750 – Ken Erdman lead.

NOTES: (Topics to Discuss, Actions to Complete, Kudos to Share)

- Spare B.5.b Fire Truck from Fort Calhoun Volunteer Fire Dept located onsite. Supply hose in rack along walkway to King Tut blocks
- Bottled water is stored in a sealand container at the top of the hill. Bring into PA as needed.
- Shift Manager and IC will continue to monitor river levels and utilize AOP-1, EPIP-TSC-2 and PE-RR-AE-1001 as the principle governing documents. NOTE: New Revision to EPIP-TSC-2 issued on 7-01-11 for alternate filling of D/G fuel oil.
- If communications or ERDS is lost in the Control Room or TSC; Incident Commander is to call NRC Senior Resident Inspector and the NRC operations center (per AOP).
- Use controls in place for boat safety – Need supervisor brief before getting boat keys from Incident Commander.
- Spare gas generators are staged on the Turbine Deck. Spare gas pumps are in the Maintenance Shop.
- The current National Weather Service prediction for Blair river level is a rise of about 5" between now and Thursday night (7/21). Based on trending as the river went up, we may not see this full rise at Fort Calhoun but should expect some river level rise during the week.

Mizuno, Geary

From: Mehrhoff, Vivian
Sent: Thursday, September 29, 2011 9:16 AM
To: Blount, Tom
Subject: Emailing: Dam Backfit Panel Charter.docx
Attachments: Dam Backfit Panel Charter.docx

The message is ready to be sent with the following file or link attachments:

Dam Backfit Panel Charter.docx

Note: To protect against computer viruses, e-mail programs may prevent sending or receiving certain types of file attachments. Check your e-mail security settings to determine how attachments are handled.

From: Wilkins, Lynnea
Sent: Wednesday, October 12, 2011 1:55 PM
To: Mensah, Tanya; Smith, Edward; Li, Yong; Uribe, Juan; Haire, Mark; Holian, Brian; Rosenberg, Stacey; Goel, Vijay; DLRCalendar Resource; Hoang, Dan; Hair, Christopher; Wilson, George; Markley, Michael; Murphy, Martin
Subject: RE: Continuation: Internal PRB Meeting: G20110492/G20110506 - Fort Calhoun/Cooper Petitions Re: Flooding (ME6622 & ME6681)

All,

Please see the attached for tomorrow's meeting. I've update the Internal PRB notes based on our last meeting. I've also attached a "thumbnail" of Mr. Saporito's concerns as expressed in the teleconference (ML11256A036).

Thanks
Lynnea

Cooper Internal PRB Notes - G2... Fort Calhoun Internal PRB Not... Saporito Concerns From Transcr...

-----Original Appointment----
From: Mensah, Tanya
Sent: Friday, September 23, 2011 4:22 PM
To: Mensah, Tanya; Wilkins, Lynnea; Smith, Edward; Li, Yong; Uribe, Juan; Haire, Mark; Holian, Brian; Rosenberg, Stacey; Goel, Vijay; DLRCalendar Resource; Hoang, Dan; Hair, Christopher; Wilson, George; Markley, Michael; Murphy, Martin
Subject: Continuation: Internal PRB Meeting: G20110492/G20110506 - Fort Calhoun/Cooper Petitions Re: Flooding (ME6622 & ME6681)
When: Thursday, October 13, 2011 2:45 PM-3:30 PM (GMT-05:00) Eastern Time (US & Canada).
Where: HQ-OWFN-11B02-12p

When: Thursday, October 13, 2011 2:45 PM-3:30 PM (GMT-05:00) Eastern Time (US & Canada).
Where: HQ-OWFN-11B02-12p

Note: The GMT offset above does not reflect daylight saving time adjustments.

~~*~*~*~*~*~*~*~*

Purpose: The PRB will continue its internal discussion to make the initial recommendation to accept/reject the petition for review. Due to the schedules of the various PRB members and advisors (i.e, training, travel, AL), the earliest time to permit PRB participation from is 10/13/11.

Handouts: Will be provided by Lynnea via separate email.

Dial-In: Will Be Provided

Tanya Mensah, 2.206 Coordinator

301-415-3610

Jones, Bradley

From: Mehrhoff, Vivian
Sent: Thursday, October 20, 2011 3:15 PM
To: Blount, Tom; Kellar, Ray; Farnholtz, Thomas; Clark, Jeff; Wilson, George
Cc: Collins, Elmo; Loveless, David; Vegel, Anton; Jones, Bradley; Wilson, George
Subject: Charter for Backfit Panel on Postulated Failure of Upstream Dams Affecting Fort Calhoun Station - ML11293A198
Attachments: Charter for Backfit Panel on Postulated Failure of Upstream Dams Affecting Fort Calhoun Station.docx

Charter for Backfit Panel on Postulated Failure of Upstream Dams Affecting Fort Calhoun Station – ML11293A198

Vivian L Mehrhoff
Division Secretary
Division of Reactor Safety
Region IV - Arlington, Texas 76011
Phone: 817-860-8166 FAX: 817-860-8212
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

"The Perfect Human has a Man's Strength and a Woman's Compassion!"

Boyer, Rachel

From: Collins, Elmo
Sent: Tuesday, November 01, 2011 8:59 AM
To: Borchardt, Bill
Cc: Virgilio - Disabled 5-4-2012 per 574504, Martin
Subject: Ft Calhoun

Bill

Last Friday during my plant update with the Chairman, we discussed Ft Calhoun –
Key points:

1. CAL for post-flooding test/inspections needed before restart – schedule shows January
2. Col. 4 – licensee beginning review/assessment in prep for 95003
3. Plant shutdown greater than 6 months – we are using MC 0351 for PI and baseline inspection program impacts
4. Region IV conducted event f/u inspections after the water went down for Alert/fire in electrical switch gear, SGI unattended, EP response to Alert – there appear to be several additional issues, potentially greater than green – accordingly – Region IV in conjunction with NRR is entering a decision process of whether or not to use MC 0350 – this would allow us to put performance issues in CAL
5. I formed a backfit panel, in conjunction with NRR, to consider what actions are appropriate for an Army Corp of Engineers report addressing dam failures on the Missouri River system

We can discuss if you need

Elmo

Boyer, Rachel

From: Wiggins, Jim
Sent: Tuesday, December 06, 2011 8:06 AM
To: Borchardt, Bill
Cc: Virgilio - Disabled 5-4-2012 per 574504, Martin; Collins, Elmo; Lubinski, John
Subject: Fw:
Attachments: Fort Calhoun Station - talking points.docx

Here's a 1-pager to facilitate a walk-around among the Cnrs re: Ft Calhoun 0350 decision.

We'll await feedback before launching on the plan. First stop externally should be a call from you, Marty, Elmo or me to Gary Gates.

From: Lubinski, John
To: Wiggins, Jim
Cc: Pruett, Troy; Collins, Elmo; Kennedy, Kriss; Brown, Frederick; Evans, Michele; Hilton, Nick; McGinty, Tim; Boger, Bruce; Balazik, Michael; Roche, Kevin; Franovich, Rani; Clark, Jeff
Sent: Tue Dec 06 07:59:52 2011
Subject: RE:

Jim,

The attached incorporates your recommended changes and other adjustments. We deleted the bullet you highlighted and instead modified the subsequent bullet ("The effort the licensee has put forth has been narrowly focused on the specific issues rather than looking at broader performance issues"). This message was communicated to the licensee by Region IV.

From: Wiggins, Jim
Sent: Tuesday, December 06, 2011 7:26 AM
To: Lubinski, John
Cc: Pruett, Troy; Collins, Elmo; Kennedy, Kriss; Brown, Frederick; Evans, Michele; Hilton, Nick; McGinty, Tim; Boger, Bruce; Balazik, Michael; Roche, Kevin; Franovich, Rani
Subject: RE:

The language in this bullet is fairly extreme. Licensee has put forth minimal effort to evaluate extent of condition and extent of cause with more effort on flood recovery
Have we said this to the licensee? Need to plan on this document seeing the light of day.

Also, add as the first bullet on initial steps – Develop a Communications Plan to inform internal and external stakeholders on the reasons for the 0350 decision and its implecations on the licensee and NRC.

Please make some adjustments and finalize, then I'll send it up to Bill B later this morning.

From: Lubinski, John
Sent: Tuesday, December 06, 2011 7:16 AM
To: Wiggins, Jim
Cc: Pruett, Troy; Collins, Elmo; Kennedy, Kriss; Brown, Frederick; Evans, Michele; Hilton, Nick; McGinty, Tim; Boger, Bruce; Balazik, Michael; Roche, Kevin; Franovich, Rani
Subject:

14

Jim,

Attached are the talking points you requested for the Fort Calhoun transition.

Please let me know if you need any additional information.

Fort Calhoun Station (FCS) assessment of transition into IMC 0350 process

- Licensee initially shut-down for a refueling outage, but has remained shut-down because of flooding issues
- A CAL is in place to address mitigating flooding and technical issues
- High probability of 3 degraded cornerstones (IE, MS, Sec) with a potential for a 4th (EP) because of several potential significant findings (> green)
- The effort the licensee has put forth has been narrowly focused on the specific issues rather than looking at broader performance issues
- Current gap for entry into 0350 process is the existing CAL would need to be modified to address additional performance issues
- IMC 0350 allows NRC to implement inspections to address deficiencies to ensure adequate safety for plant start-up/operation
- Assuming no additional major modifications or program changes, Region IV thinks the earliest restart date is June 2012 due to its estimate of the additional work needed by the licensee to address all performance issues.

Initial Steps into 0350

- Modify CAL to address additional performance deficiencies
- Charter for 0350 Oversight Panel
- Determine inspection plan to address all performance deficiencies, including extent of condition and causes

Other

- Develop communication plan
- Determine enforcement/assessment for current and additional deficiencies identified
- Transition from 0350 back into the Action Matrix(IMC 0305)

Murphy, Martin

From: Murphy, Martin
Sent: Thursday, December 08, 2011 2:17 PM
To: Matharu, Gurcharan
Cc: Wilkins, Lynnea; Uribe, Juan; McConnell, Matthew; Andersen, James; Hoang, Dan
Subject: FW: 2011 Flooding of Fort Calhoun

FYI – see below from EMCB

From: Uribe, Juan
Sent: Thursday, December 08, 2011 2:13 PM
To: Murphy, Martin
Subject: RE: 2011 Flooding of Fort Calhoun

Gurcharan

The FCS DBF is 1006' based on the UFSAR, but the site is protected up to 1009' with flood gates and up to 1013' with special provisions, as also stated in the UFSAR.
I believe the max height of the flood waters was 1008'. Therefore, the site DID NOT exceed their max DBF of 1013'.
Lynnea can also confirm this since she is the FCS pm.

Keep in mind that changing the DBF is "voluntary" at this point and at the discretion of the licensee. For example, TMI reviewed and updated their FSAR DBF levels after they competed a study and FCS also recently updated their UFSAR based on studied by the US Army Corps of Engineers. So the response is yes, the sites still rely on their 100 year flood data for their DB. However, the regions are leading an effort of exploring data at several sites to verify if they are adequately protected and if further review should be done

I cannot confirm that ALL plants on the Mississippi have been reviewed (I don't know the answer to this question) but some of the, like Cooper on the Missouri River have been analyzed.

From: Khanna, Meena
Sent: Thursday, December 08, 2011 12:42 PM
To: Matharu, Gurcharan
Cc: Uribe, Juan; Hoang, Dan; McConnell, Matthew; Andersen, James
Subject: FW: 2011 Flooding of Fort Calhoun

Singh,
Pls note that I have asked Marty to reach out to you...the appropriate staff would be Dan and Juan...thanks!
Meena

From: Khanna, Meena
Sent: Thursday, December 08, 2011 12:05 PM
To: Murphy, Martin
Subject: FW: 2011 Flooding of Fort Calhoun

Marty,
Can you or one of your staff members pls reach out to Mr. Matharu.
Thanks,
Meena

From: Matharu, Gurcharan
Sent: Thursday, December 08, 2011 10:02 AM
To: Khanna, Meena
Cc: Andersen, James; McConnell, Matthew
Subject: 2011 Flooding of Fort Calhoun

Hi Meena,
I am reviewing a LAR for Prairie Island which has a EDG fuel oil requirement for 14 day operation of EDGs based on site flooding.
As a consequence of the 2011 floods, I believe Fort Calhoun exceeded their Design Basis Flood. Did the Agency take any steps for reviewing design basis (flooding) for plants on the Mississippi River? Are we still relying on 100 year flood data for DB?

Singh

Boyer, Rachel

From: Franke, Mark
Sent: Friday, December 09, 2011 12:14 PM
To: Virgilio - Disabled 5-4-2012 per 574504, Martin; Borchardt, Bill
Cc: Collins, Elmo; Pruett, Troy; Kennedy, Kriss; Bowman, Gregory; Brown, Frederick; Zimmerman, Roy; Clark, Jeff; Holahan, Gary; Ash, Darren
Subject: Chairman requested Monday briefing on Fort Calhoun Comm Plan, RIV/EDO/OPA/OCA

Region IV briefed the Chairman this morning on the status of Fort Calhoun and MC 0350, and there was one action item.
The Chairman requested a separate high level briefing Monday, if ready, to discuss the Communications Plan. He suggested EDO, OPA and OCA participation. He mentioned the upcoming Congressional hearings.
The Chairman's assistants are working with the offices to find a time. I let them know we preferred early afternoon rather than morning.
I will work with Region IV to see if we can set up a TA brief.

Boyer, Rachel

From: Holahan, Gary
Sent: Friday, December 09, 2011 12:10 PM
To: Collins, Elmo; Virgilio - Disabled 5-4-2012 per 574504, Martin
Cc: Johnson, Michael; Kotzalas, Margie; Brown, Frederick; Pruett, Troy; Ash, Darren; Borchardt, Bill; Taylor, Renee; Cianci, Sandra
Subject: Chairman briefing on Ft Calhoun MONDAY

Elmo,

After the conference call today, the Chairman asked for a meeting/conference call on Monday with the EDO (Marty will be acting), Region IV and NRR on the Ft Calhoun 0350 Communications Plan.

Since Reg. IV is developing the Communications Plan, I'm reluctant to put the meeting on the Chairman's and EDO's calendars until you think that you will be ready to support it.

Please let the EDO's office or Marty's office know when on Monday you can support the Communications Plan discussion. I have already called the EDO's office to let them know that a meeting sometime Monday will be needed.

If you need help getting the meet set-up, let me know.

Gary

Uribe, Juan

From: Wang, Weijun
Sent: Thursday, December 15, 2011 1:42 PM
To: Xi, Zuhan; Candelario, Luissette; Uribe, Juan
Subject: FW: Fort Calhoun Flooding

This is one of the earliest e-mail communication regarding the Fort Calhoun site flooding.

Weijun

From: Wang, Weijun
Sent: Tuesday, September 06, 2011 12:37 PM
To: Williams, Megan; Wilson, George
Cc: Manoly, Kamal; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Megan:

I am not sure whether we have regulatory authority to ask the licensee to monitor cracks – Kamal may know more.

Regarding the soil types and properties, you may want to get the soil profile and soil properties to see if there are clayey soil and cemented sandy soil because the clayey soil may cause additional long term settlement if it became saturated from unsaturated state for a while, and the cemented sandy soil may greatly reduce its strength when becomes saturated (non-cemented sand does not have this issue, saturation only reduce the effective stress and it should be considered during design).

By the way, usually cracking is an indication of differential settlement. If the cracks continue increasing, then local foundation failure is possible.

Please let me know if you have any questions. Thanks.

Weijun

From: Williams, Megan
Sent: Tuesday, September 06, 2011 11:53 AM
To: Wang, Weijun; Wilson, George
Cc: Manoly, Kamal; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Thank you, Weijun. This is most helpful.

We are seeing many cracks in concrete walls in the turbine building (below grade), which have been leaking water since the flood started. I am trying to find out the Structures Monitoring Program owner, to see if they had a baseline inspection documentation (they should have completed for license renewal ~2004) indicating what cracks were evident before the flood, and their size, etc.

Do we have any kind of regulatory authority to get them to monitor these cracks more quantifiably (i.e. crack monitors, etc.)? Should we be assessing the soil type and saturation now as well as susceptibility from its original state? Would that information be helpful?

R/,

Megan

From: Wang, Weijun
Sent: Tuesday, September 06, 2011 9:56 AM
To: Williams, Megan; Wilson, George
Cc: Manoly, Kamal; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Megan:

Without knowing much details, I'd like to suggest the follows:

1. Get documents to see how the hydraulic loading was considered during the original structure and foundation design. If the actual flood level is higher than the original design, then the additional lateral pressure and uplift force may cause some damage to the foundation walls and foundation floor concrete slabs, and also may have negative impact on the stability of foundation soils.

2. Regardless the flood levels considered in the original design, you may still need to inspect the structure and foundation to see if there is any damages caused by flooding, such as cracks and settlement. The GPR is a good method to detect voids and the licensee should also perform additional NDT testing to inspect the integrity of the foundation walls and floor concrete slabs or mats, should cracks be discovered.

3. Pay attention to settlements, both in vertical and horizontal directions. Flood normally will reduce the strength of foundation soil, especially if the water did not dissipate for a longer period of time. For certain type of soils and drainage conditions, as well as the actual foundation condition after the flood, the additional settlement caused by flood may continue for certain period of time, and therefore the settlement monitoring should be kept for a longer time until no detectible settlement increase is observed.

The above just for your reference. Please let me know if you have questions.

Thanks.

Weijun
(301)415-1175

From: Williams, Megan
Sent: Tuesday, September 06, 2011 9:44 AM
To: Wilson, George
Cc: Manoly, Kamal; Wang, Weijun; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Good morning,

I am at Fort Calhoun this week, and trying to get some details from the contractors on their approach to evaluating subgrade conditions at the site, now that most of the water has receded.

In reviewing the USAR, certain structures had hydraulic loading designs based on different flood levels (Class I versus Class II, and several references to 2.7.1.2 for design peak flood elevation, which itself references multiple different flood elevations within its paragraphs). Is it possible to get documentation that we have regarding this hydraulic loading design?

I am trying to get information on the GPR they are using to look for voids. Are there any other specific questions or things you all can think of that I should look at while on site?

Thank you,

Megan Williams
RIV

From: Wilson, George
Sent: Tuesday, August 23, 2011 5:53 AM
To: Williams, Megan
Cc: Manoly, Kamal; Wang, Weijun; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Megan use Kamal Manoly as your reference person he will get assistance from Weijun Wang in NRO

From: Williams, Megan
Sent: Friday, August 19, 2011 12:28 PM
To: Wilson, George
Subject: Fort Calhoun Flooding

Hey, Mr. Wilson,

I left you a voicemail, and understand you are out of the office until next week, but I thought I would also send you an email, since I will be out of the office next week on inspection.

We are beginning to engage in reviewing the licensee's efforts for restart at the plant after extensive flooding this spring/summer. You probably know a large portion of the plant is still under water, but they have engaged a consultant to start assessing geotechnical conditions around the site. I am looking for resources that can help us know what to look for in these assessments – do we have any history of plants in the agency recovering from this sort of water conditions? Do you know of any references that would tell us what kinds of tests of studies should be completed to assess the condition of the soils, etc.?

~~I appreciate any guidance you have in this area.~~

r/,

Megan Williams

Uribe, Juan

From: Wang, Weijun
Sent: Thursday, December 15, 2011 1:44 PM
To: Xi, Zuhan; Candelario, Luissette; Uribe, Juan
Subject: FW: Fort Calhoun Flooding
Attachments: Picture 001.jpg; Picture 002.jpg; Picture 003.jpg; Picture 004.jpg; Picture 005.jpg; Picture 006.jpg; Picture 007.jpg; FCS settled column.jpg

FYI.

Weijun

From: Williams, Megan
Sent: Wednesday, September 07, 2011 4:48 PM
To: Wang, Weijun; Wilson, George
Cc: Manoly, Kamal; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Good afternoon. I wanted to let you know about developments today.

We met with HDR (geotechnical investigation sub to the licensee) to discuss their approach, tasks to date, etc. They are in the process of getting us a good amount of requested information, such as original design documents, etc. They have yet to characterize the current condition of the soil, but strongly suspect it is still completely saturated.

As they remove mud/silt from the site with bobcats, an exterior walkway slab showed settlement in one corner, and also a large void where the concrete completely gave way. (see attached). There is also a column that has settled, and it is taking the adjacent masonry walls with it (stepped cracking) – you may have seen this photo before.

Again, I will pass along information as I receive it, but if you think of anything I should be asking for or looking at, I appreciate any guidance you can provide.

R/,

megan

From: Williams, Megan
Sent: Tuesday, September 06, 2011 10:53 AM
To: Wang, Weijun; Wilson, George
Cc: Manoly, Kamal; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Thank you, Weijun. This is most helpful.

We are seeing many cracks in concrete walls in the turbine building (below grade), which have been leaking water since the flood started. I am trying to find out the Structures Monitoring Program owner, to see if they had a baseline inspection documentation (they should have completed for license renewal ~2004) indicating what cracks were evident before the flood, and their size, etc.

Do we have any kind of regulatory authority to get them to monitor these cracks more quantifiably (i.e. crack monitors, etc.)? Should we be assessing the soil type and saturation now as well as susceptibility from its original state? Would that information be helpful?

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Megan

From: Wang, Weijun
Sent: Tuesday, September 06, 2011 9:56 AM
To: Williams, Megan; Wilson, George
Cc: Manoly, Kamal; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Megan:

Without knowing much details, I'd like to suggest the follows:

1. Get documents to see how the hydraulic loading was considered during the original structure and foundation design. If the actual flood level is higher than the original design, then the additional lateral pressure and uplift force may cause some damage to the foundation walls and foundation floor concrete slabs, and also may have negative impact on the stability of foundation soils.

2. Regardless the flood levels considered in the original design, you may still need to inspect the structure and foundation to see if there is any damages caused by flooding, such as cracks and settlement. The GPR is a good method to detect voids and the licensee should also perform additional NDT testing to inspect the integrity of the foundation walls and floor concrete slabs or mats, should cracks be discovered.

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The above just for your reference. Please let me know if you have questions.

Thanks.

Weijun
(301)415-1175

From: Williams, Megan
Sent: Tuesday, September 06, 2011 9:44 AM
To: Wilson, George
Cc: Manoly, Kamal; Wang, Weijun; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Good morning,

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I am trying to get information on the GPR they are using to look for voids. Are there any other specific questions or things you all can think of that I should look at while on site?

Thank you,

Megan Williams
RIV

From: Wilson, George
Sent: Tuesday, August 23, 2011 5:53 AM
To: Williams, Megan
Cc: Manoly, Kamal; Wang, Weijun; Cook, Christopher
Subject: RE: Fort Calhoun Flooding

Megan use Kamal Manoly as your reference person he will get assistance from Weijun Wang in NRO

From: Williams, Megan
Sent: Friday, August 19, 2011 12:28 PM
To: Wilson, George
Subject: Fort Calhoun Flooding

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I left you a voicemail, and understand you are out of the office until next week, but I thought I would also send you an email, since I will be out of the office next week on inspection.

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I appreciate any guidance you have in this area.

r/,

Megan Williams

Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

LIC-11-0132
December 17, 2011

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

References:
1. Docket No. 50-285
2. Letter from OPPD (J. A. Reinhart) to NRC (Document Control Desk), dated April 4, 2011 (LIC-11-0023)
3. Letter from OPPD (T. R. Nellenbach) to NRC (Document Control Desk) dated May 13, 2011 (LIC-11-0051)
4. Letter from OPPD (T. R. Nellenbach) to NRC (Document Control Desk) dated May 16, 2011 (LIC-11-0039)
5. Letter from OPPD (T. R. Nellenbach) to NRC (Document Control Desk) dated August 12, 2011 (LIC-11-0088)

Subject: Licensee Event Report 2011-003, Revision 3, for the Fort Calhoun Station

Please find attached Licensee Event Report 2011-003, Revision 3, dated, December 17, 2011. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(v)(B) and (D). If you should have any questions, please contact me.

Sincerely,

D. J. Bannister
Vice President and Chief Nuclear Officer
Fort Calhoun Station

DJB /epm

Attachment
c: E. E. Collins, Jr., NRC Regional Administrator, Region IV
L. E. Wilkins, NRC Project Manager
J. C. Kirkland, NRC Senior Resident Inspector
INPO Records Center

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Fort Calhoun Station	05000285	1 OF 5

4. TITLE

Inadequate Flooding Protection Due To Ineffective Oversight

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
2	3	2011	2011	- 003 -	3	12	16	2011	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☒ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☒ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Erick Matzke	402-533-6855

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

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

During identification and evaluation of flood barriers, unsealed through wall penetrations in the outside wall of the intake, auxiliary and chemistry and radiation protection buildings were identified that are below the licensing basis flood elevation. Additionally a potential flooding issue was identified on the inside of the Intake Structure. Holes were noted in the floor at the 1007'6" level, which is the ceiling of the Raw Water Vault.

A summary of the root causes included: a weak procedure revision process; insufficient oversight of work activities associated with external flood matters; ineffective identification, evaluation and resolution of performance deficiencies related to external flooding; and "safe as is" mindsets relative to external flooding events.

The penetrations were temporarily sealed and a configuration change was developed and implemented whereby permanent seals were installed. A one foot sandbag berm was placed around the holes. Comprehensive corrective actions to address the root and contributing causes are being addressed through the corrective action program.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Fort Calhoun Station	05000285	2011	- 003 -	3	2 OF 5

NARRATIVE

BACKGROUND

As a result of a Nuclear Regulatory Commission (NRC) inspection conducted from January 1 to June 21, 2010, the NRC determined that Fort Calhoun Station (FCS) did not have adequate procedures to protect the intake structure and auxiliary building against external flooding events. Specifically, contrary to Technical Specification 5.8.1.a, the station failed to maintain procedures for combating a significant flood as recommended by Regulatory Guide 1.33, Appendix A, section 6.w, "Acts of Nature." The NRC identified the following violation of NRC requirements associated with a yellow significance determination process finding in the mitigating systems cornerstone in inspection report 05000285/2010008 dated October 6, 2010:

> Technical Specification 5.8.1.a, "Procedures," states, "Written procedures and administrative policies shall be established, implemented, and maintained covering the following activities: (a) The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978." NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Appendix A, "Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors," Section 6, recommends procedures for combating emergencies and other significant events. Section 6.w, "Acts of Nature," includes, in part, procedures for combating floods.
>
> Contrary to Technical Specification 5.8.1.a, since 1978, written procedures and administrative policies were not maintained covering the applicable procedures recommended by NRC Regulatory Guide 1.33, Revision 2, Appendix A. Specifically, the licensee failed to maintain written procedures for combating a significant external flood as recommended by NRC Regulatory Guide, Appendix A, Section 6.w, "Acts of Nature." The licensee's written procedures did not adequately prescribe steps to mitigate external flood conditions in the Auxiliary Building and Intake Structure up to 1014 feet mean sea level, as documented in the Updated Final Safety Analysis Report [USAR].

The NRC reported that the station's flood protection strategy was not fully effective during worst-case Missouri River flooding scenarios. The strategy required workers to install floodgates in front of the doors to the plant's auxiliary building and intake structure, and then stack and drape sandbags over the top of the floodgates up to a height of five feet. The procedural guidance was inadequate because the cross-section on top of the floodgates would not support a stacked sandbag configuration that would retain five feet of moving water.

NARRATIVE

EVENT DESCRIPTION

During identification and evaluation of flood barriers (condition report (CR) 2010-2387), in response to NRC findings previously noted, unsealed through wall penetrations in the intake structure were identified that are below the licensing basis flood elevation. These penetrations were installed during the installation of upgrades to the plant fire protection system. As a result of the penetrations not being sealed, the intake structure was vulnerable to water inflow during an extreme flooding event. This inflow had the potential to affect the operability of both trains of safety related raw water pumps (ultimate heat sink).

On February 4, 2011, an eight (8) hour report was made under 10 CFR 50.72 (b)(3)(v)(D) to the NRC Headquarters Operation Office (HOO) at 1717 CST (Event Number (EN) 46594). The report should have been made on September 9, 2009. (Documented in Condition Report (CR) 2011-0801)

During the extent of cause analysis for this issue the following penetrations were identified as not having been reported.

- Unsealed penetrations were created during the installation of the original plant security system and were abandoned when the security system was replaced (approximately 1985). The penetrations for the "new" security system were appropriately sealed. The old penetrations were abandoned and not sealed. As a result of the penetrations not being sealed, the intake structure was vulnerable to water inflow during an extreme flooding event. This inflow had the potential to affect the operability of both trains of safety related raw water pumps (ultimate heat sink). On February 27, 2011, an eight (8) hour report was made under 10 CFR 50.72 (b)(3)(v)(D) to the NRC Headquarters Operation Office (HOO) at 1640 CST (EN 46590). The report should have been made on January 27, 2011. (Documented in CR 2011-0609)

- Two additional conduits have been identified that are not sealed. These conduits penetrate the south wall of the auxiliary building near the transformers into Room 19 and were created as part of a station modification that was in progress at the time. Flooding through the penetrations could have impacted the ability of the station's auxiliary feedwater (AFW) pumps to perform their design accident mitigation functions. On March 31, 2011, an eight (8) hour report was made under 10 CFR 50.72 (b)(3)(v)(D) to the NRC Headquarters Operation Office (HOO) at 2232 CDT (EN 46716) (Documented in CR 2011-2470)

- A weakness in the flood protection strategy was discovered that would prevent protection of the raw water pumps for floods above 1007 feet-6 inches mean sea level (MSL). The flood protection strategy includes control of the intake cell level by throttling the sluice gates and running raw water pumps to maintain cell level. During the preparation of a calculation to demonstrate the validity of this method it was determined that the grid backwash pipe for each grid and the surface sluice penetrate the east wall of the intake structure through an unsealed penetration (7 penetrations). The grid backwash line is an 18 inch pipe passing through a 24 inch sleeve. Cell in-leakage through these penetrations would be beyond the capacity of the raw water pumps resulting in flooding of the intake structure and loss of the raw water pump function. On March 22, 2011, an eight (8) hour report was made under 10 CFR 50.72 (b)(3)(v)(D) to the NRC Headquarters Operation Office (HOO) at 1611 CDT (EN 46690). (Documented in CR 2011-2161)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
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NARRATIVE

- An unsealed conduit (location 56E-S-43 in pull box 127) was identified that penetrates the auxiliary building into Room 56. Room 56 contains the safety related electrical switchgear. In addition, a drain flow path from the Chemistry and Radiation Protection (CARP) building into Room 23 which can drain into the rooms holding the Emergency Core Cooling System (ECCS) Pumps was identified. On April 8, 2011, an eight (8) hour report was made under 10 CFR 50.72 (b)(3)(v)(D) to the NRC Headquarters Operation Office (HOO) at 1240 CDT (EN 46741). (Documented in CRs 2011-2072 and 2011-2448)

- A potential flooding issue in the Intake Structure, 1007' 6" level was identified. The areas of concern are the holes in the floor at the 1007' 6" level, where the screen wash header penetrates the ceiling of the Raw Water (RW) vault. There are five penetrations of concern. Flooding through the penetrations could impact the ability of the station's RW pumps to perform their design accident mitigation functions. On May 26, 2011, an eight (8) hour report was made under 10 CFR 50.72 (b)(3)(v)(D) to the NRC Headquarters Operation Office (HOO) at 0508 CDT (EN 46893). (Documented in CR 2011-5012)

- Three fittings into room 19 (auxiliary feedwater and plant air compressors) and fittings into room 56E (electrical switchgear) were found to contain no filling material. One additional fitting into room 56E that was thought to be capped was found to be open with a sheet metal box covering the inside access thereby obscuring inspections. The stations auxiliary feedwater and safety related electrical switch gear could be affected. On October 20, 2011, an eight (8) hour report was made under 10 CFR 50.72 (b)(3)(v)(B) and (D) to the NRC Headquarters Operation Office (HOO) at 1246 CDT (EN 47359). (Documented in CR 2011-8547)

Since each of these incidents are events that are related (i.e., they have the same cause and consequences) and they were discovered during a single activity (i.e., investigation to correct the initial problem) then per NUREG 1022, Revision 2, these are being reported in one LER. This report is being made per 10 CFR 50.73(a)(2)(v)(B) and (D).

CONCLUSION

A root cause determination was prepared in connection with CR 2010-2387 which documents the causes of the problem.

The following four (4) root causes explain why written procedures were inadequate to mitigate the external flood conditions prescribed by the Updated Safety Analysis Report (USAR). These root causes address the NRC issued yellow finding as well as the specific penetrations being addressed in this LER.

- Historically, when procedures for flooding protection were restructured or substantially augmented, a weak procedure revision process did not assure FCS met its USAR requirements.
- Supervisory and management oversight of work activities associated with external flood matters was not sufficient to prevent this issue from occurring.
- The FCS organization has not been effective in ensuring that performance deficiencies related to external flooding are adequately identified, evaluated, and resolved.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Fort Calhoun Station	05000285	2011	- 003 -	3	5 OF 5

NARRATIVE

- Mindsets existed that FCS was "safe as is" relative to external flooding events. These mindsets collectively led to the incorrect conclusion that regulatory requirements were being met.

CORRECTIVE ACTIONS

Configuration changes were developed as needed and the penetrations sealed. Comprehensive corrective actions to address the root and contributing causes have been developed and will be addressed through the corrective action program.

SAFETY SIGNIFICANCE

The Fort Calhoun Station is required to be protected from flooding within the station's licensing basis. The safety related equipment required to mitigate the consequences of an accident were affected by these findings. The openings could have jeopardized the ability of the safety related equipment to perform their design basis function during an accident. Other methods of removing decay heat and mitigating the consequences of a flooding event were available. Therefore, this external flooding concern has substantial importance to safety as indicated by the Yellow Finding issued for this event.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event does result in a safety system functional failure in accordance with NEI-99-02.

PREVIOUS EVENTS

No licensee event report prior to this has identified the flooding issue. LER 2011-001 which had been written to report a flooding penetration has been cancelled as indicated in Reference 3 to the cover letter to this LER. The penetration is discussed in this LER.

Uribe, Juan

From: Karas, Rebecca
Sent: Friday, December 23, 2011 1:21 PM
To: Williams, Megan; Wang, Weijun; Xi, Zuhan; Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Cook, Christopher; Murphy, Martin
Subject: RE: Geotechnical question

Zuhan, Weijun and I are all in next week, so that would be fine.

Rebecca Karas, Chief
Geosciences and Geotechnical Engineering Branch 1 Division of Site and Environmental Analysis
Office of New Reactors U.S. Nuclear Regulatory Commission
Phone: 301-415-7533
Fax: 301-415-5397

-----Original Message-----
From: Williams, Megan
Sent: Friday, December 23, 2011 12:16 PM
To: Wang, Weijun; Karas, Rebecca; Xi, Zuhan; Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Cook, Christopher; Murphy, Martin
Subject: RE: Geotechnical question

I am not sure I can get a bridge going today - since our office move, we have had trouble getting them set up. I have my cell phone, and we can try this afternoon, if you all are in one place, or if one of you can set up a phone bridge. Otherwise, it may be next week.

Thank you,

Megan

From: Wang, Weijun
Sent: Thursday, December 22, 2011 4:22 PM
To: Karas, Rebecca; Xi, Zuhan; Williams, Megan; Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Cook, Christopher; Murphy, Martin
~~Subject: RE: Geotechnical question~~

Yes, I will be in tomorrow.

Weijun

From: Karas, Rebecca
Sent: Thursday, December 22, 2011 4:21 PM
To: Xi, Zuhan; Williams, Megan; Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Wang, Weijun; Cook, Christopher; Murphy, Martin
Subject: RE: Geotechnical question

Maybe it would be a good idea to have a phone call so we can understand a little better the information you have and if there are other aspects we can help with? Zuhan, Weijun and I are all in the office tomorrow.

Rebecca Karas, Chief
Geosciences and Geotechnical Engineering Branch 1 Division of Site and Environmental Analysis
Office of New Reactors U.S. Nuclear Regulatory Commission

Phone: 301-415-7533
Fax: 301-415-5397

From: Xi, Zuhan
Sent: Thursday, December 22, 2011 11:35 AM
To: Williams, Megan; Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Wang, Weijun; Karas, Rebecca; Cook, Christopher; Murphy, Martin
Subject: RE: Geotechnical question

Megan,

Personally I haven't looked at any data. It was just a general discussion with the NRR staff. If the post-flooding evaluation indicated that the site currently is not under saturated conditions, the results of Vp from refraction method is more meaningful. However, since Vs is derived from Vp and depends on the value of Poisson's ratio as Weijun pointed out, the impact to Vs on selecting Poisson's ratio is need to be considered (It can be done by some sensitive calc.)

I assume all the tests performed are for the soil surround NI, and am not sure any tests underneath the NI. From a geotechnical point of view, the soil property under the NI is more important. Some tests such as cross-hole for Vs, inclined boring for sampling may provide some info on the soil property under the NI.

Regards,
Zuhan

From: Williams, Megan
Sent: Thursday, December 22, 2011 10:54 AM
To: Xi, Zuhan; Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Wang, Weijun; Karas, Rebecca; Cook, Christopher; Murphy, Martin
Subject: RE: Geotechnical question

Zuhan,

I understand the difficulties you describe below in having confidence in readings with high saturation rates. Have you looked at the soil boring logs for the Fort Calhoun site completed post-flooding to evaluate whether the conditions are such that we would doubt the conclusions reached by the consultant and licensee that soil characteristics have not significantly changed from original design data?

R/,

Megan

From: Xi, Zuhan
Sent: Friday, December 16, 2011 8:37 AM
To: Williams, Megan; Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Wang, Weijun; Karas, Rebecca; Cook, Christopher; Murphy, Martin
Subject: RE: Geotechnical question

Megan,

It is my pleasure to have the opportunity to provide my comment. As Luissette pointed out that for saturated soil, the compression wave velocity by refraction method will reflect mostly velocities in the water. In contrast to shear waves, compressional waves propagate in saturated soils with a velocity that is strongly affected by the water filling the interstices of soil grains. If the site is under mainly saturated conditions after flooding, compression wave velocity by refraction method is not indicative of soil properties, therefore, the shear wave velocity derived from compression wave velocity may also not reflect the true material properties, no matter what correlations used between Vs and Vp.

Thanks,
Zuhan

From: Williams, Megan
Sent: Thursday, December 15, 2011 4:05 PM
To: Uribe, Juan
Cc: Candelario, Luissette; Vega, Frankie; Xi, Zuhan; Wang, Weijun; Karas, Rebecca; Cook, Christopher; Murphy, Martin
Subject: RE: Geotechnical question

Juan,

Thank you very much for getting me a response to my questions. And thank you, Luissette, Weijun, and Zuhan, for the additional information. I am most grateful for your time and expertise.

R/,

Megan Williams

From: Uribe, Juan
Sent: Thursday, December 15, 2011 2:58 PM
To: Williams, Megan
Cc: Candelario, Luissette; Vega, Frankie; Xi, Zuhan; Wang, Weijun; Karas, Rebecca; Cook, Christopher; Murphy, Martin
Subject: FW: Geotechnical question

Megan

After some discussion w/ NRO staff, please read the following email. It will provide additional insights that hopefully can help in your review. They are very knowledgeable and have the most expertise in this field.
Thanks

From: Xi, Zuhan
Sent: Thursday, December 15, 2011 2:59 PM
To: Wang, Weijun; Candelario, Luissette; Uribe, Juan; Vega, Frankie
Cc: Murphy, Martin
Subject: RE: Geotechnical question

I agree with Weijun and Luissette.

Even use a better relationship between Vs and Vp Weijun suggest. Poisson's ratio needs to assumed, which will probably lead over- or under-estimate the Vs. The best way I still think is still cross-hole. And some basic soil lab tests are highly recommended.

Zuhan

From: Wang, Weijun
Sent: Thursday, December 15, 2011 2:49 PM
To: Candelario, Luissette; Uribe, Juan; Xi, Zuhan; Vega, Frankie
Cc: Murphy, Martin
Subject: RE: Geotechnical question

I agree with Luissette.

In addition, the correlation between Vp and Vs given in the email below can only be used to estimate the distance between the recording seismographs and the epicenter. The Vp and Vs presented in the formula are AGERAGE velocities for the media between the recording seismograph and the epicenter. Therefore this relationship cannot be used to develop soil Vs profile, nor to determine layered soil properties.

A better relationship between Vs and Vp at the same location within a soil layer is:

Vp/Vs = SQRT[(2-2v)/1-2v)]   where v is Poisson's ratio (note that error will increase dramatically when v approach to 0.5)

Weijun

From: Candelario, Luissette
Sent: Thursday, December 15, 2011 2:21 PM
To: Uribe, Juan; Wang, Weijun; Xi, Zuhan; Vega, Frankie
Cc: Murphy, Martin
Subject: RE: Geotechnical question

Juan

I do not have much experience with the refraction testing. The applications I have dealt with finds the shear wave velocities (Vs) mainly with downhole testing, PS-suspension logging or cross holes techniques. What I can said is that this site appears to have mainly saturated conditions and for the refraction method the compression wave velocity will reflect mostly velocities in the water, instead of giving accurate data for the soil.

Please refer to RG 1.132 and NUREG/CR-5738 for more details on this method and to evaluate the applicability of it on this site conditions.

Thanks
Luissette

From: Uribe, Juan
Sent: Thursday, December 15, 2011 10:56 AM
To: Wang, Weijun; Xi, Zuhan; Candelario, Luissette; Vega, Frankie
Cc: Murphy, Martin
Subject: FW: Geotechnical question

Guys

I drafted the email below in order to inform Megan (FCS pm in RIV) about Weijun's insights . After some reading and in talking with Luissette, I also added the Vs relation formula with Vp that would tie with Weijun's insight in bullet #3.

Just wanted to make sure you guys were ok with what's written and make sure this relationship between Vp and Vs wasn't used anymore, or was inaccurate.

Megan

FYI
Further discussion with NRO has yielded the following insights. Good insights to keep in mind when conducting your review.
I know for fact that they have determined the compression wave, or p-wave velocities (Vp) by refraction testing, as stated in the report. In following the bullets suggested below by Weijun Wang, I believe that the shear wave velocities can be approximated by using p-wave (compression waves) data with a simple relationship between variables. This is:

D= delta t/ (1/Vs - 1/Vp)

Where delta t is the time difference between the two waves and D denotes the distance between the seismograph and the epicenter.

This can also yield useful data in measuring the change in soil properties. Keep in mind that as an approximation, shear waves account for 26%, compression waves 7% and Rayleigh waves for 67% of the total energy in wave propagatin from an earthquake (from Woods, 1968)

From: Wang, Weijun
Sent: Monday, December 12, 2011 9:25 AM
To: Uribe, Juan
Cc: Xi, Zuhan; Vega, Frankie; Candelario, Luissette
Subject: RE: Geotechnical question

Juan:

1. Any SPT blow counts need to be normalized before the results can be properly used. The correction factors will mainly consider the effects of energy efficiency of the equipment, but other factors, such as the OCR should also be considered when determining soil properties. It doesn't matter when the SPT tests were conducted, the principle has not changed.

2. The SPT test results can be used to estimate some properties of soil, such as the strength of the soil (there are established correlations between SPT blow count and soil strength parameters) and liquefaction potential. As to evaluate liquefaction potential, other methods should be considered as well to increase the confidence level.

3. Please keep in mind that only conducting SPT around the foundation is not sufficient to estimate the soil condition underneath the foundation. If they could drill holes on the bottom of the foundation floor would be idea, but at least they should measure Vs under the corners of the foundation.

Please let me know if you have any questions.

Weijun
(301)415-1175

From: Uribe, Juan
Sent: Friday, December 09, 2011 1:09 PM
To: Xi, Zuhan; Wang, Weijun
Subject: Geotechnical question

Gentlemen

Thanks for the help and insights the other day. Going back to the topic of SPT blowcounts and the ASTM standard, are you aware if ASTM D 6066 allows for correction factors to be applied to blowcounts for the purpose of comparing results done in the same area, ~45 years later, to account for differences in equipment and technology? Or is this practice simply limited to assessing liquefaction potential of soils?

Thanks

JUan f. uribe Civil engineer (EIT)
nrr/DE/EMCB| 301-415-3809 | O9F10 | Juan.Uribe@nrc.gov<mailto:pia@contoso.com>

U.S. Nuclear Regulatory Commission

P Please consider the environment before printing this e-mail

Robles, Jesse

From: King, Mark
Sent: Friday, January 20, 2012 10:16 AM
To: Robles, Jesse
Subject: FW: Ft Calhoun and a need for an OpE COMM soon ... here's a status item update draft it up and let me know when you want the assignment

Jesse,
The below info is being used to develop our AO Report to Congress on Ft. Calhoun as an "Other Event of Interest" item. But may give you a good start on an OpE COMM. In particular see the 0350 letter: hyperlinked item →(**ML113470721**).
Mark

From: King, Mark
Sent: Wednesday, January 18, 2012 11:32 AM
To: Tomon, John
Cc: Cartwright, William; Chernoff, Harold
Subject: Ft Calhoun Status... and my suggestions for covering the on-going issues there

John,
I tend to agree with the OGC rep...we should try to be completely "open" with what's going on at Ft. Calhoun... even though a significant portion of this has occurred after the end of the Fiscal Year 2011.

For your awareness related to Ft. Calhoun ---The site has now been placed under **IMC 0350** oversight...

...*For Fort Calhoun effective December 13, 2011* the NRC made a change to the units regulatory oversight by transitioning it from MC 0305 "**Operating Reactor Assessment Program**" to oversight under inspection manual chapter 0350, "**Oversight of Reactor Facilities in a Shutdown Condition due to Significant Performance and/or Operational Concerns.**" For more information please see the December 13, 2011 letter available in ADAMS at: hyperlinked item →(**ML113470721**).

Therefore I suggest we revise/expand the Ft. Calhoun title and write-up section to cover these broader topics that may result in either an AO item (due to the breaker fire item, which occurred in June 2011) going RED, or perhaps the ASP review driving Ft. Calhoun issues to be in the AO report section next year for FY 2012.

John - Please feel free use /adapt these input suggestions into whatever Region-IV provides you, as needed/ you see fit. ***We (NRR) will review the final write-up for concurrence.***

Here are my suggested changes/ additions in yellow below.....

EOI-02 Fort Calhoun Station, Unit 1, Nuclear Power Plant: Unusual Event Due to High River Level and other performance issues leading to a reactor facility in an extended shutdown condition

This event is included in this report because it received significant media attention, and the public, as well as the local and national media, perceived it to be of high health and safety significance. However, as described below, the 2011 flooding of the Missouri River and the subsequent high water levels surrounding Unit 1 of the Fort Calhoun Station (FCS) were actually of low safety significance Additionally, the Omaha Public Power District (OPPD) (the licensee) always maintained plant safety, and the NRC maintained oversight.

The FCS, located approximately 19 miles north of Omaha, NE, on the Missouri River, consists of a single pressurized-water reactor (PWR) designed by Combustion Engineering. On June 6, 2011, FCS declared a

A2a

Notification of Unusual Event (NOUE) in anticipation that the Missouri River level at the plant would reach 1,004 feet mean sea level (MSL). By design, the plant is protected to a river level elevation of 1,014 feet MSL. Record snowfall totals during the winter, followed by a rapid snowpack melt and significant rainfall during the spring and summer, caused this rise in the Missouri River. FCS had been shut down on April 9, 2011, for a planned refueling outage and remained shut down during the entire period of flooding. FCS emergency preparedness and response capabilities were maintained throughout the flooding event, and the physical security of the plant was maintained in accordance with the security plan.

On June 26, 2011, OPPD reported the failure of the water berm, which was installed as a supplemental floodwater protection measure. The failure of the water berm allowed floodwaters to surround the main electrical transformers, which prompted OPPD to take the precautionary measure of transferring from offsite power to onsite emergency diesel generators (EDGs). Reactor shutdown cooling and spent fuel pool cooling were unaffected during the transfer power to the onsite EDGs, and the water berm failure had no impact on safety-related equipment. The NRC entered monitoring mode, with the Region IV Incident Response Center having the response lead and 24-hour site coverage. On August 29, 2011, the licensee terminated the NOUE for flooding when the Missouri River level receded to less than NOUE entry criteria.

On August 10, 2011, OPPD provided the NRC with a Post-Flooding Recovery Action Plan, which called for extensive reviews of plant systems, structures, and components to assess the impact of the floodwaters (available at Agencywide Documents Access and Management System (ADAMS) Accession No. ML112231755). The NRC issued a confirmatory action letter (CAL 4-11-003) on September 2, 2011 (available at ADAMS Accession No. ML112490164), which described various focus areas for site restoration, plant systems and equipment status, equipment reliability, design and licensing basis, emergency planning and security impacts, and the recovery actions that would occur before the unit proceeds to startup. The focus areas are divided into action plans and associated specific action items, and as part of this commitment, OPPD committed to meet with the NRC to ensure that there was agreement that the facility was ready for restart. The NRC expectation for this meeting is that OPPD will discuss the results of the assessments called for in the plan, actions taken to address any problems identified during the assessments, and overall assessment of the readiness to return the plant to full-power operation. NRC review and approval in accordance with CAL 4-11-003 are required before startup of the FCS reactor.

In addition, some of the media attention related to this FCS flooding issue was attributed to an earlier NRC inspection finding issued to FCS that involved flooding issues. On October 6, 2010, the NRC had issued a final finding of substantial safety significance (a yellow finding), which was identified during a 2009 NRC component design-basis Inspection (available at ADAMS Accession No. ML102800342). The NRC team identified deficiencies in the licensee's coping strategies for protecting vital areas between 1,009.5 and 1,014 feet MSL. By identifying and having the licensee address this issue earlier and before the flooding began on June 26, 2011, the NRC enhanced the safety of the site. At no time was the actual health and safety of the public compromised by either the yellow finding or the actual flooding on June 26, 2011.

NOTE: Other performance issues subsequently identified were also under evaluation and not completed until after the end of the Fiscal Year. These issues and their review lead to the extended plant shutdown continuing after the flooding condition no longer existed and a decision by the NRC to transition the unit to oversight under inspection manual chapter 0350, "**Oversight of Reactor Facilities in a Shutdown Condition due to Significant Performance and/or Operational Concerns.**" For more information please see the NRC letter dated December 13, 2011 available in ADAMS at: **ML113470721**

This event did not meet the AO reporting criteria for FY 2011; however, information identified during the OPPD assessment of flooding impacts could potentially cause the NRC to determine that this event is of high safety significance (i.e, an AO report item per Criterion II.C in Appendix A to this report). The NRC staff continues to evaluate the event/ issues under both the NRC's Accident Sequence Precursor (ASP) Program and the significance determination process (SDP).

The ASP Program provides an integrated risk analysis of all deficiencies, equipment failures, and degraded conditions that were observed during the event. The inspection program separately assesses the risk associated with each performance deficiency. Therefore, for events involving multiple licensee performance deficiencies and equipment failures, as in the FCS event, it is not unexpected that the ASP and inspection programs would assign different risk significance levels. As such, the integrated approach used by the ASP Program complements the inspection program.

If NRC evaluation for the issues at Ft. Calhoun results in a final red finding determination or if the final ASP analysis of these events results in its identification as a significant precursor, the NRC will report this event in Section II, "Commercial Nuclear Power Plant Licensees," of the next fiscal year's AO report and in the FY 2012 "Performance and Accountability Report to Congress."

FYI, for your consideration.
Mark

Mark King
Senior Reactor Systems Engineer
NRR/ADRO/DIRS/IOEB
Operating Experience Branch
301-415-1150
Mark.King@nrc.gov

NRC – One Mission – One Team

March XX, 2012

MEMORANDUM TO: Harold Chernoff, Chief
Operating Experience Branch
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

FROM: Jesse Robles, Reactor Systems Engineer
Operating Experience Branch
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

SUBJECT: CLOSURE MEMORANDUM: ISSUE FOR RESOLUTION 2011-01, "FORT CALHOUN – FAILURE TO MAINTAIN EXTERNAL FLOODING PROCEDURES"

The Nuclear Regulatory Commission (NRC) staff has completed their review of Issue for Resolution (IFR) 2011-01: Fort Calhoun – Failure to Maintain External Flooding Procedures.

On September 17, 2009, during a Component Design Basis Inspection (CDBI) at Fort Calhoun, the inspection team identified that the licensee failed to maintain adequate procedures that protect the intake structure and auxiliary building during external flooding events. This procedure consisted of stacking and draping sandbags on top of installed floodgates to protect the plant up to a flood elevation of 1,014 feet msl~~msl~~. When inspectors asked plant staff to demonstrate this procedure, they were unable to complete the procedure because the cross section on the top of the floodgates was too small to accommodate the sandbags so that it would retain a four foot static head of water. This resulted in a Yellow Finding (substantial safety significance) being issued to Fort Calhoun Station.

The subject event was screened into the IFR program by the Clearinghouse on October 14, 2010, based on LIC-401 criteria: 1.B, "Reactor Oversight Process Significance Determination Process finding of white or higher" (i.e., yellow), and 2.J, "potential new or novel failure mode, system interaction, material condition or degradation, or other phenomena that may have instructive value for the industry or the NRC." The NRC staff issued an internal communication pertaining to the inadequate flood procedure issue in the form of a Reactor Operating Experience Community Forum posting on October 19, 2010. Additionally, this event has been captured in the Reactor Operating Experience database for future tracking and trending purposes (reference ROE record 7432).

CONTACT: Jesse Robles, NRR/DIRS
301-415-2940

In addition to this IFR, the Office of Research ha~~i~~s reviewed~~ing~~ Proposed Generic Issue (GI) PRE-GI-0009, "Flooding of Nuclear Power Plant Sites Following Upstream Dam Failures,"(See GI Submission Document, ML101900305, ~~and the~~ and associated documents, such as the Acceptance Review Document, ML102210339), Screening Analysis Report (MLXXXX) and the Approval of the Establishment of GI-204 (MLXXXXX). Currently, the GI-204 issue has been transferred for Regulatory Office Implementation (MLXXXXX). This GI was proposed to address the uncertainty in the risk associated with external flooding caused by dam failures upstream of Nuclear Power Plants (NPP), and that more recent flooding studies may also indicate an overall change in risk not previously considered in other original studies.

Fort Calhoun experienced prolonged flooding conditions during the summer of 2011 that prompted them to declare a Notification of Unusual Event. A Reactor Operating Experience Community Forum posting was posted for this event as well. The NRC issued a Confirmatory Action Letter (CAL) (ML112490164) on September 2, 2011, to confirm the actions FCS plans to take in its submitted Post-Flooding Recovery Action Plan (ML112430102). Additionally, Fort Calhoun was placed in IMC 0350, "Oversight of Reactor Facilities in a Shutdown Condition Due to Significant Performance and/or Operational Concerns" (see Notification of Change to Regulatory Oversight of Fort Calhoun Station (ML113470721) and Fort Calhoun IMC 0350 Charter (ML120120661)). Additional inspections related to the flood recovery will be performed as part of this oversight.

The Mechanical and Civil Engineering Branch (EMCB) provided input to the evaluation of this Issue for Resolution. The staff's conclusions and recommendations are presented in the enclosed staff evaluation (ML111680450). Based on the nature of the issue, EMCB recommends the NRC develop a process for periodic interactions~~interacting~~ with relevant federal agencies, such as USACE and the Federal Emergency Management Agency (FEMA) in order to update licensees on flood studies. Some~~This~~ procedures already exist~~s~~, like ~~through~~ the Memorandum of Understanding (MOU) with USACE (ML062920211) and the MOU with FEMA (ML051680117). This IFR Closure Memorandum and enclosure will be sent to the points-of-contact of~~for~~ these MOUs for their awareness. EMCB also recommends the issuance of an Information Notice (IN) in order to address the fact that the availability of third-party information, such as studies, technical papers, assessments, etc., has the potential to affect the design basis of a licensed facility and therefore should be evaluated. There is no explicit requirement that requires licensees to determine if third-party information represents a safety concern to the site. TAC ME6847 was opened on August 12, 2011 for this IN entitled "Third-Party Information That Can Potentially Affect the Design Basis of a Licensed Facility" (ML101400109). This information originated from Region IV staff, but will be issued by EMCB. Additionally, the staff recommends that Inspection Procedure (IP) 71111.01 "Adverse Weather Protection" be revised to add a caveat to perform the inspection if any substantial change to the design basis flood levels and/or operating procedures has occurred~~reflect the findings of their evaluation~~ (see enclosure). A feedback form will be submitted to the appropriate Reactor Inspection Branch (IRIB) staff member so that the enclosed recommendations and the IN mentioned above are incorporated in the next revision to the IP.

The Operating Experience Branch requests your approval to close IFR 2011-01. You may approve by signing below.

I approve the staff's recommendation to close IFR 2011-001 and associated TAC ME4866 in accordance with NRR Office Instruction LIC-401, "NRR Reactor Operating Experience Program."

Enclosure:
As stated

APPROVAL: ____________________________ Date: ____________________
Harold Chernoff, Chief
Operating Experience Branch
Office of Nuclear Reactor Regulation

I approve the staff's recommendation to close IFR 2011-001 and associated TAC ME4866 in accordance with NRR Office Instruction LIC-401, "NRR Reactor Operating Experience Program."

Enclosure:
As stated

APPROVAL: ______________________________ Date: ____________________
Harold Chernoff, Chief
Operating Experience Branch
Office of Nuclear Reactor Regulation

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ADAMS Accession Number: ML12066A030

OFFICE	NRR/DIRS/IOEB	NRR/DE/EMCB	NRR/DIRS/IOEB	NRR/DIRS/IOEB
NAME	JRobles	JUribe	MKing	HChernoff
DATE	03/06/12	03/ /12	03/07/12	03/ /12

OFFICIAL RECORD COPY

Brock, Kathryn

From: Virgilio, Martin
Sent: Friday, March 09, 2012 5:14 PM
To: Chang, Lydia
Cc: Brock, Kathryn; Mamish, Nader
Subject: REPLY: Ft Calhoun Preliminary Red Finding

Thanks, Lydia

Please ensure this gets incorporated into the Monday morning note to the Chairman. I do not know who has the duty next week.

Marty

From: Chang, Lydia
Sent: Friday, March 09, 2012 3:37 PM
To: Virgilio, Martin
Subject: Ft Calhoun Preliminary Red Finding

Hi Marty:

Just to let you know the Region IV informed us that Ft. Calhoun inspection report will be released on Monday with Preliminary Red Finding associated with the breaker fire from last June. Three violations are cited: (1) design modification control procedure, (2) maintenance practice procedure, and (3) train separation. The findings are determined to have high safety significance. High media interest is expected and Region IV is currently finalizing the Communication Plan. Thanks...

Lydia C.

March 29, 2012

MEMORANDUM TO: Harold Chernoff, Chief
Operating Experience Branch
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

FROM: Jesse Robles, Reactor Systems Engineer ***/RA/***
Operating Experience Branch
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

SUBJECT: CLOSURE MEMORANDUM: ISSUE FOR RESOLUTION 2011-01, "FORT CALHOUN – FAILURE TO MAINTAIN EXTERNAL FLOODING PROCEDURES"

The Nuclear Regulatory Commission (NRC) staff has completed their review of Issue for Resolution (IFR) 2011-01: Fort Calhoun – Failure to Maintain External Flooding Procedures.

On September 17, 2009, during a Component Design Basis Inspection (CDBI) at Fort Calhoun Station (FCS), the inspection team identified that the licensee failed to maintain adequate procedures that protect the intake structure and auxiliary building during external flooding events. This procedure consisted of stacking and draping sandbags on top of installed floodgates to protect the plant up to a flood elevation of 1,014 feet mean sea level (msl). When inspectors asked plant staff to demonstrate this procedure, they were unable to complete the procedure because the cross section on the top of the floodgates was too small to accommodate enough sandbags to retain a four foot static head of water. This resulted in a Yellow Finding (substantial safety significance) being issued to Fort Calhoun Station.

The subject event was screened into the IFR program by the Operating Experience (OpE) Clearinghouse on October 14, 2010, based on LIC-401 criteria: 1.B, "Reactor Oversight Process Significance Determination Process finding of white or higher" (i.e., yellow), and 2.J, "potential new or novel failure mode, system interaction, material condition or degradation, or other phenomena that may have instructive value for the industry or the NRC." The NRC staff issued an internal communication pertaining to the inadequate flood procedure issue in the form of a Reactor Operating Experience Community Forum posting on October 19, 2010. Additionally, this event has been captured in the Reactor Operating Experience database for future tracking and trending purposes (reference ROE record 7432).

CONTACT: Jesse Robles, NRR/DIRS
301-415-2940

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In addition to this IFR, the Office of Research has opened GI-204 (See GI Submission Document (ML101900305), Acceptance Review Document (ML102210339), Screening Analysis Report (ML112430114), and the Approval of the Establishment of GI-204 (ML12062A199)). Currently, GI-204 has been transferred for Regulatory Office Implementation (ML120261155). This GI was proposed to address the uncertainty in the risk associated with external flooding caused by dam failures upstream of Nuclear Power Plants (NPP). The GI stipulates that more recent flooding studies may also indicate an overall change in risk not previously considered in other original studies.

FCS experienced prolonged flooding conditions during the summer of 2011 that prompted them to declare a Notification of Unusual Event. A Reactor Operating Experience Community Forum posting was posted for this event as well. The NRC issued a Confirmatory Action Letter (CAL) (ML112490164) on September 2, 2011, to confirm the actions FCS plans to take in its submitted Post-Flooding Recovery Action Plan (ML112430102). Additionally, FCS was placed in the Inspection Manual Chapter (IMC) 0350, "Oversight of Reactor Facilities in a Shutdown Condition Due to Significant Performance and/or Operational Concerns," process on December 13, 2011 (see Notification of Change to Regulatory Oversight of Fort Calhoun Station (ML113470721) and Fort Calhoun IMC 0350 Charter (ML120120661)). This IMC establishes criteria for the oversight of reactors that are shutdown due to significant performance or operational problems. IMC 0350 establish criteria for the oversight of licensee performance for licensees that are in a shutdown condition as a result of significant performance problems or operational events. It establishes a record of the major regulatory and licensee actions taken and technical issues resolved leading to approval for restart and to the eventual return of the plant to the routine Reactor Oversight Process (ROP). It does this by establishing a review panel that verifies that licensee corrective actions are sufficient prior to restart and provides assurance that following restart the plant will be operated in a manner that provides adequate protection of public health and safety. Additional inspections related to the flood recovery will be performed as part of this oversight.

The Mechanical and Civil Engineering Branch (EMCB) provided input to the evaluation of this IFR. The staff's conclusions and recommendations are presented in the enclosed staff evaluation (ML111680450). Based on the nature of the issue, EMCB recommends the NRC develop a process for periodic interactions with relevant federal agencies such as the U.S. Army Corps of Engineers (USACE) and the Federal Emergency Management Agency (FEMA) in order to update licensees on flood studies. Some procedures for this already exist, including the Memorandum of Understanding (MOU) with USACE (ML062920211) and the MOU with FEMA (ML051680117). This IFR Closure Memorandum and enclosure will be sent to the points-of-contact of these MOUs for their awareness. EMCB also recommends the issuance of an Information Notice (IN) in order to address the fact that the availability of third-party information such as studies, technical papers, and assessments has the potential to affect the design basis of a licensed facility, and therefore should be evaluated. There is no explicit requirement that requires licensees to determine if third-party information represents a safety concern to the site. TAC ME6847 was opened on August 12, 2011 for this IN entitled "Third-Party Information That Can Potentially Affect the Design Basis of a Licensed Facility" (ML101400109). This information originated from Region IV staff, but the IN will be issued by EMCB. Additionally, the staff recommends that Inspection Procedure (IP) 71111.01, "Adverse

Weather Protection," be revised to add a caveat to perform the inspection if any substantial change to the design basis flood levels and/or operating procedures has occurred (see enclosure). A feedback form will be submitted to the appropriate Reactor Inspection Branch (IRIB) staff member so that the enclosed recommendations and the IN mentioned above are incorporated in the next revision to the IP.

IOEB staff recommends closure of IFR 2011-01.

Enclosure:
As stated

APPROVAL: ____/RA/__(Eric Thomas for)_ Date: __March 29, 2012_
Harold Chernoff, Chief
Operating Experience Branch
Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation

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NAME	JRobles	JUribe	MKing	HChernoff (EThomas for)
DATE	03/06/12	03/28/12	03/07/12	03/29/12

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Brock, Kathryn

From: Temp, Edo
Sent: Monday, April 09, 2012 7:51 AM
To: Bates, Andrew; Hart, Ken; Sola, Clara; Vietti-Cook, Annette; Wright, Darlene
Cc: Ash, Darren; Borchardt, Bill; Brock, Kathryn; EDO_ETAs; Ellmers, Glenn; Landau, Mindy; Mamish, Nader; Rakovan, Lance; Rihm, Roger; Virgilio, Martin; Weber, Michael; Temp, Edo
Subject: April 9, 2012: Items of Interest to the Chairman

Good morning. Here are the items of interest for the Chairman:

- The staff plans to issue its Red finding to Ft. Calhoun today or tomorrow.

Thanks,

Breeda

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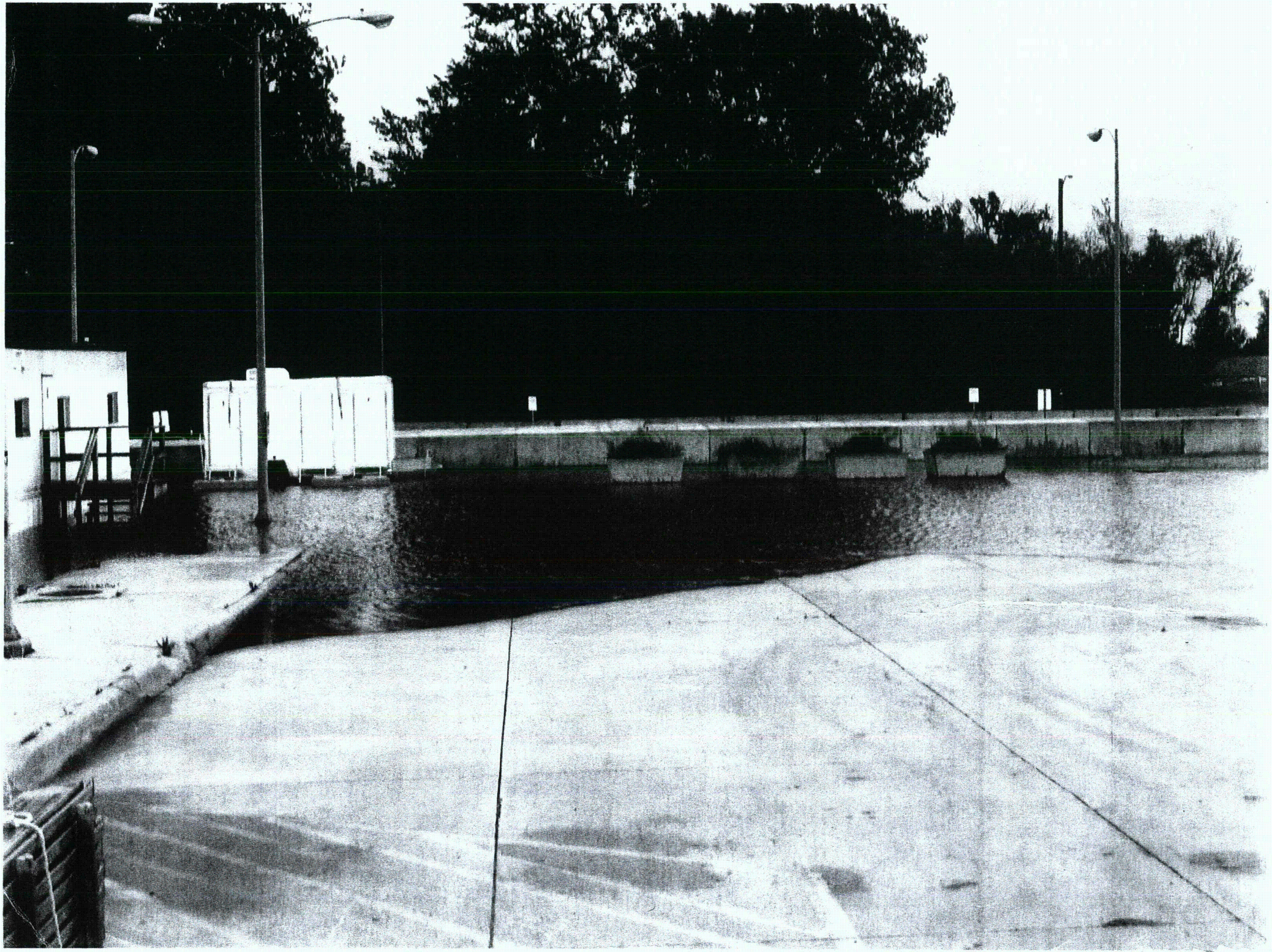

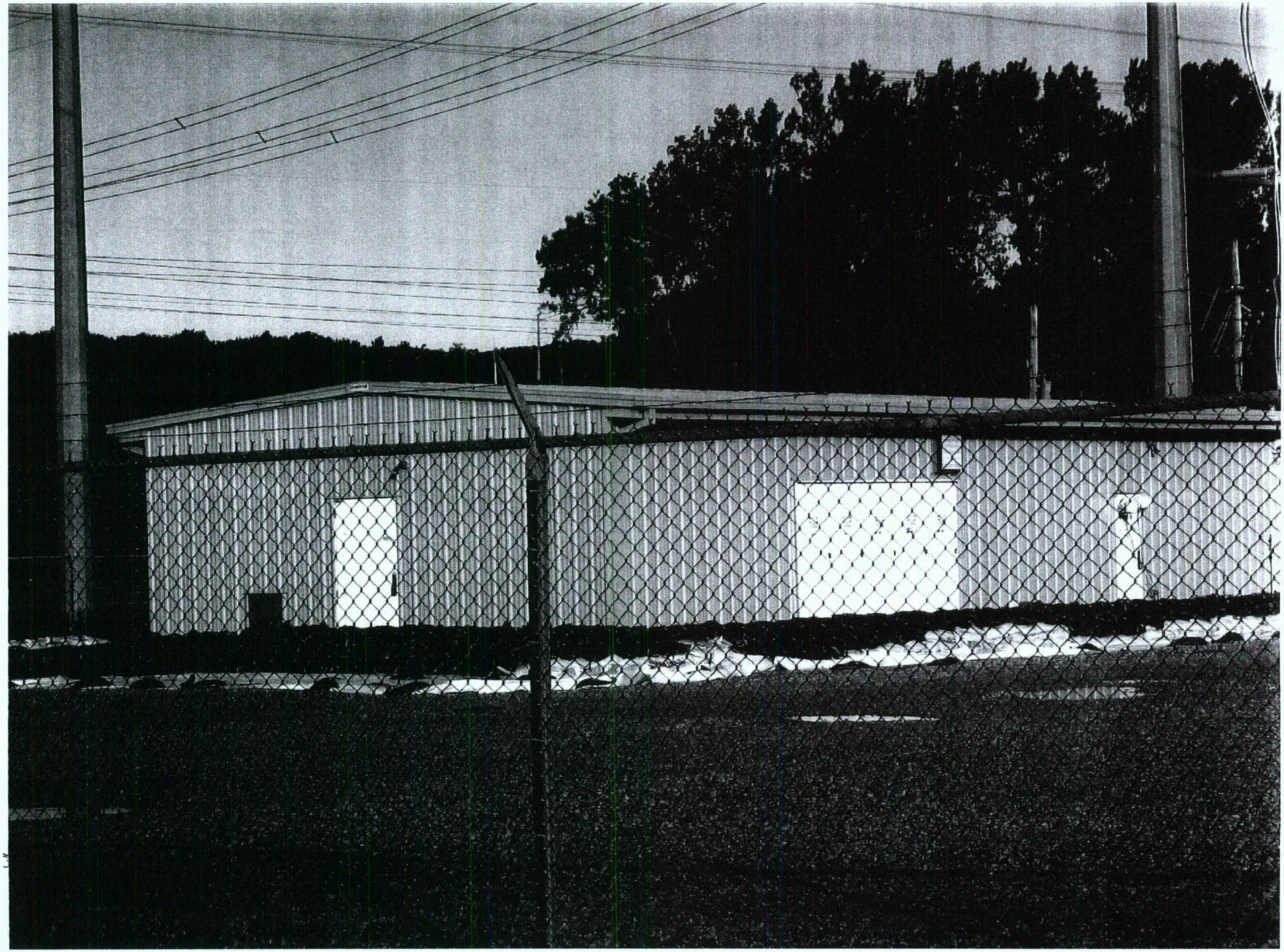

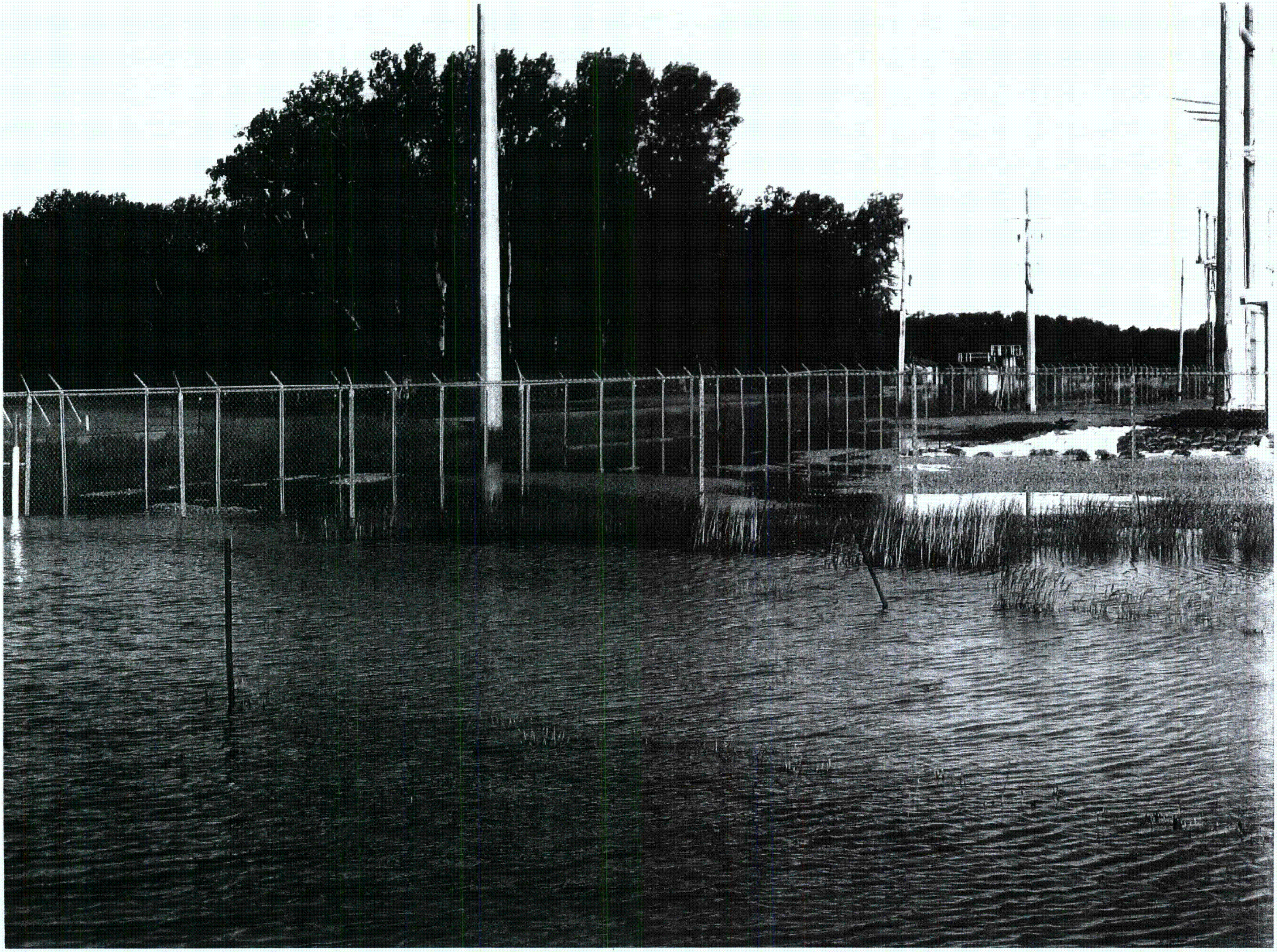

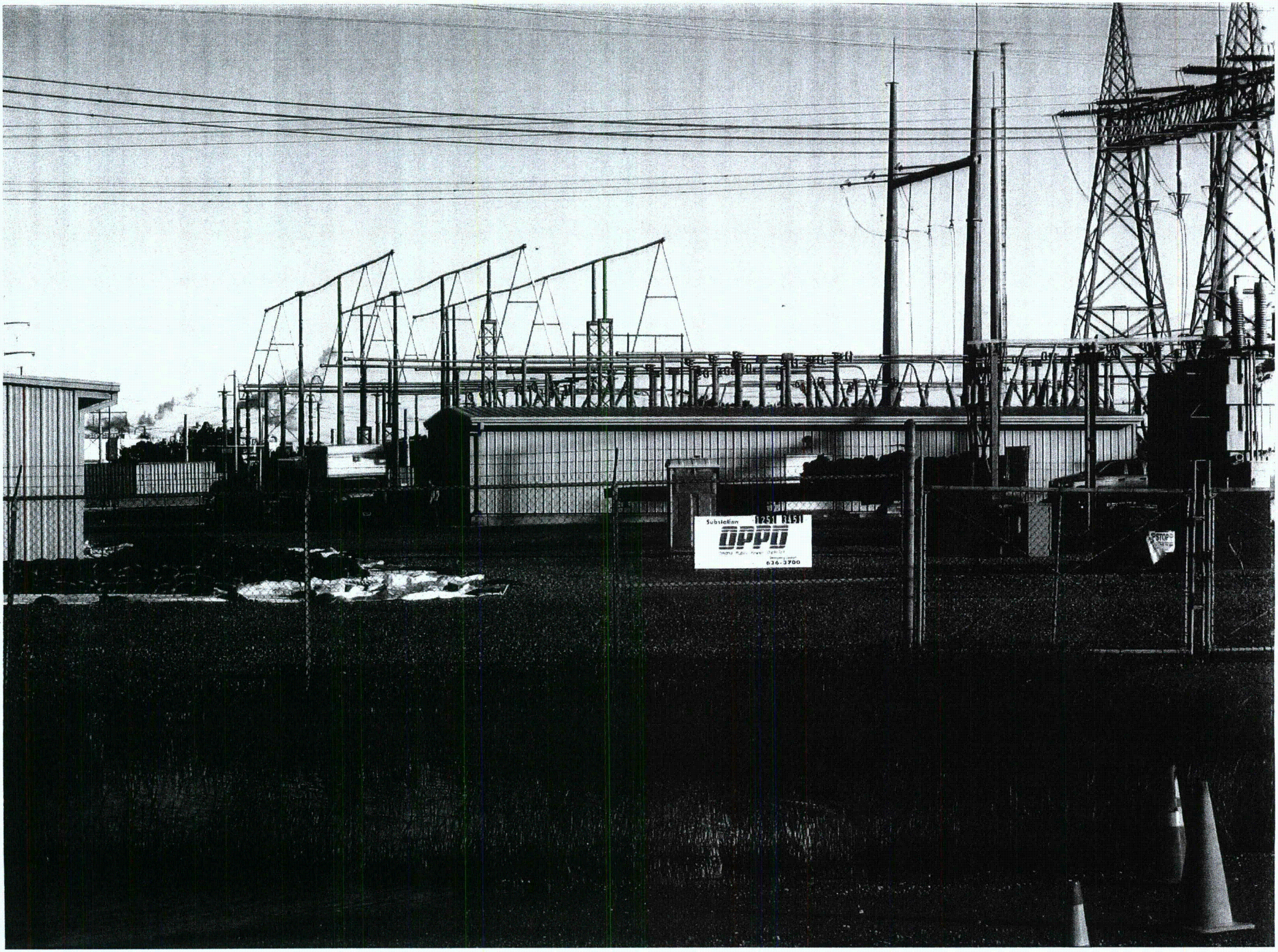
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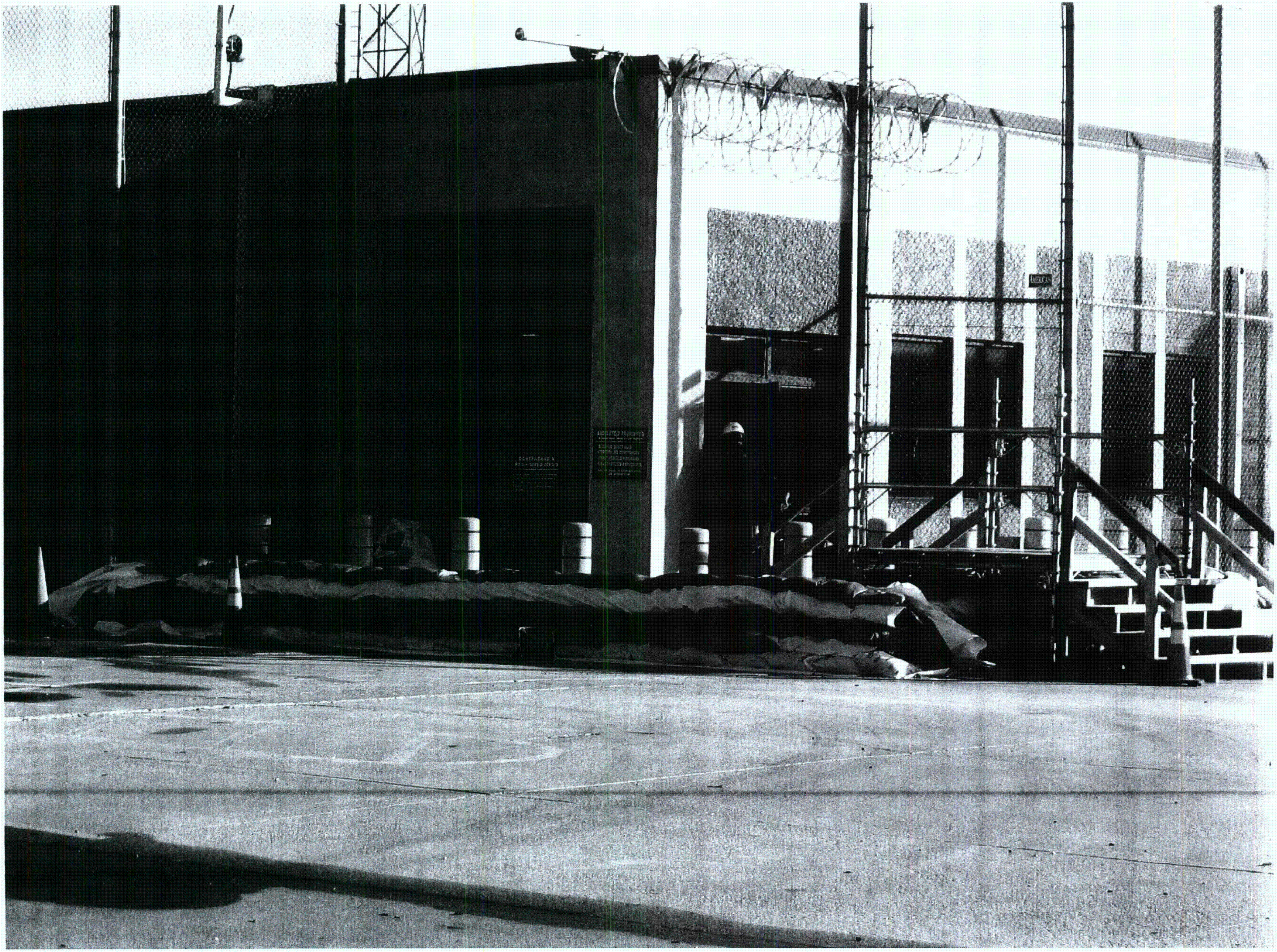

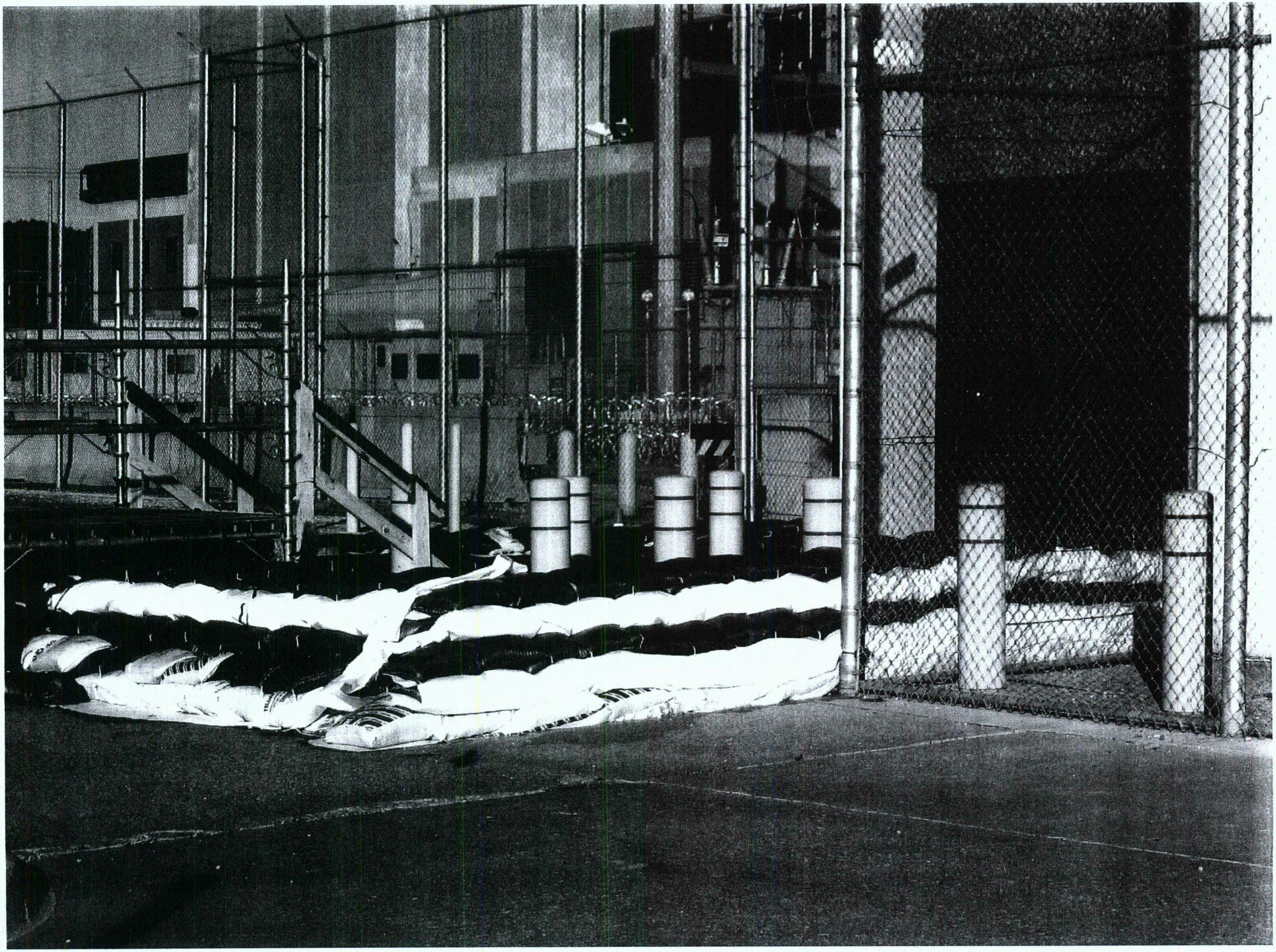


CAUTION
Tripping Hazard
KEEP EYES ON PATH