UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

February 7, 2003

Paul D. Hinnenkamp, Vice President - Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: NRC SPECIAL INSPECTION REPORT 50-458/02-07

Dear Mr. Hinnenkamp:

On November 14, 2002, the NRC completed a Special Inspection at the River Bend Station. The enclosed report documents the inspection findings which were discussed with you and members of your staff on November 14, 2002.

The inspectors examined activities associated with a reactor scram and subsequent system interactions that occurred on September 18, 2002. The inspection was conducted in accordance with Inspection Procedure 93812, “Special Inspection,” and the inspection team charter. The inspectors reviewed selected procedures, records, and evaluation activities. The inspectors also interviewed plant personnel. As a result of this inspection, the NRC developed a sequence of events, initiated a risk significance determination of the event, and assessed your response to and evaluation of the event.

Based on the results of this inspection, the NRC has identified an issue that requires further analysis to determine the significance. This issue was an apparent violation of Technical Specification 5.4.1.a, which requires the licensee to establish and implement procedures to operate the condensate system. Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 was not locked open as required and constitutes an apparent violation of this Technical Specification.

The final significance of this apparent violation is to be determined. The issue was initially characterized as having a safety significance of more than minor. Further determination of significance will be made through a Phase 3 significance determination analysis by the NRC.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC’s document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

David N. Graves, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-458
License: NPF-47

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NRC Inspection Report
50-458/02-07

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-458
License:	NPF-47
Report:	50-458/02-07
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	September 24 through November 14, 2002
Inspectors:	M. O. Miller, Resident Inspector River Bend Station, Team Leader J. S. Dodson, Project Engineer J. F. Drake, Operations Engineer
Approved By:	D. N. Graves, Chief, Project Branch B

ATTACHMENT: Special Inspection Charter

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SUMMARY OF FINDINGS

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NRC Inspection Report 50-458/02-07

IR 50-458/02-07; Entergy Operations, Inc.; on 09/24/2002-11/14/2002; River Bend Station. Special Inspection Report. Event Followup.

The report covers a special inspection conducted by Region IV inspectors concerning a reactor scram with a loss of the condensate and feedwater systems and the condensate storage tank discharge of steam. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, “Significance Determination Process.” Findings for which the Significance Determination Process does not apply are indicated by the severity level of the applicable violation. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- (to be determined) The inspectors identified an apparent violation of Technical Specification 5.4.1.a, which requires that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Item 4.n requires instructions for operation of the condensate system.

  System Operating Procedure SOP-0007, “Condensate System,” Revision 21, required Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 to be locked open. On September 18, 2002, Valve CNM-FCV200 was found to be not properly locked open. The failure to properly lock Valve CNM-FCV200 in the open position resulted in unexpected closure of the valve and a loss of feedwater flow to the reactor vessel following a reactor scram.

  The final significance of this issue will be determined using the Significance Determination Process (Section 3.5).

REPORT DETAILS

1 Special Inspection Activities

The NRC conducted this special inspection to better understand the circumstances surrounding a malfunction in the turbine control system, the subsequent reactor scram, loss of feedwater, and a discharge of steam from the Condensate Storage Tank (CST) that occurred on September 18, 2002. The decision to conduct a special inspection was based on these collective factors:

- The licensee's risk evaluation documented a Conditional Core Damage Probability for the scram as 9.3E-7. This was statistically equivalent to the threshold (1E-6) established in Manual Chapter 8.3 for conducting special inspections.
- The event included an unexpected loss of feedwater.
- Steam was unexpectedly released from the CST.

The inspectors used Inspection Procedure 93812, "Special Inspection," to conduct the inspection. The team reviewed procedures, logs, instrumentation printouts, corrective action documents, and design and maintenance records for the equipment of concern. The team interviewed key station personnel regarding the event and subsequent posttrip review investigation.

2 Event Description

2.1 Summary

On September 18, 2002, at 8:24 p.m. with the plant at full power, a turbine control system malfunction caused the turbine control and intercept valves to slowly close. The turbine bypass valves (BPVs) opened in response to the increase in steam line pressure. An upscale neutron flux automatic reactor scram occurred 3.5 seconds after the onset of the transient. When reactor water level began to decrease following the scram, feedwater flow increased. This caused Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 to close unexpectedly, isolating the condensate system from the feedwater system. The reactor feedwater pumps (RFP) tripped on low suction pressure shortly after Valve CNM-FCV200 closed. A reactor low water level alarm (Level 3) was received as reactor water level continued to decrease.

The operators manually started reactor core isolation cooling (RCIC), stabilized the plant in Mode 3, Hot Shutdown, controlled reactor water level between +10 and +51 inches on narrow range (NR) reactor water level instruments, and maintained reactor pressure between 500 and 1090 psig using the turbine BPVs.

Operators in the turbine building reported water coming from under the doors to each Steam Jet Air Ejector (SJAE) room. The reactor operators shut down the condensate pumps and nuclear equipment operators (NEO) isolated the SJAE intercondensers to terminate the water coming from the SJAE rooms.

At approximately 10 p.m., an inspector arriving onsite observed a plume of steam coming from the CST vent. In addition, loud banging noises were noted coming from the CST. The inspector notified control room personnel of these observations. It was later determined that water was flowing from the reactor water cleanup system (RWCU) filter demineralizers to the CST via the feedwater and condensate system.

2.2 Details

At 8:24 p.m., a transient in the +22 volt supply to the turbine control system caused the turbine control valves and intercept valves to slowly close. The BPVs opened automatically to control reactor pressure as designed. When the BPVs reached their full open position and the turbine control valves continued to close, reactor pressure increased to 1087.6 psig. This pressure increase resulted in a reactor power increase and reactor water level decrease.

Reactor Power Response

Approximately 3.5 seconds into the transient, the average power range monitor (APRM) system generated a scram signal on upscale neutron flux from five of the eight APRMs. The licensee analysis showed that a relatively slow transient of this type could be expected to trip sufficient APRM channels to terminate the transient without causing all channels to trip. The highest rated thermal power value recorded was 121.99 percent rated thermal power on APRM E. Licensee analysis of the core thermal performance showed that no core thermal limit was exceeded.

Reactor Pressure Response

The maximum reactor pressure of 1087.6 psig was reached approximately 4.4 seconds into the transient. This was less than one second after the scram was initiated. The scram terminated the pressure transient. There were no reactor safety relief valve actuations because the maximum pressure reached was below the minimum lift setpoint of 1133 psig.

Reactor Water Level Response

Prior to the event, reactor water level was in the normal range at 36 inches as indicated by the NR level indicators. At the same time, the average reactor water level was 21.8 inches as indicated on the wide range (WR) reactor water level instruments. The WR level instruments normally indicate lower than the NR instruments due to the recirculation pump suction effect on the pressure at the WR level instrument variable leg sensing taps. The higher the power level (therefore the higher the reactor recirculation system flow) the lower the indicated reactor water level on the WR level instruments.

At 3.4 seconds from the start of the turbine control system malfunction (just prior to the scram), the average WR level instruments indicated 17.73 inches. When the scram was initiated, the reduction in reactor power caused a further reduction in reactor water level as the steam voids in the core collapsed. At 9.42 seconds from the start of the turbine control system malfunction, reactor water level stopped decreasing at -24 inches, as indicated on the WR level instruments.

As a result of the decrease in reactor water level, feedwater flow increased and reactor water level reached about +20 inches when the feedwater flow was lost (at approximately 35 seconds into the event). This caused reactor water level to drop again quickly to about zero inches. Over the next 9 minutes, reactor water level continued to decrease because the turbine control valves were not yet completely shut. At 8:33 p.m., the turbine control valves closed when the turbine tripped. Reactor water level reached -21 inches WR before the operators manually started RCIC and restored reactor water level.

Feedwater and Condensate Systems Response

Following the scram, the feedwater flow control valves opened in response to lowering reactor water level. This increased feedwater flow caused Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 to close unexpectedly (Figure 1).

The closure of Valve CNM-FCV200 isolated flow to the RFPs and they tripped on low suction pressure. When Valve CNM-FCV200 closed, condensate system pressure increased to greater than 800 psig for a short duration (less than one second). The gaskets on the SJAE intercondensers failed as a result of the pressure surge. The failed gaskets allowed the condensate pumps to transfer water from the hotwell to the turbine building 95 foot elevation. At 8:33 p.m., the operators shut down the condensate pumps to stop this transfer of water.

Flow in the feedwater system and much of the condensate system stopped when Valve CNM-FCV200 closed, which, in turn, caused the condensate system and RFP minimum flow valves to open. This created a flowpath to vacuum-drag water from the feedwater and condensate systems to the hotwell, which was still under condenser vacuum. Interlocks closed the minimum flow valves when the RFPs tripped and the condensate pumps were shut down.

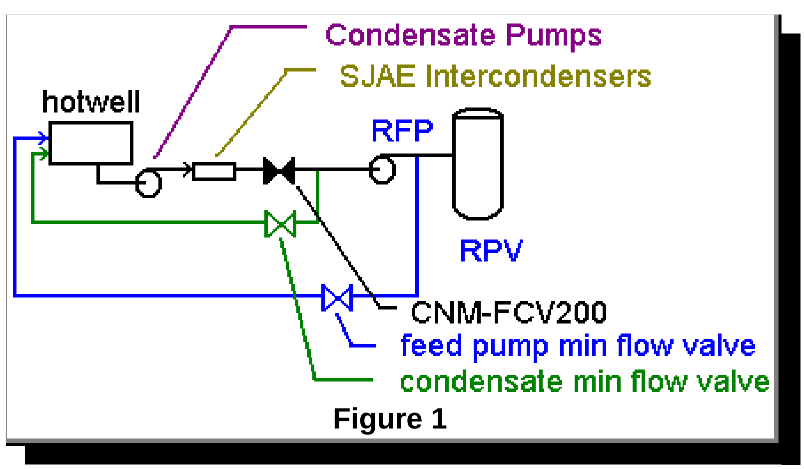


Figure 1

The resulting high hotwell level caused condenser hotwell reject level control Valve CNS-LCV105 to the CST to open. When CNS-LCV105 opened, it created an abnormal and reverse flowpath from the condensate and feedwater systems to the CST (Figure 2).

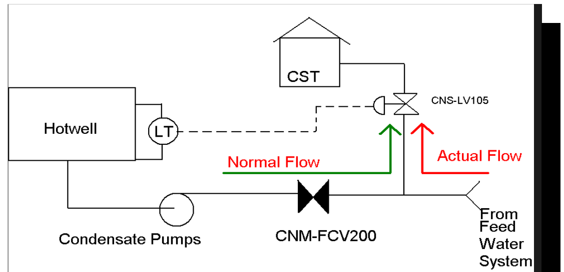


Figure 2

The RWCU system, which takes suction from the reactor recirculation system and discharges to the feedwater lines, was in service at the time of the event. At 8:25 p.m., immediately after the RFPs tripped, RWCU flow increased rapidly. As the feedwater system depressurized (due to RFP and condensate minimum flow valves being open to the main condenser) the flow path for RWCU was from the reactor recirculation system, through RWCU filter demineralizers, backwards through feedwater Line B into the feedwater and condensate systems (Figure 3). A check valve on feedwater Line A prevented a similar flowpath through feedwater Line A. The check valve on feedwater Line A had been installed to prevent reverse flow in the feedwater system during RCIC injection.

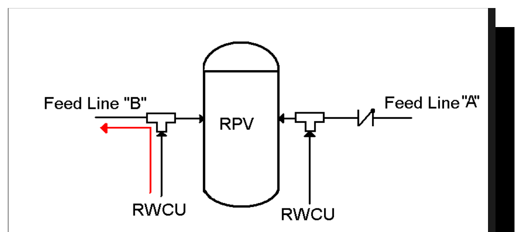


Figure 3

There were two separate flowpaths from the condensate system into the CST discharging 200 psig fluid that was between 356 and 378°F. One flowpath was through Valve CNS-LV105. This line terminates in the CST below the surface of the water. The water in the CST was cooler and quenched the flashing steam, which accounted for the loud noises. The second flow path was through the control rod drive (CRD) pump's minimum flow line and into the space above the water level in the CST. That fluid was flashing into steam and issuing from the vent on the roof of the CST. The operators closed Valve CNS-LCV105 at 11:30 p.m., which

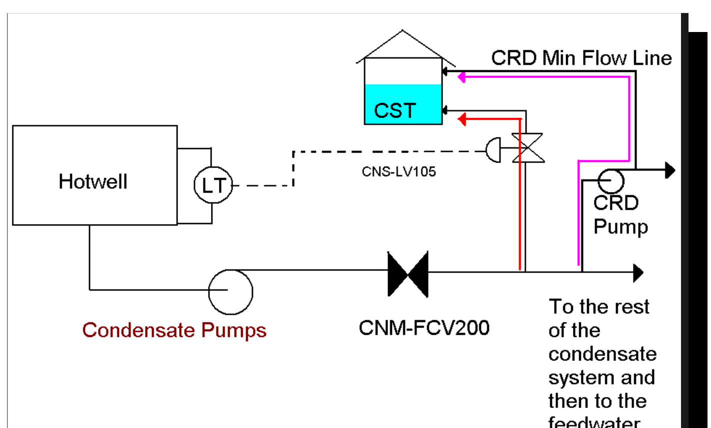


Figure 4

isolated one of the two flowpaths of water from RWCU to the CST. At 11:36 p.m. the operators closed the low pressure feedwater heater inlet valves to start a condensate pump. This isolated the remaining path of water from RWCU to the CRD to the CST, and temperatures in the condensate and feedwater system began to return to their normal values (Figure 4).

<u>Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200</u>

The licensee had previously developed a plant modification to install full flow filtration equipment (prefilters) in the condensate system. The licensee planned to install the modification in two phases. The first phase of the modification was installed in May 2002. The installed portion of the modification included Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 (Figure 5).

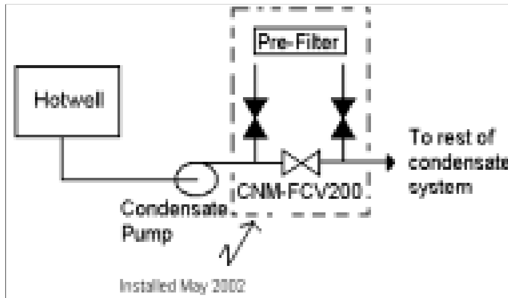
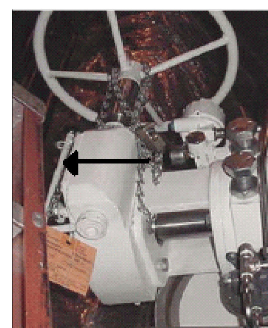


Figure 5

CNM-FCV200 was a triple eccentric metal seated valve, similar to a butterfly valve. The valve actuator was air-operated with a handwheel for manual operation.

The photos on the right were taken with the handwheel engaged and the handwheel disengaged. When the lever was in the engaged position (moved to the left as shown in the top photo), the handwheel was physically connected to the valve through gears. Chaining the handwheel would then keep the valve from moving. When the lever was in the disengaged position (moved to the right as shown in the lower photo), the valve was not connected to the handwheel.



Handwheel engaged

Handwheel Disengaged

Because the handwheel was disengaged, the only thing holding the valve open was friction from the packing and internal components and the weight of the disc. When flow through the valve increased suddenly, as occurred when the feedwater system tried to restore reactor water level following the scram, the force on the disc was enough to cause the valve to go closed.

<u>CRD System Response</u>

<u>Design</u>

The CRD system contains two pumps, with one pump normally running. The pump takes a suction from the condensate system and supplies high pressure water to the control rod hydraulic control units for normal control rod movement and control rod scram functions. A portion of the CRD pump discharge flow is diverted through the minimum flow bypass line to the CST. This flow is controlled by an orifice and is sufficient to prevent pump damage if the pump discharge was inadvertently closed.

Condensate water to the CRD system is processed by two sets of filters. The CRD pump suction filters are disposable element type with a 25-micron absolute rating. A 250-micron strainer in the CRD suction filter bypass line protects the CRD pumps when the filters are being serviced. Downstream of the CRD pumps are two parallel drive water filters which have a filtration rating ranging from 50-micron to 15-micron absolute. A differential pressure indicator and main control room alarm monitor the in-service suction filter element as it collects foreign materials. A similar arrangement is provided for the drive water filters.

<u>CRD Pump Suction Filters and Drive Water Filter</u>

Normally, the suction supply to the CRD pumps was from the condensate system. When pressure in the condensate system decreased to less than that from the head of water in the CST, the CST became the suction source. When this occurred, the suction filter differential-pressure alarm annunciated. The operators responded to the high differential-pressure annunciator by implementing the alarm response procedure. The operators bypassed the suction filter and placed the standby drive water filter in service. The CRD suction filter was found to be fouled with red iron oxide debris. The licensee stated that this was not an unusual occurrence when the CRD pump suction shifted from the condensate system to the CST.

<u>Temperature Response</u>

The normal temperature in the part of the condensate system upstream of the feedwater heaters was approximately 180°F. During the transient on September 18, 2002, condensate temperature increased to approximately 350°F due to RWCU flow through the system from about 9:40 p.m. until 11:15 p.m. The inservice CRD pump was cavitating during this time as a result of the high temperature in the CRD pump suction. The following was a list of CRD Loads:

- Charging header
- Drive header
- Cooling header
- Reactor recirculation pump seal purge
- Reactor water level reference leg backfill

While the CRD pump was cavitating, there was no flow through the charging or drive headers. Therefore, these headers were not subjected to the higher condensate system temperatures. These headers acted as a thermal barrier between the CRD pumps and control rod hydraulic control units. The scram occurred at 8:25 p.m. with CRD suction temperature at 128°F. At 8:58 p.m., the scram was reset and CRD pump suction temperature had increased to 143°F as a result of residual heat in the low pressure feedwater heaters. The temperature remained about 143°F for approximately 35 minutes following the scram reset. At the time of the scram reset, the CRD flow control valves were closed, diverting flow to the charging header. The scram accumulators were fully charged approximately 10 minutes later, before the temperature of the water supply to the CRD pumps began to increase from RWCU flow. When the scram accumulators were fully charged, the CRD flow control valve began to open, redirecting flow to the cooling water header. Therefore, the scram accumulators were not exposed to the elevated condensate water temperatures.

CRD fluid temperature peaked at approximately 350°F at 10 p.m. As CRD fluid temperature began to increase, flow was supplied to the cooling water header and then to the CRD mechanisms. The CRD mechanisms were normally exposed to high temperatures and can withstand temperatures in excess of 500°F. Therefore, based on the observed temperatures and design capabilities of the mechanisms, no functions associated with the CRD mechanisms were affected.

High temperature water (350°F) from the CRD system was also being injected into the reactor recirculation pump seals at a flow rate of 2 gpm per recirculation pump. This CRD water was cooled by the reactor plant component cooling water system before injection into the reactor recirculation pump seal cavity area. The highest reactor recirculation pump seal temperature was 165°F, which was within the normal range. No seal high temperature alarms were received.

The reactor water level reference leg backfill system was designed to reduce the probability of step changes in indicated reactor water level caused by steam bubble formation in the reference leg. The formation of steam bubbles in the reference leg can force water mass out of the reference leg, which would cause indicated reactor water level to increase. The steam bubbles then either collapse or vent out of the reference leg and the indicated reactor water level returns to normal. There were no signs of step changes in indicated reactor water level seen by the operators or noted during a review of reactor water level graphs. At the nominal settings of 0.004 to 0.012 gpm through long, uninsulated, small bore lines that supply the CRD water to the reference legs, ambient losses to atmosphere would be such that the temperature of the water arriving at the reference legs would be well below the temperature necessary for step changes in indicated reactor water level to occur.

3 Special Inspection Areas

3.1 Sequence of Events

a. Inspection Scope

The inspectors developed a sequence of events related to the September 18, 2002, reactor scram and compared it to the licensee's sequence of events to determine if the event had been accurately reviewed.

b. Background

The inspectors developed a sequence of events related to the identification and timeliness of actions taken in response to the event of September 18, 2002. The time line was generated from the sequence of events printout, annunciator log report, archived operator logs, and interviews with the licensee's staff.

This sequence of events was then compared to the licensee's sequence of events. The only differences identified in the licensee's sequence of events were minor event omissions. These were resolved as being related to the level of detail chosen by the licensee for their sequence of events and had no impact on the licensee's ability to assess the event.

c. Findings

No findings of significance were identified.

3.2 Operator Response

a. Inspection Scope

The inspectors evaluated the adequacy of the operator response to this transient. The sequence of events log, the annunciator report, and the operator logs were reviewed. The inspectors also interviewed 10 control room operators and NEOs that were on duty at the time of the event.

b. Background

The operators implemented the following procedures in response to the reactor scram and loss of the condensate and feedwater systems:

- EOP-001, "RPV Control," Revision 19
- AOP-001, "Reactor Scram," Revision 19
- AOP-002, "Main Turbine and Generator Trips," Revision 15A
- AOP-003, "Automatic Isolations," Revision 17B
- AOP-006, "Condensate/Feedwater Failures," Revision 13

The licensed operators recognized the loss of condensate and feedwater systems and started the RCIC system to maintain reactor water level. The operators placed residual heat removal Loop A in service to remove the heat of RCIC from the suppression pool. The licensee was unaware of the steam issuing from the CST vent and the loud noises in the CST until informed by one of the inspectors. The licensee initially believed that the source of the steam and noise was RCIC recirculation flow diverted back to the CST, although this had not been observed in the past. The RWCU flow into the CST was unintentionally terminated when the low pressure feedwater heater inlet valves were closed as part of the preparations for starting a condensate pump, almost 3 hours after the scram. The source of the vent steam was not determined until after the event was terminated and the plant response was evaluated by the licensee.

Two NEOs were contaminated upon entering the SJAE rooms to investigate the water leaking from under the door. The contaminated personnel were properly decontaminated. No internal exposures occurred.

Fire alarms sounded on the turbine building 67 foot elevation as a result of the paint melting off the condensate system piping. Operators responded appropriately to the fire alarms. No injuries were reported.

c. Findings

No findings of significance were identified.

3.3 Unmonitored Release Evaluation

a. Inspection Scope

The inspectors interviewed cognizant personnel and reviewed the CST venting to atmosphere (unmonitored release) during the event. The following items were reviewed and compared with regulatory requirements:

- Radiation Section Procedure RSP-0008, "Offsite Dose Calculation Manual," Revision 11

- Condition Reports CR-RBS-2002-01371 and -01372 and current offsite dose calculations contained in Condition Report CR-RBS-2002-01384

- "Evaluation of Potential for Unmonitored Release of Radioactive Material from Turbine Building during Post SCRAM Period on September 18, 2002," by Davey Wells, Superintendent Radiation Protection

- "Radiological Evaluation of the CST Release," by Senior Environmental Specialist (1136)

- Technical Requirements Manual Section 3.11, "Radioactive Effluents"

The inspectors performed independent calculations of the potential radiation release using RWCU effluent sample data to determine if the licensee's calculations were accurate and no release in excess of 10 CFR Part 50, Appendix I, requirements occurred.

b. Background

There were two flowpaths of reactor coolant from the reactor to the CST from 9:25 p.m. until 11:30 p.m. The reactor coolant traveled from the reactor to the RWCU pumps through two parallel filter demineralizers and into feedwater Line B. Reactor coolant then flowed backwards through the feedwater system into the condensate system. Once in the condensate system the reactor coolant had two flowpaths to the CST. One flowpath was through Condenser Hotwell Reject Level Control Valve CNS-LCV105.

Valve CNS-LCV105 was opened early in the event due to high water level in the hotwell. At 11:30 p.m., the operators closed Valve CNS-LCV105, isolating this flowpath. The other flowpath was through the CRD pumps. The CRD pumps normally take suction on the condensate system. The CRD pump minimum flow line discharges to the CST. During the event, reactor coolant from the RWCU system, via the condensate system, and CRD pump's minimum flow line, was being discharged to the CST. At 11:36 p.m., the operators closed the low pressure heater string inlet valves to start a condensate pump. This stopped RWCU flow to the CST.

c. Findings

No findings of significance were identified.

3.4 Posttrip Review

a. Inspection Scope

The inspectors evaluated the adequacy of the licensee's posttrip review. Included in this review was the thoroughness of the licensee's assessment of the event, whether potential complications on the plant systems (i.e., extent of condition) were properly considered and whether the immediate corrective actions were comprehensive and appropriate.

b. Background

1. Scram Report and Post-Trip Review Checklist

General Operating Procedure GOP-003, "Scram Recovery," Enclosure 1, "Post Trip Review Checklist," dated September 18, 2002, and Enclosure 2, "Scram Report," dated September 18, 2002, were completed and presented to the Operational Safety Review Committee (OSRC) for review.

The OSRC meeting was held in two parts on September 19, 2002. The first session reviewed the following items related to a plant restart:

- GOP-003, "Scram Recovery," dated September 18, 2002
- Cause of the loss of feedwater and pressure transient
- Cause of the electrohydraulic control (EHC) transient
- Cause of the change in feedwater Line B temperature following the scram
- Cause of the increase in temperature in the CST

The OSRC concluded that no further review of GOP-003 was required and that nuclear plant response was satisfactory. Four issues required further review:

- Provide a troubleshooting plan for EHC and plant conditions necessary for that troubleshooting.

- Revise report of the reactor feedwater Line B temperature rise to reflect OSRC discussions relative to RWCU as the source of the elevated temperatures.

- Correlate CRD flow to the temperature rise in the CST.

- Provide the results of an assessment of components served by CRD, with regard to the effect of the high temperature water being pumped by the CRD pumps for the 3-hour period.

The second OSRC meeting was convened at 12 midnight on September 19, 2002, to address the four questions remaining from the first OSRC meeting.

The OSRC approved plant restart and unrestricted operation under the following conditions:

- The troubleshooting plan as outlined for the EHC system be pursued with appropriate instrumentation installed for monitoring during power operation, and

- A mode restraint condition report be written to address the impact of the higher CRD water temperatures on equipment in the CRD system and answered prior to entering Mode 2.

The licensee made appropriate entries into their corrective action program. These corrective actions were assigned and given a due date. The mode restraint corrective actions were completed prior to entering Mode 2.

2. <u>Licensee Evaluation of Piping and Component Pressure Transient</u>

Operations personnel reported condensate piping coating damage, piping noises during the reactor scram transient, and water flooding on the turbine building 67 foot and 95 foot elevations. Consequently the licensee conducted two separate walkdowns (described below) of the condensate and feedwater system. Other than the SJAE intercondenser gasket failure, no physical damage was observed.

At 1 a.m. on September 19, 2002, the licensee walked down the condensate full flow filtration area, condensate and feedwater piping, both heater bays on the turbine building 67 foot elevation, the SJAE rooms on the turbine building 95 foot elevation, the condenser recycle valve room, and piping/hangers in RFP and condensate pump areas. This walkdown focused primarily on identifying transient-induced physical damage to these systems.

The following items were identified:

- Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 indicated 80 percent closed. This valve handwheel was normally locked open. This was immediately reported to the outage control center and the operations shift manager.

- SJAE gasket failures were confirmed.

- Pipe coupling for condensate system flow Element CNM-FE114 leaking

- Component cooling water to condensate Pump C seal leak

- Low pressure feedwater heater BPV CNM-MOV136 broken tie rod on the actuator cover

- Damaged paint on the low pressure feedwater heater bypass line

- Extensive amounts of unidentified debris were noted around several floor drains.

The licensee entered each of these items in the corrective action program and appropriate corrective actions were taken.

At 8 p.m. on the evening of September 19, 2002, the licensee repeated this walkdown (with the exception of the full flow filtration) after condensate Pump CNM-P1B had been placed in service to pressurize the system. This walkdown focused on leak detection and capturing conditions potentially missed during the first walkdown. Seventeen leaks were identified as well as a loose pipe clamp and a detached leak seal injector nozzle.

An evaluation of the pressure transient in the piping as a result of the closure of Valve CNM-FCV200 was also conducted. The value of the pressure used in the evaluations of piping and components was 925 psig. This value was derived from failure of the gaskets in the end-bells of the SJAE intercondensers, plant process computer recordings of the event, and mechanical engineering stress analyses.

An analytical assessment of the condensate piping indicated that static stresses were allowable up to a maximum pressure of approximately 1850 psig. During a water hammer event, the stress applied was short in duration and thus the pressure the piping could withstand was much higher than the allowable static pressure of 1850 psig. The condensate pressure values were sampled by the plant process computer at one second intervals. The highest recorded value was approximately 800 psig. Any pressure higher than this lasted less than one second.

3. <u>Licensee Evaluation of Piping and Component Temperature Transient</u>

A series of events resulted in an increased fluid temperature in the condensate and other interconnected systems. The temperature transient was an unexpected event and not evaluated in the design basis of the affected systems. Licensee engineers evaluated the impact of the temperature on the affected piping, equipment nozzle, pipe supports, and in-line plant components. It was concluded that the evaluated piping and plant components were not adversely affected by the temperature transient and can be relied upon to perform their design functions.

The following was a summation of evaluations for each component:

- The normal operating temperature of the affected condensate piping ranges from 142°F to 304°F. The condensate piping and feedwater heaters were not safety related. The affected piping was qualified to the requirements of ANSI B31.1. Evaluations concluded that the affected piping, pipe supports, and equipment nozzles did not exceed the design code allowable values. The effects of the thermal transient on the instrumentation tubing were bounded by the existing design.

- The condensate demineralizers were qualified to a maximum temperature of 175°F for 2 hours or less. The maximum temperature transient above 180°F lasted for less than 5 minutes. The maximum temperature recorded was 238°F for approximately one minute. Due to the relatively short duration of the thermal transient and the inherent flexibility of the piping, the effects of the transient on the piping, pipe supports, and equipment nozzles were considered negligible.

- The CST was qualified to a temperature of 200°F, higher than the maximum transient temperature noted during this event (180°F). The line from Condenser Hotwell Reject Level Control Valve CNS-LCV105 is supported by spring hangers. Thus this pipe section was very flexible and capable to accommodate expansion in the riser.

- The suction and discharge piping of the CRD pumps within the fuel building was Class II piping and was qualified to an operating temperature of 120°F in pipe stress calculations. A preliminary review of the calculated maximum stresses shows that there was sufficient margin to accommodate the effects of the temperature transient to 350°F, and no code allowables were exceeded.

- CRD system piping inside the containment and drywell, including recirculating pump seal purge piping, was qualified by a licensee vendor. A sample review of the qualifying calculations showed that there was sufficient margin in the piping thermal stress to accommodate the effects of the temperature increase in the system for a short duration as permitted by ASME Section III, NC-3612.3.

- No evaluation was required for the high pressure feedwater heaters since their design temperatures were higher than the transient maximum temperature of 350°F. The feedwater and condensate heater manufacturer was contacted and concluded that a 350°F temperature in the tube side would not cause adverse effects on the structural integrity of the heaters.

- The licensee determined that no damage to the CRD pumps occurred as a result of the cavitation. The pump was subsequently restarted and proper operation was observed.

c. Findings

No findings of significance were identified.

3.5 Root Cause Evaluation

a. Inspection Scope

The inspectors reviewed the licensee's root cause determinations for completeness and accuracy. Key assumptions and facts were independently verified.

b. Background

The licensee prepared two Root Cause Analyses.

- "Automatic Reactor Scram on High Neutron Flux," dated October 28, 2002
- "Closure of Condensate Valve CNM-FCV200," dated November 21, 2002

Generic implications were considered. Each root cause analysis presented evaluations of previous similar occurrences at other plants and a corrective action plan.

Automatic Reactor Scram on High Neutron Flux

The licensee conducted a Kepner-Trego problem analysis and concluded that the most likely cause for the event was a momentary bus ground or failure of a power supply.

Short-term corrective actions were completed and included the following:

- The permanent magnet generator +22 volt and the house +22 volt power supplies were replaced.

- Field cables, wiring, circuitry, connections, and critical card edge connectors were inspected.

- Tests with the turbine simulated to be running at 1800 rpm to determine system response were conducted.

- A high speed data acquisition system was installed on additional points in the turbine control system cabinet to obtain additional data if the event should recur.

There were 18 long-term corrective actions identified. Several were completed during this inspection. The following were representative of the actions:

- Test removed power supplies, in house and at the vendor's facilities. No malfunctions were identified.

- Evaluate alternate power supply vendors and alternate configurations.

- Develop plans and inspect 22vdc bus work.

- Determine a plan for additional inspections/testing to be performed in the next refueling outage if cause for the trip was not determined first.

- Perform additional evaluations of minimum reactor water level reached after the scram (approximately -24 inches). This was completed satisfactorily.

- Investigate whether any offsite power line transients occurred around the time of the event which may have contributed to the event or the plant response after the event.

<u>Closure of Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200</u>

The licensee employed several methods during this root cause analysis. They included:

- document reviews
- personnel interviews
- field walkdowns
- barrier analysis
- TAP root
- Event and causal factor analysis
- Entergy standard root cause investigation guide

Two root causes were identified for the closure of Valve CNM-FCV200:

- Failure to meet management expectations
- Failure to follow up on identified problems

For the root cause "failure to meet management expectations," seven inappropriate actions were identified by the licensee:

1. The Valve CNM-FCV200 request was issued with only subtle indication that the valve was other than what was normally installed in the plant.

2. The training department failed to identify the need for operator training.

3. A component engineer failed to identify different operational aspects of the valve.

4. After providing assurance to management that the valve would not go closed during manual operation, the responsible engineering analysis was limited to feedback provided by the vendor, rather than a comprehensive analysis.

5. Following identification of a potential nonfamiliarity with the new valve actuator, an operator and an engineer failed to resolve the issue.

6. Following difficulty getting the valve open, the organization failed to flag that personnel did not know how to operate the valve.

7. An operator directed a contractor, who was not a qualified representative, to lock the valve when he opens it. At the time the valve was locked open, it was within a tagging boundary and under control of the contractors. No procedural violation occurred.

For the root cause "failure to follow up on identified problems," there were five inappropriate actions identified:

1. Following identification of a potential nonfamiliarity with the new valve actuator, an operator and an engineer failed to resolve the issue.

2. An operator identified the need to learn valve operation without capturing in the process.

3. A procedure writer identified a need to learn valve operation without capturing it in the process.

4. An untrained operator was assigned to verify valve locked open.

5. A valve lineup change in condensate system operating Procedure SOP-0007 did not include handwheel "engage[d]."

The licensee determined that, during the plant modification to install the condensate full flow filtration system, the construction team installed Valve CNM-FCV200 in the closed position so that it could be inserted between two flanges. After the valve was installed, the construction team engaged the manual handwheel and opened the valve. The intent had been to keep the handwheel engaged with the valve and locked in position. At some point following opening, however, the handwheel was disengaged from the valve. The licensee wrapped the handwheel and engaging lever with a chain and put a padlock on the chain. The licensee believed that the construction workers that opened the valve also disengaged the handwheel to prevent inadvertent manual operation. With the handwheel and manual operator disengaged from the valve, the only thing holding the valve in position was the weight of the valve disc and the friction of the valve packing on the valve stem. When the scram took place on September 18, 2002, feedwater flow increased in response to a reactor water level decrease. This increased feedwater flow, with the handwheel disengaged from the valve, caused Valve CNM-FCV200 to close unexpectedly.

There was only one other valve in the plant from this particular vendor with a manual operator, and the manual operator was significantly different and smaller than the valve used for this application. The Engineering Request for this modification did not call out this difference and, as a result, the operations and training staff did not recognize a difference in the operator/valve interface. As a result, training personnel did not obtain the documents necessary to determine that valve operation would be different. Training personnel took credit for previous training experience and assumed the new valve was similar to other valves already installed in the plant.

The inspectors reviewed the licensee's assessment of the failure to properly lock open Valve CNM-FCV200 and the corrective action taken. The licensee appropriately used their processes to subsequently provide training to all of the operators, and made the necessary revision to Condensate System Operating Procedure SOP-007.

c. Findings

An apparent violation of Technical Specification 5.4.1.a was identified when the licensee failed to lock open Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200, as required by System Operating Procedure SOP-0007, "Condensate System," Revision 21. The risk significance determination was still in progress when this report was completed. The final risk significance determination has yet to be determined.

On September 19, 2002, at 1 a.m. following a reactor scram and loss of feedwater on September 18, 2002, the licensee determined that Valve CNM-FCV200 had not been properly locked open, as required by procedure. This performance deficiency occurred on May 15, 2002, when an individual signed a valve lineup affirming Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 was locked open. It was not properly locked open and, as a result, the feedwater flow transient resulting from a reactor scram on September 18, 2002, caused Valve CNM-FCV200 to close unexpectedly and cause a complete loss of feedwater. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program as Condition Report CR-RBS-2002-1372.

The inspectors conducted a Phase I significance determination of the performance deficiency in accordance with Manual Chapter 0609, which led to a Phase II significance determination. The Phase II analysis indicated that the significance was potentially greater than very low. The Phase II determination was validated by an NRC Senior Reactor Analyst (SRA), and a Phase III analysis was initiated. The final risk determination was in progress at the end of this inspection.

Technical Specification 5.4.1.a requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Item 4.n, requires instructions for operation of the condensate system. System Operating Procedure SOP-0007, "Condensate System," Revision 21, required Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 to be locked open. On September 18, 2002, Valve CNM-FCV200 was not properly locked opened. This is an apparent violation of Technical Specification 5.4.1.a (50-458/2002-07-01). The licensee has revised Procedure SOP-0007 and trained the operators on the proper operation Valve CNM-FCV200.

3.6 Risk Analysis

a. Inspection Scope

The inspectors reviewed the licensee's risk analysis of the event, as documented by Entergy Inter-Office Correspondence SA-02-030, "Risk Impact of Feedwater System Out of Service," dated September 27, 2002. An independent risk analysis being conducted by NRC was in progress at the end of this inspection.

b. Background

The licensee conducted an analysis of the risk impact of the feedwater system being out of service. The licensee calculated the incremental change in core damage frequency at 9.3E-7 and concluded that the calculated incremental risk value with the feedwater out of service for 4 months was nonrisk-significant. Four months was the approximate time that Condensate Prefilter Vessel Bypass Flow Control Valve CNM-FCV200 was not properly locked open.

The inspectors examined the significance of this issue by completing Phases 1 and 2 Significance Determination Process (SDP) worksheets in accordance with NRC Manual Chapter 0609. The evaluations assumed that Valve CNM-FCV200 would have failed shut during any full power reduction event, that this condition existed for approximately 4 months, that the packing resistance did not change appreciably over that duration, and that the valve failing closed would result in a complete loss of condensate and feedwater flow to the reactor. The existing condition was the result of a performance deficiency (failure to properly lock open Valve CNM-FCV200) affecting the Mitigating Systems cornerstone and represented a loss of safety function of non-Technical Specification equipment, specifically the condensate and feedwater systems. The dominant accident sequences identified from the Phase 2 SDP River Bend Station site-specific risk-informed notebooks included Table 3.1, Transients (Reactor Trip), Sequence 4; Table 3.11, Loss of 120 VDC Emergency Division I, Sequences 1, 2, and 3; and Table 3.12, Loss of 120 VDC Emergency Division II, Sequences 1 and 2. The issue was mitigated by the fact that all emergency core cooling systems were available. The Phase 2 SDP evaluation determined that the issue was potentially of greater than minor safety significance. The Phase 2 analysis was validated by a regional Senior Reactor Analyst, and a Phase 3 analysis was initiated. The Phase 3 analysis was in progress at the issuance of this report. Therefore, the final significance of this issue is to be determined.

c. Findings

No findings of significance were identified.

3.7 10 CFR 50.72 Report Evaluation

a. Inspection Scope

The inspectors reviewed the 10 CFR 50.72 report submitted by the licensee to determine whether it satisfied the subject reporting requirements.

b. Background

The reactor scram occurred at 8:25 p.m. on September 18, 2002. The NRC Operations Center was notified at 11:41 p.m. in accordance with 10 CFR 50.72 requirements (Event Notification 39200).

The licensee completed NRC Form 361, “Reactor Plant Event Notification Worksheet,” and transmitted a copy to the NRC Operations Center, in addition to telephone notification. Several observations were noted by the inspectors in their review of the completed Form 361. In the description section of the form, the licensee stated “The plant systems performed as required post scram.” The cause of the scram was stated as being “still under investigation.” In addition, the section of the Event Notification Worksheet asking “Anything Unusual or Not Understood” was checked “No.” The question “Did All Systems Function As Required” was checked “Yes.”

The licensee indicated that their interpretation of the required reporting information was to focus on the reactor/core, emergency core cooling systems, and other safety-related systems. The condensate and feedwater systems, the CRD system, the CST, and RWCU were not safety-related systems, and the licensee did not believe those systems needed to be addressed as long as the reactor was shut down safely and being maintained in a safe and shut down condition. With this view in mind, the Event Notification stated that plant systems performed as required post scram, that nothing unusual or not understood occurred, and that all systems functioned as required.

The inspectors reviewed the reporting requirements identified in 10 CFR 50.72 and the guidance provided in NUREG 1022, “Event Reporting Guidelines 10 CFR 50.72 and 50.73." Although no regulatory noncompliance was identified, the report to the Headquarters Operations Center was lacking detail and information regarding the complexity of the event and the impact the event had on nonsafety-related, although important, systems.

b. Findings

No findings of significance were identified.

4 Exit Meeting Summary

The inspectors presented the inspection results to Paul D. Hinnenkamp, Vice President - Operations, and other members of licensee management during an exit meeting on November 14, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether or not any materials discussed during the exit should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

B. Biggs, Coordinator, Licensing
W. Brian, Director, Engineering
D. Burnett, Superintendent, Chemistry
T. Gates, Manager, System Engineering
W. Holland, Radiation Protection Outage Coordinator
J. Leavines, Manager, Licensing
T. Lynch, Manager, Operations
J. Malara, Manager, Design Engineering
J. McGhee, Manager, Maintenance
D. Mims, General Plant Manager
P. Page, Supervisor, Health Physics
W. Spell, Senior Environmental Specialist
T. Trepanier, Assistant General Manager
W. Trudell, Manager, Corrective Action and Assessment
D. Wells, Superintendent, Radiation Protection

Operators Interviewed

David Bowman, Turbine Building Operator
Kevin Burnett, Auxiliary Control Room Operator
Forrest Drummond, Radwaste Operator/Auxiliary Control Room Operator
Ken Jelks, Control Building Operator
Brian Kelley, Operations Shift Manager
Lemar Palmer, Outside Operator
Eric Pickrell, Reactor Building Operator
Scott Shultz, Unit Operator
Erich Weinfurter, Shift Technical Advisor
Terry Wymore, Control Room Supervisor

ITEMS OPENED AND CLOSED

Opened

50-458/2002-07-01	AV	Failure to properly lock open Valve CNM-FCV200

Closed

None

LIST OF ACRONYMS AND INITIALISMS USED

APRM	average power range monitor
BPV	turbine bypass valves
CFR	Code of Federal Regulations
CR-RBS	River Bend Station Condition Report
CRD	control rod drive
CST	condensate storage tank
HCU	hydraulic control units
NEO	nuclear equipment operator
NR	narrow-range
NRC	U.S. Nuclear Regulatory Commission
OSRC	operational safety review committee
RCIC	reactor core isolation cooling system
RFP	reactor feedwater pump
RWCU	reactor water cleanup system
SDP	significance determination process
SJAE	steam jet air ejector
SRA	senior reactor analyst
WR	wide-range

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

ATTACHMENT TO NRC INSPECTION REPORT 50-458/02-07

September 23, 2002

MEMORANDUM TO: Michael O. Miller, Resident Inspector, River Bend Station

FROM: Ken E. Brockman, Director, Division of Reactor Projects

SUBJECT: SPECIAL INSPECTION CHARTER TO EVALUATE THE RIVER BEND STATION REACTOR TRIP WITH COMPLICATIONS

In response to the reactor trip that occurred at the River Bend Station on September 18, 2002, and the subsequent complications due to the isolation of the condensate system and the loss of physical integrity of the Steam Jet Air Ejector condensers, a Special Inspection Team is being chartered. You are hereby designated as the Special Inspection Team leader. The Special Inspection Team will consist of yourself; Mr. James Drake, Reactor Engineer; and James Dodson, Reactor Inspector. Additional regional resources are available for consultation as needed.

A. Basis

At approximately 8:24 p.m. on September 18, 2002, the River Bend Station scrammed from 100 percent reactor power due to a high average power range monitor flux trip. All control rods fully inserted in the core. The cause of the reactor trip and the details surrounding the subsequent complications in the plant response are currently under investigation by the licensee. Preliminary evidence indicates that a failure of a control card within the electrohydraulic control system, resulting in rapid cycling of the turbine generator bypass valves, may have been the cause for the high power scram. Following the scram, both steam jet air ejector condensers experienced gasket failures on the end bells of the condensers, and all reactor feed pumps tripped on low suction pressure. Operators responded to the condensate system gasket failure by securing all condensate pumps and manually isolating the steam jet air ejector condensers. With the loss of condensate to the reactor vessel, operators manually initiated the reactor core isolation cooling system to maintain the appropriate reactor level and remove decay heat.

Upon securing the condensate pumps, the control rod drive pumps automatically aligned suction to the condensate storage tank. Following this transfer, alarms in the control room indicated high differential pressure conditions on the control rod drive pumps' suction and discharge filters. Operators bypassed the filters and were able to regain flow. The high differential pressure condition was a result of particulate in the condensate storage tank being collected in the control rod drive system filters. The transient causing the steam jet air ejector condenser gasket failure and loss of feed pump suction pressure were both apparently caused by an inadvertent closure of the condensate full flow filter bypass valve following the scram.

B. Scope

The team is expected to perform fact-finding in order to address the following:

- Develop a complete sequence of events related to the September 18, 2002, reactor trip.

- Review the licensee's root cause determination for completeness and accuracy. Independently verify key assumptions and facts.

- Evaluate the adequacy of the operator response to the transient (i.e., timeliness in initiating manual reactor trip, emergency operating procedure usage, etc.).

- Evaluate the accuracy and completeness of the licensee's 10 CFR 50.72 report.

- Review the adequacy of the posttrip review. Include in this review the thoroughness of their assessment of the event and whether potential complications on the plant systems (i.e., extent of conditions) were properly considered, the quality and adequacy of the operability evaluations, and the comprehensiveness and appropriateness of the immediate and long-term corrective actions.

- Review the licensee's risk analysis of the event.

- Review the event to determine whether there are any generic impact issues related to the condensate storage tank design of venting to the atmosphere, controls of all operated valves, foreign material exclusion, and the appropriateness and concerns of an unmonitored release point.

C. Guidance

Inspection Procedure 93812, "Special Inspection," provides additional guidance to be used by the Special Inspection Team. Your duties will be as described in Inspection Procedure 93812. During performance of the Special Inspection, designated team members are separated from their normal duties and report directly to you. The team is to emphasize fact-finding in its review of the circumstances surrounding the event, and it is not the responsibility of the team to examine the regulatory process. Safety concerns identified that are not directly related to the event should be reported to the Region IV office for appropriate action.

The Team will report to the site, conduct an entrance, and begin inspection no later than Tuesday, September 24, 2002. Tentatively, the inspection should be completed by the close of business on September 27, 2002, with a report documenting the results of the inspection issued within 30 days of the completion of the inspection. While the team is on site, you will provide daily status briefings to Region IV management, who will coordinate with NRR to ensure that all other parties are kept informed.

This Charter may be modified should the team develop significant new information that warrants review. Should you have any questions concerning this Charter, contact Ken Brockman, Director, Division of Reactor Projects at (817) 860-8248.

cc:
E. Merschoff
T. Gwynn
S. Morris
W. Ruland
A. Howell
M. Webb
A. Gody
D. Graves
P. Alter