UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

October 25, 2001

Randal K. Edington, Vice President - Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: RIVER BEND STATION--NRC INTEGRATED INSPECTION REPORT 50-458/01-03

Dear Mr. Edington:

On September 29, 2001, the NRC completed inspections at your River Bend Station facility. The enclosed integrated inspection report documents the inspection findings which were discussed on October 4, 2001, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Since September 11, 2001, the River Bend Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to Entergy Operations, Incorporated. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

Based on the results of this inspection, the NRC has identified four findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with two of these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violation or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the River Bend Station facility.

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/NRC/ADAMS/index.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/

William D. Johnson, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-458
License: NPF-47

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NRC Inspection Report
50-458/01-03

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-458
License:	NPF-47
Report No.:	50-458/01-03
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	June 24 through September 29, 2001
Inspectors:	P. J. Alter, Senior Resident Inspector S. M. Schneider, Resident Inspector M. E. Murphy, Senior Reactor Engineer, Operations Branch L. T. Ricketson, P.E., Senior Health Physicist, Plant Support Branch
Approved By:	W. D. Johnson, Chief, Project Branch B
ATTACHMENT:	Supplemental Information

SUMMARY OF FINDINGS

River Bend Station
NRC Inspection Report 50-458/01-03

IR 05000458-01-03; on 06/24/2001-09/29/2001; Entergy Operations, Inc; River Bend Station. Integrated Resident & Regional Report. Fire Protection, Maintenance Risk Assessment, Postmaintenance Test, and ALARA Planning and Controls. Two Green NCVs and two Green Findings.

The inspections were conducted by the resident inspectors, a regional health physicist inspector, and a regional operations inspector. The inspections identified four Green findings, two of which were noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or 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/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The licensee did not maintain a 3-hour rated fire barrier as described in the plant fire hazards analysis. Specifically, the inspectors identified a penetration into a 3-hour rated floor barrier in the standby cooling tower that had not been sealed.

  The inspectors determined that the safety significance of the degraded fire barrier was very low since it did not separate redundant safe shutdown equipment. The failure to maintain a 3-hour rated fire barrier as described in the Fire Hazards Analysis is a noncited violation of Attachment 4 to Facility Operating License NPF-47. This violation is documented in the licensee's corrective action program as CR-RBS-2001-0898 (Section 1R05).

- Green. The inspectors identified deficiencies with the conduct of maintenance risk assessments of planned and emergent work. Specifically, inadequate risk assessments were identified, a plant component was not identified by the licensee to be in the quantitative risk assessment tool, an opportunity to identify an error in the risk assessment tool was missed, and corrective actions taken for prior inadequate risk assessments failed to preclude the recently identified deficiencies.

  The inspectors determined that the safety significance of the maintenance risk assessment deficiencies was very low in that there was no actual loss of safety function and that the difference between the actual plant risk and the licensee determined risk was small enough such that significant risk management actions would not have been required. This finding is documented in the licensee's corrective action program as CR-RBS-2001-0674 (Section 1R13).

Cornerstone: Initiating Events

- Green. The inspectors identified that the licensee failed to specify or document postmaintenance test requirements in two main feedwater pump seal replacement work packages. The failure to specify and document postmaintenance testing (PMT) for maintenance work activities precluded the ability to evaluate test results to ensure the affected equipment was capable of performing its design function. The inspectors determined that corrective actions for prior PMT program deficiencies failed to preclude the recently identified deficiencies.

  The safety significance of the failure to specify or document postmaintenance test requirements in the two feedwater pump work packages was very low. The issue would not contribute to both the likelihood of an initiating event and the failure of mitigating equipment. Only two of the three main feedwater pumps were affected and only one main feedwater pump is required for mitigation of the reactor trip transient. This finding is documented in the licensee's corrective action program as CR-RBS-2001-0695 (Section 1R19).

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified a noncited violation of very low safety significance because the licensee's work control process failed to ensure that all work activities were reviewed to identify opportunities to reduce radiation doses. The failure resulted from the lack of an implementing procedure that required the review of temporary electrical power installations to take into account factors for minimizing radiation exposure to workmen, in violation of Technical Specification 5.4.1. A total of 94 temporary power installations were scheduled for the outage but had not been reviewed. Three installations had been completed before the identification of the problem. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. The licensee placed this item in its corrective action program as CR-RBS-2001-1149.

  The failure to implement dose saving measures had a credible impact on safety. The occurrences involved workers unplanned, unintended doses that resulted from actions that were contrary to licensee procedures and Technical Specifications. However, the safety significance was determined to be very low because there was no exposure in excess of regulatory limits or significant potential for exposure in excess of regulatory limits (Section 2OS2).

B. <u>Licensee Identified Findings</u>

None

Report Details

Summary of Plant Status: The reactor was operated at 100 percent power from the beginning of the inspection period until shutdown at the beginning of a planned refueling outage on September 23, 2001. The plant remained shut down for the remainder of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors assessed the licensee's severe weather operations and hurricane preparedness guidelines at the beginning of hurricane season. Design features and implementation of procedures for the protection of mitigating systems from adverse weather conditions were evaluated. Emergency stores (e.g., meals ready to eat (MRE), blankets, water, etc.) required to be on site to support site operations and personnel in the event of severe weather were visually verified to be on site and MRE expiration dates were verified to be acceptable. The inspectors reviewed the following documents and procedures as part of this assessment:

- AOP-0029, "Severe Weather Operation," Revision 14
- RBNP-0089, "Hurricane Readiness," Revision 02
- Updated Safety Analysis Report (USAR)
- Weather advisories and operations shift logs documenting tropical storm, "Allison"
- 2001 emergency preparedness recurring tasks
- Shipping and material receipt records for survival equipment and first aid supplies
- MRE shelf life documentation

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed safety-related system walkdowns to verify equipment alignment and discrepancies that impact the function of the system and potentially increase risk. The inspectors also verified that the licensee has properly identified and resolved equipment alignment problems that could impact mitigating system availability.

.1 125 Vdc Engineered Safety Feature Distribution System Walkdown

The inspectors performed a complete system walkdown of all three divisions of the 125 Vdc engineered safety feature distribution system. Specifically, the inspectors: (1) reviewed the listed documents to determine the correct system lineup; (2) reviewed outstanding maintenance work requests to ensure that no deficiencies exist that could affect the ability of the system to perform its safety function; and (3) reviewed outstanding design issues, including temporary modifications, operator workarounds, and pending design changes.

- System Operating Procedure SOP-0049, “125 VDC System,” Revision 16
- USAR Section 8.3.2, “DC Power Systems”
- Technical Specifications Section 3.8, “Electrical Power Systems”
- 125 Vdc distribution system health report and maintenance rule report

Additionally, the inspectors sampled the licensee’s corrective action program to ensure that the licensee has identified equipment alignment problems at the appropriate threshold and evaluated their resolution for risk significant systems. Condition Reports (CRs) reviewed included:

- CR-RBS-2000-1623, reply to IN-94-080, “Inadequate DC Ground Detection“
- CR-RBS-2001-0267, “DC motor operated valve calculations non-conservative”

.2 Reactor Core Isolation Cooling (RCIC) System Walkdown

On July 19, 2001, the inspectors performed a partial system walkdown of RCIC while high pressure core spray was out of service. The inspectors reviewed System Operating Procedure SOP-0035, “Reactor Core Isolation Cooling System,” Revision 21, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.3 Low Pressure Core Spray (LPCS) System Walkdown

On September 20, 2001, the inspectors performed a partial system walkdown of the LPCS while Division II low pressure coolant injection systems were unavailable. The inspectors reviewed System Operating Procedure SOP-0032, “Low Pressure Core Spray,” Revision 18A, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

.4 Division I Standby Service Water (SSW) System Walkdown

On September 18, 2001, the inspectors performed a partial system walkdown of the Division I SSW while Division II SSW was unavailable. The inspectors reviewed Standard Operating Procedure SOP-0042, “Standby Service Water System,” Revision 18F, to determine the correct system lineup. Then the inspectors walked down critical portions of the system to identify any discrepancies between the existing equipment lineup and the correct lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Throughout the period the inspectors toured the following plant areas important to reactor safety to observe conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational lineup, and operational effectiveness of fire protection systems, equipment and features; and (3) the material condition and operational status of fire barriers used to prevent fire damage or fire propagation.

- RCIC pump room
- Standby cooling tower Divisions I and II switchgear rooms
- Standby cooling tower Divisions I and II SSW pump rooms
- Division I 125 Vdc battery and inverter rooms
- Division II 125 Vdc battery and inverter rooms
- Division III 125 Vdc battery and inverter rooms
- LPCS pump room

The inspectors reviewed the following documents during the fire protection inspections:

- Pre-Fire Strategy Book
- USAR Section 9A.2, “Fire Hazards Analysis”
- River Bend postfire safe shutdown analysis

b. Findings

The inspectors identified a noncited violation of Attachment 4 to Facility Operating License NPF-47, which involved a degraded fire barrier between Fire Area PT-1 (E, F, and G Tunnels) and Fire Area PH-2/Z-2 (standby cooling tower Division II Standby Service Water switchgear room). These areas contained redundant equipment for safe shutdown. The issue was determined to be of very low safety significance (Green).

On August 1, 2001, the inspectors identified a section of 4-inch diameter, vertical pipe projecting through the floor of Fire Area PH-2/Z-2 which did not contain any sealant material. The USAR Fire Hazards Analysis, Figure 9A.2-6, "Fire Area Boundaries Standby Service Water Pump House and Cooling Tower," shows this floor area to be a 3-hour rated fire barrier. The inspectors contacted fire protection personnel who inspected the affected fire barrier, determined the fire barrier was inoperable, and initiated Condition Report CR-RBS-2001-0898. The licensee concluded that this vertical pipe was an unused equipment drain that had not been sealed. The vertical pipe penetration was subsequently sealed and the fire barrier returned to an operable condition.

A noncited violation (NCV 50-458/0016-03) discussed a previous inspector identified fire barrier degradation at River Bend. The NCV identified that the licensee conducted periodic fire barrier inspections as required by Technical Requirement 3.7.9.6, "Fire-rated Assemblies," Surveillance Requirement 3.7.9.6.6, every 18 months. Procedure STP-000-3602, "Fire Barrier Visual Inspection," Revision 11B, provided guidance on conducting floor, wall, and ceiling inspections, including checking that fire barriers are free of damage or defects such as cracks, separations, gouges, holes or openings, and each penetration is sealed. The inspectors identified that CR-RBS-2000-1944, which was written in response to this NCV simply addressed repairing the fire barrier. It did not include an assessment of the effectiveness of the fire barrier inspection program. The inspectors determined that this was a missed opportunity by the licensee to evaluate the effectiveness of their fire barrier inspection program.

The inspectors determined that the unsealed penetration into the fire barrier had a credible impact on safety and involved a degradation of a fire protection feature. Since the finding involved the degradation of a fire barrier which affected SSW safe shutdown equipment, the inspectors evaluated the significance of the finding in accordance with inspection manual (IMC) Chapter 0609, "Significance Determination Process." The inspectors determined that the degraded fire barrier was not an impairment or degradation of a fire barrier used to protect safe shutdown capability, since the licensee takes credit for the normal service water system as a redundant safe shutdown system for SSW for a fire in Fire Area PT-1. Therefore, the finding screens out as having very low safety significance (Green).

Attachment 4 to Facility Operating License NPF-47, specified that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report. Figure 9A.2-6 of the Updated Safety Analysis Report (USAR) Fire Hazards Analysis identifies this area of the floor of Fire Area PH-2/Z-2 to be a 3-hour rated floor barrier separating the fire area from adjacent areas. The inspectors determined that the failure to maintain the 3-hour rated floor barrier was a violation of Attachment 4 to Facility Operating License NPF-47 (NCV 50-458/0103-01). This violation is associated with an inspection finding that is characterized by the SDP as having very low safety significance (Green) and is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR-RBS-2001-0898.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors conducted a periodic flooding assessment to verify that the licensee's flooding mitigation plans and equipment were consistent with design requirements and risk analysis assumptions. The inspectors conducted a walkdown of the auxiliary building 70' elevation crescent area on September 15, 2001. Specifically, the inspectors examined: (1) sealing surfaces of watertight doors, (2) sealing of equipment below design flood level, (3) sealing of penetrations in floors and walls, (4) operable sump pumps and level alarm circuits, (5) interconnections with common drain systems and (6) sources of potential internal flooding from plant systems. The inspectors reviewed the following documents during the inspection:

- River Bend individual plant examination of external events
- USAR Section 3.4.1, “Flood Protection”
- G13.18.12.3*15, “Internal Flooding Screening Analysis”
- G13.2.3 PN-317, “Max Flood Elevations for Moderate Energy Line Cracks in Cat I Structures”
- ER-98-0444, “Research into Design [Requirements] of Doors AB076-01 and AB076-02"

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

.1 Biennial Requalification Program Inspection

During the week of June 25, 2001, the inspector: (1) evaluated examination security measures and procedures for compliance with 10 CFR 55.49; (2) evaluated the licensee's sample plan for the written examinations for compliance with 10 CFR 55.59 and NUREG-1021, as referenced in the facility requalification program procedures; and (3) evaluated maintenance of license conditions for compliance with 10 CFR 55.53 by review of facility records, procedures, and tracking systems for licensed operator training, qualification, and watchstanding. The inspector also reviewed remedial training and examinations for examination failures for compliance with facility procedures and responsiveness to address areas failed.

In addition, the inspector: (1) interviewed seven personnel (two operators, three instructors/evaluators, and two training supervisors) regarding the policies and practices for administering examinations; initiating and incorporating feedback from plant and industry events; developing and administering remedial training and retake examinations; (2) observed the administration of three dynamic simulator scenarios to two requalification crews by facility evaluators, including an operations department manager, who participated in the crew and individual evaluations; and (3) observed three facility evaluators administer five job performance measures. Each job performance measure was observed being performed by an average of two requalification candidates. The inspector also reviewed the remediation process for two individuals, one of which involved a written examination failure and one a simulator examination failure.

.2 Quarterly Requalification Training Inspection

On July 3, 2001, the inspectors observed simulator training of an operating crew, as part of the operator requalification training program, to assess licensed operator performance and the training evaluator's critique. Emphasis was placed on observing weekly evaluation exercises of high risk licensed operator actions, operator activities associated with the emergency plan, and lessons learned from industry and plant experiences. In addition, the inspectors compared simulator control panel configurations with the actual control room panels for consistency, including recent modifications implemented in the plant.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed structure, system, or component (SSC) performance problems to assess the effectiveness of the licensee's maintenance efforts for SSCs scoped under the licensee's maintenance rule program. The inspectors verified the licensee's implementation of the maintenance rule (10 CFR 50.65) for the performance problems reviewed by answering the following questions: (1) was the SSC scoped for monitoring in accordance with 10 CFR 50.65; (2) was the SSC assigned the proper safety significance; (3) were the problems characterized properly; (4) as a result of the problems, was the SSC assigned the proper classification under 10 CFR 50.65; and (5) were the appropriate performance criteria established for the SSC or, when necessary, were appropriate goals set and corrective actions taken to restore the SSC status under the maintenance rule. The following documents were reviewed as part of this assessment:

- CR-RBS-2001-1119, containment airlock inner door seal leak rate failure
- CR-RBS-2001-1120, containment air lock outer door seal leak rate failure
- CR-RBS-2001-0853, standby cooling tower ventilation damper failures

- CR-RBS-2000-1134, restoration of SSW system to maintenance rule (a)(2) status
- CR-RBS-2001-0898, inoperable fire barrier in standby cooling tower
- Calculation G13.18.3.6*12, 10 CFR Part 50 Appendix R Analysis of Fire Area PT-1
- NUMARC 93-01, Revision 2, Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
- River Bend maintenance rule function list
- River Bend maintenance rule performance criteria list

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed maintenance activities to verify the performance of assessments of plant risk related to planned and emergent maintenance work activities. The inspectors verified: (1) the adequacy of the risk assessments and the accuracy and completeness of the information considered; (2) management of the resultant risk and implementation of work controls and risk management actions; and (3) effective control of emergent work, including prompt reassessment of resultant plant risk.

.1 Risk Assessment and Management of Risk

On a routine basis, the inspectors verified performance of risk assessments, in accordance with administrative Procedure ADM-096, “Risk Management Program Implementation and On-Line Maintenance Risk Assessment,” Revision 01, for planned maintenance activities and emergent work involving SSCs within the scope of the maintenance rule. Specific work activities evaluated included planned and emergent work for the weeks of June 4, July 23, and August 20, 2001.

.2 Emergent Work Control

During emergent work, the inspectors verified that the licensee took actions to minimize the probability of initiating events, maintained the functional capability of mitigating systems, and maintained barrier integrity. The inspectors also reviewed the emergent work activities to ensure the plant was not placed in an unacceptable configuration. Specific emergent work activities evaluated included: (1) the failure of a reactor pressure transmitter; and (2) the unexpected inoperability of the SSW system.

b. Findings

The inspectors identified deficiencies in the implementation of the risk assessment program associated with scheduled and emergent maintenance. The issue was determined to be of very low safety significance (Green).

On June 4, 2001, the inspectors independently performed a risk assessment of scheduled work involving the RCIC system. The scheduled maintenance caused the system to be unavailable during the performance of a surveillance test procedure. Separate maintenance on a closed cooling water pump was scheduled during the same time period that RCIC would be unavailable. The inspectors used the licensee’s quantitative risk evaluation tool (EOOS) for this configuration and determined that the combined risk of both activities would result in a plant safety index (PSI) of 9.0. The risk profile published by the licensee for this maintenance showed a PSI of 9.7, which required no risk management activities. For a PSI less than 9.4, Administrative Procedure ADM-0096, “Risk Management Program Implementation and On-Line Maintenance Risk Assessment,” Revision 01, required the licensee to take measures to ensure that subsequent maintenance activities would not increase risk to a higher level. The licensee subsequently revised the surveillance test procedure to include dedicated operator controls in order to maintain RCIC available and increased PSI to 9.7. The inadequate risk assessment was documented in CR-RBS-2001-0674.

On July 11, 2001, the inspectors independently performed a risk assessment of emergent work for a failed reactor pressure instrument Loop B21-PTNO68F and B21-PISN668F. The licensee had conducted a risk assessment of the plant configuration and concluded that there was no increased risk and that PSI was 10.0. The inspectors, using the EOOS program, identified the PSI to be 6.7 with reactor pressure Transmitter B21-PTN068F out of service. The inspectors notified the operations shift manager, who then performed an EOOS assessment and came up with the same result (PSI of 6.7). The shift manager took immediate risk management actions in accordance with Procedure ADM-0096 to notify senior management, place all work on the opposite division on hold, evaluate all maintenance activities currently authorized, and ensure reactor pressure Instrument B21-PISN668F repair was expedited.

The shift manager also requested that the plant safety analysis engineers evaluate this EOOS result. The safety analysis engineers subsequently determined that the reactor pressure transmitter failure was incorrectly mapped to a common mode failure of several pressure instruments which would render all trains of low pressure coolant injection and spray inoperable. Since the mapping was incorrect, the licensee removed the component from the EOOS. The licensee recognized at that time that there would be some impact on plant risk from this plant configuration, but could not quantify the impact. The failure to identify this equipment in the EOOS model and the EOOS model error were documented in CR-RBS-2001-0814.

On August 31, 2001, the inspectors met with licensee safety analysis engineers to discuss the risk associated with the potential failure conditions of the reactor pressure instrument (failed low and failed high). With the instrument “failed high,” the licensee determined that there was a very small contribution to core damage frequency since two independent instrument trains would have to fail and recovery action to manually open low pressure coolant injection valves would be successful. For the “failed low” condition, the concern would be that an interfacing system loss of coolant accident (LOCA) could occur if one instrument failed with a LOCA signal present. This could subject the low pressure portions of the coolant injection systems to full reactor pressure. The licensee concluded that, due to the low frequency associated with a LOCA and the fact that a LOCA contributes to less than 1.1 percent of the total core damage frequency, the “failed low” condition could be better addressed qualitatively and the instrument should not be included in the EOOS software program. The licensee has added corrective actions to CR-RBS-2001-0814 to evaluate the EOOS software for other potential modeling deficiencies.

The inspectors determined that the licensee had previously identified an inadequate risk assessment in CR-RBS-2000-1611, dated September 12, 2000. One corrective action for this CR added peer checks of risk assessments by work week managers for planned work schedules. CR-RBS-2001-0730, dated June 15, 2001, also cited peer checks by work week managers as a corrective action. This CR addressed several prior risk assessment CRs and recommended an evaluation of the risk assessment program. CR-RBS-2001-0854, dated July 23, 2001, documented another licensee identified inadequate risk assessment of scheduled work. Additionally, following discussions with operations department supervision, the inspectors determined that on-shift control room personnel do not routinely peer check risk assessments of emergent equipment problems.

The inspectors consider the above deficiencies to indicate a recurring situation that was not isolated and where prior corrective actions were ineffective. These examples reflect weaknesses in the implementation of the maintenance risk assessment program. The inspectors determined that deficiencies with risk assessments have a credible impact on safety since inadequate risk assessments could result in higher risk plant configurations without appropriate risk management actions in place to mitigate the higher risk conditions. The failure to recognize actual plant risk and thereby specify appropriate risk management actions can affect the availability of mitigating systems.

The inspectors conducted a significance determination of the finding in accordance with IMC 0609, “Significance Determination Process.” For the RCIC issue, a high pressure safety injection system was affected. For the failed reactor pressure instrument issue, low pressure coolant injection systems were affected. In both cases, the Phase 1 SDP screened these issues as having a very low safety significance (Green). There was not a design or qualification deficiency resulting in a loss of function. There was not an actual loss of safety function of a system since the inadequate risk assessment for RCIC was identified by the inspectors prior to the conduct of the maintenance, and the failed low reactor pressure instrument would not have prevented low pressure system injection on an actual LOCA signal. Also, the difference in actual plant risk and the licensee determined risk was small enough such that significant risk management actions would not have been required. Neither issue resulted in a loss of safety function for greater than the equipment Technical Specification allowed outage time nor was there a loss of risk significant, non-Technical Specification equipment for greater than 24 hours. No seismic, fire, flooding, or severe weather initiating event potential was identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors reviewed and observed personnel performance following the isolation of the auxiliary building exhaust fans on July 18, 2001. The inspectors also reviewed Procedures EOP-002, “Primary Containment Control,” Revision 13, EOP-003, “Secondary Containment Control,” Revision 12, and SOP-0059, “Containment HVAC System,” Revision 20.

The inspectors evaluated the initiating causes of the event as documented in CR-RBS-2001-0839. In addition, the inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred and that operators responded in accordance with plant procedures and training.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed six operability evaluations performed by the licensee for risk significant systems to determine that the operability was justified, such that availability was assured, and no unrecognized increase in risk has occurred. Specific areas evaluated included: (1) the technical adequacy of the evaluation; (2) whether other existing degraded conditions were considered; and (3) if operability was based on compensatory measures, were these measures in place and would they work.

- CR-RBS-2001-0853, standby cooling tower ventilation damper failures
- CR-RBS-2001-0961, air bound startup fuel oil pump on Division II emergency diesel generator
- CR-RBS-2001-0908, standby gas treatment automatic initiation
- CR-RBS-2001-0930, containment airborne radiation monitor loss of communications
- CR-RBS-2001-0995, normal service water pump deficiencies
- CR-RBS-2001-1006, reevaluation of normal service water pump deficiencies

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the PMT requirements specified for the Maintenance Action Items (MAI) listed below to ensure that testing activities were adequate to verify system operability and functional capability:

- MAI 346717, set limit switches and perform signature testing for E12-MOVF094
- MAI 348960, clean, inspect, insulation test and lubricate E12-MOVF094
- MAI 346756, replace inboard and outboard mechanical seals on feedwater Pump FWS-P1A
- MAI 345116, replace inboard and outboard mechanical seals on feedwater Pump FWS-P1B
- MAI 348141, troubleshoot and repair reactor pressure instrument Loop B21-PTNO68F and B21-PISN668F
- MAI 349175, manual operation of inclined fuel transfer system lower upender and bottom valve
- MAI 342321, inclined fuel transfer tube level transmitter replacement

b. Findings

The inspectors identified that the licensee failed to specify and document PMT in two main feedwater pump work packages. The issue was determined to be of very low safety significance (Green).

On May 31, 2001, a main feedwater pump mechanical seal failed, requiring a reduction of power to approximately 80 percent to stay within the capacity of the two remaining main feedwater pumps. On June 3, 2001, the licensee completed replacement of the mechanical seals on two of the three main feedwater pumps.

On June 7, 2001, the inspectors reviewed the work packages for the mechanical seal replacement for the two main feedwater pumps. MAIs 346756 and 345116 performed mechanical seal replacement for main feedwater Pumps A and B, respectively. The inspectors identified that no PMT was specified by planning and scheduling who developed the work packages and that no documentation of completion of PMT by mechanical maintenance technicians was included in the completed work packages. The inspectors notified the licensee who subsequently stated in-process maintenance activities provided verification that the mechanical seals and the main feedwater pumps were functioning properly. This failure to specify and document PMT for the main feedwater pump seal replacement is documented in the licensee's corrective action program as CR-RBS-2001-0695.

NCV 50-458/0010-01 documented the failure of planning and scheduling personnel to identify PMT requirements in four maintenance work packages for safety-related systems. This issue is documented in the licensee's corrective action program as CR-RBS-2000-1010 and CR-RBS-2000-1199. As a result, the licensee implemented a review of their PMT program and instituted corrective actions to address identified deficiencies. One corrective action the licensee implemented as a result of that review was to include an operations department check of required test conditions on the PMT specification sheet included in all maintenance work packages. The inspectors determined that there were several different interpretations by maintenance and operations personnel on how to fulfill this new requirement. Some work packages were completed without the required test conditions block filled in on the PMT specification sheet. Neither maintenance nor operations personnel documented this situation in the corrective action program nor did they notify their management of the problems they were experiencing meeting this new requirement. On April 25, 2001, the planning and scheduling manager became aware of this condition during an internal PMT program effectiveness audit and documented the problem in a program improvement database. The inspectors consider these deficiencies indicative of a recurring situation that was not isolated and where prior corrective actions were ineffective.

The failure to specify and document PMT for maintenance work activities precludes the ability to evaluate test results to ensure the affected equipment was capable of performing its design function. The inspectors determined that the failure to specify and document PMT for the main feedwater pump seal replacement maintenance work packages could be reasonably viewed as a precursor to a significant event. The failure to evaluate the satisfactory performance of the replacement of the mechanical seals of main feedwater Pumps A and B could cause an initiating event (low water level reactor scram). As such, this is considered a transient initiator contributor (reactor trip) and Phase 1 SDP characterizes the issue as very low safety significance (Green). The issue is not considered to contribute to the likelihood of a primary or secondary LOCA initiator. The issue does not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions will not be available, since only one of the three main feedwater pumps is required for mitigation of the reactor trip transient. This issue is not considered to increase the likelihood of a fire or internal/external flooding.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed licensee refueling outage planning and execution activities. The inspectors' review included scheduling, training, outage configuration management, decay heat removal operation and management, reactivity controls, inventory controls, tag-out and clearance activities, foreign material exclusion management, and fuel movement and storage. Specific activities monitored included:

- Reactor scram and start reactor cooldown
- Shutdown cooling operations
- Division II emergency core cooling system testing
- Control rod drive mechanism uncoupling
- Division II engineered safety features battery replacement
- As-found local leak rate testing of main steam isolation valves
- Core alterations

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors verified, by witnessing and reviewing test data, that selected risk significant systems and component surveillance tests met Technical Specification, USAR, and procedure requirements. The inspectors ensured that surveillance tests demonstrated that the systems were capable of performing their intended safety functions and provided operational readiness. The inspectors specifically evaluated surveillance tests for preconditioning, clear acceptance criteria, range, accuracy and current calibration of test equipment and verified that equipment was properly restored at the completion of the testing. The inspectors reviewed and or observed the following surveillance tests and documents:

- STP-122-6302, “Division II Instrument Air Quarterly Valve Operability Test,” Revision 04B.

- MAI 3455504, functional test of station blackout Valve SWP-AOV599

- STP-508-0201, “Manual Scram Functional Test,” Revision 08

- STP-052-3701, “Control Rod Scram Testing,” Revision 17

- REP-0026, “Manual [Core Thermal Power] Monitor,” Revision 04

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

On September 19, 2001, the inspectors reviewed the temporary modification made to the inclined fuel transfer system (IFTS) system to provide alternate IFTS bottom Valve F42-F004 position indication for the IFTS control logic. Specifically the inspectors:

(1) reviewed the temporary modification and its associated 10 CFR 50.59 screening against the systems design basis documentation, including the USAR and Technical Specifications; (2) verified that the installation of the temporary modification was consistent with the modification documents; (3) verified that plant drawings and procedures were updated; and (4) reviewed the postinstallation test results to confirm the actual impact of the temporary modification on the affected system had been adequately verified.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the licensee's July 17, 2001, emergency preparedness drill in order to evaluate the conduct of the drill and the adequacy of the licensee critique of performance to identify weaknesses and deficiencies in classification notification and protective action recommendation development. The following procedures and documents were reviewed during the assessment:

- EIP-2-001, "Classification of Emergencies," Revision 11
- EIP-2-006, "Notifications," Revision 27
- EIP-2-007, "Protective Action Guidelines Recommendations," Revision 18
- EP-M-01-033, "Drill Evaluation Report, ERO Team B," dated August 3, 2001

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs to obtain information to meet the inspection objectives. Independent radiation surveys of selected work areas within the controlled access area were performed. The following items were reviewed and compared with regulatory requirements:

- ALARA program procedures

- Processes used to estimate and track exposures

- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information

- Five radiation work permit packages for outage work activities which were projected to accrue the highest collective exposures during the outage (2001-1800, 2001-1912, 2001-1917, 2001-1933, and 2001-1950)

- Use of engineering controls to achieve dose reductions

- Individual exposures of selected work groups (health physics, electrical maintenance, and mechanical maintenance)

- Hot spot tracking and reduction program

- Radiological work planning

- ALARA Committee meeting minutes since the previous inspection

- Self-assessments reviewing ALARA performance (July and August 2001)

- Selected corrective action documentation related to the ALARA program since the last inspection in this area (2001-0312, 2001-0333, 2001-0551, 2001-0842, 2001-0881, and 2001-0919)

b. Findings

The inspector identified a noncited violation of very low safety significance (Green) because the licensee's work control process did not ensure that all work activities were reviewed to identify opportunities to reduce radiation doses.

On September 25, 2001, the inspector observed a conversation between a group of electricians and the ALARA coordinator. The electricians were attempting to determine the appropriate radiation work permit to use while installing temporary electrical power in various areas of the plant. The ALARA coordinator asked the lead electrician which MAI was used for installation. (An MAI was the document used to identify work activities and to authorize work.) The lead electrician stated that there was no MAI associated with the temporary electrical power installations. The lack of an associated MAI meant that the work activity had not been entered into the licensee's work control process, governed by Procedure WM-100, "Maintenance Action Item Generation, Screening, and Classification," Revision 0. The installation of temporary electrical power had not been reviewed by job planners with radiation protection expertise to determine the best means to keep the resulting radiation doses ALARA. Work control representatives acknowledged that their existing work control process, implemented by Procedure WM-100, did not address activities such as temporary electrical power installations.

The inspector interviewed the electricians' supervisor and determined that there were 94 temporary electrical power installations to be completed during the outage. Three of the installations had already been completed using a general maintenance radiation work permit. Radiation protection personnel stated that they were unaware that the general radiation work permit was used and would not have permitted its use had they been consulted because of the high number of temporary electrical power installations. The inspector reviewed dose records and found that two crews of electricians had already installed temporary electrical power in the drywell, a locked high radiation area, and accrued 285 millirems.

The failure to review work to implement dose saving measures had a credible impact on safety. The occurrences involved workers' unplanned, unintended doses that resulted from actions that were contrary to licensee procedures and Technical Specifications. However, the safety significance was determined to be very low because there was no exposure in excess of regulatory limits or significant potential for exposure in excess of regulatory limits.

The inspector identified the failure to have an implementing procedure or process for reviewing temporary electrical power installations and taking into account factors for minimizing radiation exposure to workmen as a violation of Technical Specification 5.4.1. Technical Specification 5.4.1 requires written procedures be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A, Section 9.e., includes procedures for the control of maintenance which take into account factors necessary for minimizing radiation exposure to workmen. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. The licensee placed this item in its corrective action program as CR-RBS-2001-1149 (NCV 50-458/200103-02).

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors verified the accuracy and completeness of the data used to calculate and report performance indicator data for the last quarter of 2000 and the first quarter of 2001. The inspectors used Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0, as guidance and interviewed licensee personnel responsible for compiling the information. The following performance indicators were reviewed:

- Unplanned Scrams per 7000 Critical Hours
- Scrams with Loss of Normal Heat Removal
- Unplanned Power Changes per 7000 Critical Hours

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meetings

The inspectors presented the inspection results to Mr. Randy Edington, Vice-President, Operations, and other members of licensee management at the conclusion of various parts of the inspection on June 28, September 28, and October 4, 2001.

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

ATTACHMENT

SUPPLEMENTARY INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Assarello, Manager, Training and Development
M. Bakarich, Manager, Emergency Preparedness
R. Biggs, Coordinator, Nuclear Safety and Regulatory Affairs
W. Brian, Director, Engineering
E. Bush, Superintendent, Operations
M. Cantrell, Supervisor, Operations Training
R. Edington, Vice President-Operations
J. Fowler, Manager, Quality Assurance
R. Frayer, Supervisor, System Engineering
H. Goodman, Superintendent, Reactor Engineering
T. Hildebrandt, Manager, Maintenance
H. Holmes, ALARA Coordinator
J. Holmes, Manager, Technical Support
R. King, Director, Nuclear Safety Assurance
J. Leavines, Manager, Nuclear Safety and Regulatory Affairs
F. Lenox, Technical Specialist IV, Maintenance Rule Coordinator
W. Mashburn, Manager, Engineering Programs
J. McGhee, Manager, Operations
D. Mims, General Manager
A. Shahkarami, Manager, System Engineering
W. Trudell, Manager, Corrective Action and Assessment
M. Wasner, Supervisor, Operations Training
D. Wells, Superintendent, Radiation Protection
M. Wyatt, Manager, Planning and Scheduling/Outage

ITEMS OPENED AND CLOSED

Opened and Closed

50-458/2001-03-01	NCV	Failure to maintain fire barrier requirements described in the plant fire hazards analysis
50-458/2001-03-02	NCV	Failure to review work and identify dose saving measures

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Lesson Plan RBS-1-LEC-LOR-00501.02
Lesson Plan RBS-1-LEC-LOR-00502.02
Lesson Plan RBS-1-LEC-LOR-00504.00
Lesson Plan RBS-1-LP-LOR-00505.00
Lesson Plan RBS-1-SIM-STG-40008.00
Lesson Plan RBS-1-SIM-STG-49811.02

JPM-200-05, Revision 4
JPM-254-01, Revision 2
JPM-05303.05
JPM-204-04, Revision 1
JPM-403-01, Revision 3
JPM-203-03, Revision 1
JPM-05201.04
JPM-309-01, Revision 5
JPM-80033a.00
JPM-800-14, Revision 6
JPM-109-03, Revision 1
JPM-800-34, Revision 0
JPM-800-08, Revision 4
JPM-700-05, Revision 3
JPM-80013.04
JPM-200-01, Revision 2
JPM-800-05, Revision 5
JPM-800-07, Revision 7
JPM-800-19, Revision 3
JPM-30904.00

Scenario RBS-1-SIM-SMS-0804.02
Scenario RBS-1-SIM-SMS-0805.01
Scenario RBS-1-SIM-SMS-0813.01

Two Biennial SRO Written Exams (2000)
Three Biennial RO Written Exams (2000)

LIST OF ACRONYMS AND INITIALISMS USED

ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
CR	condition report
CRFA	control room fresh air
EOOS	risk evaluation tool
ERO	Emergency Response Organization
IFTS	inclined fuel transfer system
LOCA	loss of coolant accident
LPCS	low pressure core spray
MAI	maintenance action item
MRE	meals ready to eat
NCV	noncited violation
NRC	U. S. Nuclear Regulatory Commission
PMT	postmaintenance testing
PSI	plant safety index
RCIC	reactor core isolation cooling
SDP	significance determination process
SSC	structure, system, or component
SSW	standby service water
USAR	Updated Safety Analysis Report