UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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September 1, 2003

Southern Nuclear Operating Company, Inc.
ATTN: Mr. H. L. Sumner, Jr.
Vice President
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR POWER PLANT - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000321/2003006 AND 05000366/2003006

Dear Mr. Sumner:

On July 25, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Hatch Nuclear Plant Units 1 and 2. The enclosed inspection report documents the inspection findings, which were discussed on that date with Mr. R. Dedrickson and other members of your staff. Following completion of additional review in the Region II office, a final exit was held by telephone with Mr. S. Tipps and other members of your staff on September 2, 2003.

The 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two findings that have potential safety significance greater than very low significance, however, a safety significance determination has not been completed. One issue involving a procedural inadequacy did present an immediate safety concern, however, your staff revised the procedure prior to the end of the inspection. The other issue did not present an immediate safety concern. In addition, the report documents three NRC-identified findings of very low safety significance (Green), all of which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these three findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Hatch Nuclear Power Plant.

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

Sincerely,

\RA

Charles R. Ogle, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 50-321, 50-366
License Nos.: DPR-57, NPF-5

Enclosure: NRC Triennial Fire Protection Inspection Report 05000321/2003006 and 05000366/2003006 w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-321, 50-366
License Nos.:	DPR-57, NPF-5
Report No.:	05000321/2003006 and 05000366/2003006
Licensee:	Southern Nuclear Operating Company
Facility:	E. I. Hatch Nuclear Plant
Location:	P. O. Box 2010 Baxley, GA. 31513
Dates:	July 7-11, 2003 (Week 1) July 21-25, 2003 (Week 2)
Inspectors:	C. Smith, P. E., Senior Reactor Inspector, (Lead Inspector) R. Schin, Senior Reactor Inspector G. Wiseman, Fire Protection Inspector K. Sullivan, Consultant, Brookhaven National Laboratory
Accompanying Personnel:	S. Belcher, Nuclear Safety Intern, Week 1
Approved by:	Charles R. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000321/2003-006, 05000366/2003-006; 7/7-11/2003 and 7/21-25/2003; E. I. Hatch Nuclear Plant, Units 1 and 2; Triennial Fire Protection

The report covered an announced two-week period of inspection by three regional inspectors and a consultant from Brookhaven National Laboratory. Three Green non-cited violations (NCVs) and two unresolved items with potential safety significance greater than Green were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

- TBD. The team identified an unresolved item in that a local manual operator action, to prevent spurious opening of all eleven safety relief valves (SRVs) during a fire event, would not be performed in sufficient time to be effective. Also, licensee reliance on this manual action for hot shutdown during a fire, instead of physically protecting cables from fire damage, had not been approved by the NRC.

  This finding is unresolved pending completion of a significance determination. The finding is greater than minor because it affects the objective of the mitigating system cornerstone. Also, the finding has potential safety significance greater than very low safety significance because failure to prevent spurious operation of the SRVs could result in them opening during certain fire scenarios, thereby complicating the post-fire recovery actions. (Section 1R05.04/.05.b.1)

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix R, Section III.G.1 and Technical Specification 5.4.1 because a local manual operator action to operate safe shutdown equipment was too difficult and was also physically unsafe. The licensee had relied on this action instead of providing physical protection of cables from fire damage or preplanning cold shutdown repairs. However, the team determined that some operators would not be able to perform the action.

  The finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating systems cornerstone. This finding is of very low safety significance because the licensee would have time to develop and implement cold shutdown repairs to facilitate accomplishment of the action. (Section 1R05.04/.05.b.2)

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix R, Section III.G.2 in that the licensee relied on some manual operator actions to operate safe shutdown equipment, instead of providing the required physical protection of cables from fire damage without NRC approval.

  The finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating systems cornerstone. Since the actions could reasonably be accomplished by operators in a timely manner, this finding did not have potential safety significance greater than very low safety significance. (Section 1R05.04/.05.b.3)

- Green. The team identified a non-cited violation 10 CFR 50, Appendix R, Section III.J because emergency lighting was not adequate for some manual operator actions that were needed to support post-fire operation of safe shutdown equipment.

  The finding is greater than minor because it affected the reliability objective and the equipment performance attribute of the mitigating systems cornerstone. Since operators would be able to accomplish the actions with the use of flashlights, this finding did not have potential safety significance greater than very low safety significance. (Section 1R05.07.b)

- TBD: The team identified a violation of 10 CFR 50, Appendix B in connection with the implementation of Design Change Request 91-134, SRV Backup Actuation via Pressure Transmitter Signals. The installed plant modification failed to implement the "one-out-of-two taken twice" logic that was specified as a design input requirement in the design change package. Additionally, implementation of a "two-out-of-two coincidence taken twice" logic has introduced a potential common cause failure of all eleven SRVs as a result of the potential for fire-induced damage to two reactor pressure instrumentation circuit cables in close proximity to each other.

  This finding is unresolved pending completion of a significance determination. This finding is greater than minor because it impacts the mitigating system cornerstone. This finding has the potential for defeating manual control of Group A SRVs that are required for ensuring that the suppression pool temperature will not exceed the heat capacity temperature limit for the suppression pool and therefore has a potential safety significance greater than very low safety significance. (Section 1R21.01.b)

B. Licensee-Identified Violations

None

REPORT DETAILS

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

1R05 Fire Protection

The purpose of this inspection was to review the Hatch Nuclear Plant fire protection program (FPP) for selected risk-significant fire areas. Emphasis was placed on verification that the post-fire safe shutdown (SSD) capability and the fire protection features provided for ensuring that at least one redundant train of safe shutdown systems is maintained free of fire damage. The inspection was performed in accordance with the Nuclear Regulatory Commission (NRC) Reactor Oversight Program using a risk-informed approach for selecting the fire areas and attributes to be inspected. The team used the licensee's Individual Plant Examination for External Events and in-plant tours to choose four risk-significant fire areas for detailed inspection and review. The fire areas chosen for review during this inspection were:

- Fire Area 2016, West 600 V Switchgear Room, Control Building, Elevation 130 feet.
- Fire Area 2104, East Cableway, Turbine Building, Elevation 130 feet.
- Fire Area 2404, Switchgear Room 2E, Diesel Generator Building, Elevation 130 feet.
- Fire Area 2408, Switchgear Room 2F, Diesel Generator Building, Elevation 130 feet.

The team evaluated the licensee's FPP against applicable requirements, including Operating License Condition 2.C.(3)(a), Fire Protection; Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), Appendix R; 10 CFR 50.48; Appendix A of Branch Technical Position (BTP) Auxiliary and Power Conversion Systems Branch (APCSB) 9.5-1; related NRC Safety Evaluation Reports (SERs); the Hatch Nuclear Plant Updated Final Safety Analysis Report (UFSAR); and plant Technical Specification (TS). The team evaluated all areas of this inspection, as documented below, against these requirements.

Documents reviewed by the team are listed in the attachment.

.01 Systems Required to Achieve and Maintain Post-Fire Safe Shutdown

a. Inspection Scope

The licensee's Safe Shutdown Analysis Report (SSAR) was reviewed to determine the components and systems necessary to achieve and maintain SSD conditions in the event of fire in each of the selected fire areas. The objectives of this evaluation were as follows:

- Verify that the licensee's shutdown methodology has correctly identified the components and systems necessary to achieve and maintain a SSD condition.
- Confirm the adequacy of the systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring and support system functions.
- Verify that a SSD can be achieved and maintained without off-site power, when it can be confirmed that a postulated fire in any of the selected fire areas could cause the loss of off-site power.
- Verify that local manual operator actions are consistent with the plant's fire protection licensing basis.

b. Findings

The team identified a potential concern in that the licensee used manual actions to disconnect terminal board sliding links in order to isolate two 4 to 20 milli-amp (ma) instrumentation loop control circuits in order to prevent the spurious actuation of eleven safety relief valves (SRVs). This issue is discussed in Section 1R05.03.b of the report. No other findings of significance were identified.

.02 Fire Protection of Safe Shutdown Capability

a. Inspection Scope

For the selected fire areas, the team evaluated the frequency of fires or the potential for fires, the combustible fire load characteristics and potential fire severity, the separation of systems necessary to achieve SSD, and the separation of electrical components and circuits located within the same fire area to ensure that at least one SSD path was free of fire damage. The team also inspected the fire protection features to confirm they were installed in accordance with the codes of record to satisfy the applicable separation and design requirements of 10 CFR 50, Appendix R, Section III.G, and Appendix A of BTP APCSB 9.5-1. The team reviewed the following documents, which established the controls and practices to prevent fires and to control combustible fire loads and ignition sources, to verify that the objectives established by the NRC-approved FPP were satisfied:

- UFSAR Section 9.1-A, Fire Protection Plan
- Administrative Procedure 40AC-ENG-008-0S, Fire Protection Program
- Administrative Procedure 42FP-FPX-018-0S, Use, Control, and Storage of Flammable/Combustible Materials
- Preventive Maintenance Procedure 52PM-MEL-012-0, Low Voltage Switchgear Preventive Maintenance

The team toured the selected plant fire areas to observe whether the licensee had properly evaluated in-situ fire loads and limited transient fire hazards in a manner consistent with the fire prevention and combustible hazards control procedures. In addition, the team reviewed the licensee's fire safety inspection reports and corrective action program (CAP) condition reports (CRs) resulting from fire, smoke, sparks, arcing, and overheating incidents for the years 2000-2002 to assess the effectiveness of the fire prevention program and to identify any maintenance or material condition problems related to fire incidents.

The team reviewed fire brigade response, fire brigade qualification training, and drill program procedures; fire brigade drill critiques; and drill records for the operating shifts from January 1999 - December 2002. The reviews were performed to determine whether fire brigade drills had been conducted in high fire risk plant areas and whether fire brigade personnel qualifications, drill response, and performance met the requirements of the licensee's approved FPP.

The team walked down the fire brigade equipment storage areas and dress-out locker areas in the fire equipment building and the turbine building to assess the condition of fire fighting and smoke control equipment. Fire brigade personal protective equipment located at both of the fire brigade dress-out areas and fire fighting equipment storage area in the turbine building were reviewed to evaluate equipment accessibility and functionality. Additionally, the team observed whether emergency exit lighting was provided for personnel evacuation pathways to the outside exits as identified in the National Fire Protection Association (NFPA) 101, Life Safety Code, and the Occupational Safety and Health Administration (OSHA) Part 1910, Occupational Safety and Health Standards. This review also included examination of whether backup emergency lighting was provided for access pathways to and within the fire brigade equipment storage areas and dress-out locker areas in support of fire brigade operations should power fail during a fire emergency. The fire brigade self-contained breathing apparatuses (SCBAs) were reviewed for adequacy as well as the availability of supplemental breathing air tanks and their refill capability.

The team reviewed fire fighting pre-fire plans for the selected areas to determine if appropriate information was provided to fire brigade members and plant operators to facilitate suppression of a fire that could impact SSD. Team members also walked down the selected fire areas to compare the associated pre-fire plans and drawings with as-built plant conditions. This was done to verify that fire fighting pre-fire plans and drawings were consistent with the fire protection features and potential fire conditions described in the Fire Hazards Analysis (FHA).

The team reviewed the adequacy of the design, installation, and operation of the manual suppression standpipe and fire hose system for the control building. This was accomplished by reviewing the FHA, pre-fire plans and drawings, engineering mechanical equipment drawings, design flow and pressure calculations, and NFPA 14 for hose station location, water flow requirements and effective reach capability. Team members also walked down the selected fire areas in the control building to ensure that hose stations were not blocked and to verify that the required fire hose lengths to reach the safe shutdown equipment in each of the selected areas were available. Additionally, the team observed placement of the fire hoses and extinguishers to assess consistency with the fire fighting pre-fire plans and drawings.

b. Findings

No findings of significance were identified.

.03 Post-Fire Safe Shutdown Capability

a. Inspection Scope

On a sample basis, the inspectors evaluated whether the systems and equipment identified in the licensee's SSAR as being required to achieve and maintain hot shutdown conditions would remain free of fire damage in the event of fire in the selected fire areas. The evaluation included a review of cable routing data depicting the location of power and control cables associated with SSD Path 1 and Path 2 components of the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems. Additionally, on a sample basis, the team reviewed the licensee's analysis of electrical protective device (e.g., circuit breaker, fuse, relay) coordination. The following motor operated valves (MOVs) and other components were reviewed:

Component ID	Description
2E51-F029	RCIC Pump Suction from Suppression Pool Valve
2E51-F010	RCIC Pump Suction Valve from Condensate Storage Tank (CST)
2P41-C001A	Plant Service Water Pump 2A
2E11-F011A	Residual Heat Removal (RHR) Heat Exchanger A Drain to Suppression Pool Valve
2P41-C001B	Plant Service Water Pump 2B
2E41-F001	HPCI Turbine Steam Supply Valve
2E41-F002	HPCI Turbine Steam Supply Inboard Containment Isolation Valve
2E41-F006	HPCI Pump Inboard Discharge Valve
2E41-F008	HPCI Pump Discharge Bypass Test Valve to CST

b. Findings

The team identified a potential concern in that the licensee used manual actions to isolate two 4 to 20 ma instrumentation loop control circuits associated with eleven SRVs in lieu of providing physical protection. This did not appear to be consistent with the plant's licensing basis nor 10 CFR 50, Appendix R. Spurious action of these SRVs could impact the licensee's fire mitigation strategy. In addition, the licensee provided no objective evidence that post-fire safe shutdown equipment could mitigate this event.

The SSAR stated that a fire in Fire Area 2104 could cause all eleven SRVs to spuriously actuate as a result of fire damage to two cables located in close proximity in this area. The specific circuits that could cause this event were identified by the licensee as circuits ABE019C08 and ABE019C09. Each circuit separately provides a 4 to 20 ma instrumentation signal from an SRV high-pressure actuation transmitter 2B21-N127B or 2B21-N127D to its respective master trip unit (2B21-N697B or 2B21-N697D). The purpose of this circuitry was to provide an electrical backup to the mechanical trip capability of the individual SRVs. In the event of high reactor pressure, the circuits would provide a signal to the master trip units which would cause all eleven SRVs to actuate (open). The pressure signal from each transmitter would be conveyed to its respective master trip unit through a two-conductor, instrument cable that was routed through this fire area (two separate cables). Each cable consisted of a single twisted pair of insulated conductors, an uninsulated drain wire that was wound around the twisted pair of conductors, and a foil shield. In Fire Area 2104, the two cables were located in close proximity in the same cable tray. Actuation of the SRV electrical backup is completely "blind" to the operators. That is, unlike ADS, it does not provide any pre-actuation indication (e.g., actuation of the ADS timer) or an inhibit capability (e.g., ADS inhibit switch). Because the operators typically would not initiate a manual scram until fire damage significantly interfered with control of the plant, it is possible that all eleven SRVs could open at 100% power, prior to scramming the reactor. This event could place the plant in an unanalyzed condition.

Unlike a typical control circuit, a direct short or "hot short" between conductors of a 4 to 20 ma instrument circuit may not be necessary to initiate an undesired (false high) signal. For cables that transmit low-level instrument signals, degradation of the insulation of the individual twisted conductors due to fire damage may be sufficient to cause leakage current to be generated between the two conductors. Such leakage current would appear as a false high pressure signal to the master trip units. If both cables were damaged as a result of fire, false signals generated as a result of leakage current in each cable, could actuate the SRV electrical backup scheme which would cause all eleven SRVs to open. The conductor insulation and jacket material of each cable was cross-linked polyethylene (XLPE). Because both cables were in the same tray and exposed to the same heating rate, there would be a reasonable likelihood that both instrumentation cables could suffer insulation damage at the same time and both circuits could fail high simultaneously.

The licensee's SSAR recognized the potential safety significance of this event and described methods that have been developed to prevent its occurrence and/or to mitigate its impact on the plant's post-fire SSD capability (should it occur). To prevent this event, the licensee developed procedural guidance which directs operators to open link BB-10 in panel 2H11-P927 and link BB-10 in panel 2H11-P928. These panels are located in the main control room. Opening of these links would prevent actuation of the SRV trip units by removing the 4 to 20 ma signal fed by the pressure transmitters (PT) to the master trip units. In the event the SRVs were to open prior to the operators completing this action, the SSAR credits core spray loop A to mitigate the event.

The inspection team had several concerns regarding the licensee's approach to this potential spurious actuation of the SRVs. Specific concerns identified by the team include:

1. The links may not be opened in time to preclude inadvertent actuation of the SRVs.

2. The use of links to avoid inadvertent actuation of the SRVs did not appear to be consistent with the current licensing basis.
3. No objective evidence existed to demonstrate that the post-fire SSD equipment could adequately mitigate a fire in Fire Area 2104, if the SRVs were to open.
4. The operations staff would be unable to manually control the Group A SRVs, which are credited for mitigating a fire in Fire Area 2104, should they spuriously actuate as a result of fire-induced damage.

With regard to the timing of operator actions to prevent fire damage from causing all SRVs to open, the licensee performed an evaluation during the inspection which estimated that approximately thirty minutes would pass from the time of fire detection to the time an operator would implement procedural actions to open the links. The inspectors independently arrived at a similar time estimate based on their review of the procedure. In response to inspector's concerns that this interval may be too lengthy to preclude fire damage to the cables of interest and subsequent actuation of the SRVs, the licensee agreed to enhance its existing procedures so that the action would be taken immediately following confirmation of fire in areas where the spurious actuation could occur. This issue is discussed in Section 1R05.04/.05.b.1 of this report.

The team also determined that the opening of terminal board links was not in compliance with the plant's licensing basis. Current licensing basis documents, specifically Georgia Power request for exemption dated May 16, 1986, and a subsequent NRC Safety Evaluation Report (SER) dated January 2, 1987, characterized the opening of links as a repair activity that is not permitted as a means of complying with 10 CFR 50, Appendix R, Section III.G. The inspectors concluded that, the opening of links was considered a repair by both the licensee and the NRC staff in 1987. The licensee could not provide any evidence to justify why these actions should not be characterized as a repair activity in its current SSAR.

Additionally, because there is a potential for all SRVs to spuriously actuate as a result of fire in Fire Area 2104 at a time when RHR is not available, the SSAR credits the use of core spray loop A to accomplish the reactor coolant makeup function. During the inspection, the licensee performed a simulator exercise of an event which caused all 11 SRVs to open. During this exercise, simulator RPV level instruments indicated that core spray would be capable of maintaining level above the top of active fuel. However, the licensee did not provide any objective evidence (e.g., specific calculation or analysis) which demonstrated that, assuming worst-case fire damage in Fire Area 2104, the limited set of equipment available would be capable of mitigating the event in a manner that satisfied the shutdown performance goals specified in 10 CFR 50, Appendix R, Section III.L.1.e.

Finally, the logic that was installed by design change request (DCR) 91-134 for the SRVs was a "two-out-of-two coincidence taken twice" logic in addition to a "one-out-of-two coincidence taken twice" logic. The team determined that the "two-out-of-two" coincidence logic input from trip unit master relays K310D and K335D represented a common cause failure for Group A SRVs for a fire in Fire Area 2104. Specifically, cable ABE019C08 associated with PT 2B21-N127B current loop, and cable ABE019C09 associated with PT 2B21-N127D current loop, were routed in close proximity to each other in the same cable tray in Fire Area 2104. Both shielded twisted pair instrument cables were unprotected from the effects of a fire in this fire area. Fire-induced insulation damage to both cables could result in leakage currents and cause the instrument loops to fail high. This failure mode would simulate a high nuclear boiler pressure condition and would initiate SRV backup actuation of all the Group A SRVs. Whenever a SRV lifted, it would remain open until pressure reduced to about 85% of its overpressure lift setpoint However, the instrument loops, having failed high, would ensure that the trip unit master relays and the trip unit slave relays continued to energize the pilot valve of the individual SRV and keep the SRV open. This issue is discussed in more detail in Section 1R21.01. Ultimately, this failure mode would prevent the operators from manually controlling the Group A SRVs as required per the SSAR.

In response, the licensee initiated CR 2003800152, dated July 24, 2003, to evaluate actions to open links to determine if they are necessary to achieve hot shutdown, and if an exemption from Appendix R is required. Pending additional review by the NRC, this issue is identified as Unresolved Item (URI) 50-366/03-06-01, Concerns Associated with Potential Opening of SRVs.

.04/.05 Alternative Shutdown Capability/Operational Implementation of Alternative Shutdown Capability

a. Inspection Scope

The selected fire areas that were the focus of this inspection all involved reactor shutdown from the control room. None involved abandoning the control room and alternative SSD from outside of the control room. Thus, alternative shutdown capability was not reviewed during this inspection. However, the licensee's plans for SSD following a fire in the selected areas involved many local manual operator actions that would be performed outside of the control area of the control room. This section of the inspection focused on those local manual operator actions.

The team reviewed the operational implementation of the SSD capability for a fire in the selected fire areas to determine if: (1) the procedures were consistent with the SSAR; (2) the procedures were written so that the operator actions could be correctly performed within the times that were necessary for the actions to be effective; (3) the training program for operators included SSD capability; (4) personnel required to achieve and maintain the plant in hot standby could be provided from the normal onsite staff, exclusive of the fire brigade; and (5) the licensee periodically performed operability testing of the SSD equipment.

The team walked down SSD manual operator actions that were to be performed outside of the control area of the main control room for a fire in the selected fire areas and discussed them with operators. These actions were documented in Abnormal Operating Procedure (AOP) 34AB-X43-001-2, Version 10.8, dated May 28, 2003. The team evaluated whether the local manual operator actions could reasonably be performed, using the criteria outlined in NRC Inspection Procedure (IP) 71111.05, Enclosure 2. The team also reviewed applicable operator training lesson plans and job performance measures (JPMs) and discussed them with operators. In addition, the team reviewed records of actual operator staffing on selected days.

b. <u>Findings</u>

1. <u>Untimely and Unapproved Manual Operator Action for Fire SSD</u>

<u>Introduction</u>: The team found that a local manual operator action to prevent spurious opening of all eleven SRVs would not be performed in sufficient time to be effective. Licensee reliance on this manual action for hot shutdown during a fire, instead of physically protecting cables from fire damage, had not been approved by the NRC.

<u>Description</u>: The team noted that Step 9.3.2.1 of AOP 34AB-X43-001-2, Fire Procedure, Version 10.8, dated May 28, 2003, stated: "To prevent all eleven SRVs from opening simultaneously, open links BB-10 in Panel 2H11-P927 and BB-10 in Panel 2H11-P928." The team noted that spurious opening of all eleven SRVs should be considered a large loss of coolant accident (LOCA), and that a LOCA should be prevented from occurring during a fire event to comply with 10 CFR 50, Appendix R, Section III.L. Section III.L requires that, during a post-fire shutdown, the reactor coolant system process variables (e.g., reactor vessel pressure and water level) shall be maintained within those predicted for a loss of normal alternating current power. Having all eleven SRVs opened during a fire would challenge this. Additionally, the team observed that this step was sufficiently far back in the procedure that it may not be completed in time to prevent potential fire damage to cables from causing all eleven SRVs to spuriously open.

The licensee had no preplanned estimate of how long it would take operators to complete this step during a fire event. There was no event time line or operator training JPM on this step. The team noted that, during a fire, operators could be using many other procedures concurrent with the Fire Procedure. For example, they could be using other procedures to communicate with the fire brigade about the fire, respond to a reactor trip, deal with a loss of offsite power, and provide emergency classifications and offsite notifications of the fire event. During the inspection, licensee operators estimated that, during a fire event, it could take about 30 minutes before operators would accomplish Step 9.3.2.1. The team concurred with that time estimate which the team had previously determined independently. However, NRC fire models indicated that fires could potentially cause damage to cables in as short a period as five to ten minutes. Consequently, the team concluded that during a fire event, the licensee's procedures would not ensure that Step 9.3.2.1 would be accomplished in time to prevent potential spurious opening of all eleven SRVs.

The team also identified other issues with Step 9.3.2.1. There was no emergency lighting inside the panels, hence, if the fire caused a loss of normal lighting (e.g., by causing a loss of offsite power), operators would need to use flashlights to perform the actions inside the panels. Consequently, the team considered the emergency lighting for Step 9.3.2.1 to be inadequate (see Section 1R05.07.b). In addition, labeling of the links inside the panels was so poor that operators stated that they would not fully rely on the labeling. Also, the tool that operators would use to loosen and slide the links inside the energized panels was made of steel and was not professionally, electrically insulated. Further, licensee reliance on this operator action, instead of physically protecting the cables as required by 10 CFR 50, Appendix R, Section III.G.2, had not been approved by the NRC.

The licensee stated that cable damage to two reactor pressure instrument cables would be needed to spuriously open all eleven SRVs. Because the licensee stated that the two cables were in the same cable tray in Fire Area 2104, the team considered that a fire in that area could potentially cause all eleven SRVs to spuriously open (see Section 1R21.01.b).

In response to this issue, the licensee initiated CR 2003008203 and promptly revised the Fire Procedure before the end of the inspection, moving the actions of Step 9.3.2.1 to the beginning of the procedure. The procedure change enabled the actions to be accomplished much sooner during a fire in the Unit 2 east cableway or in other fire areas that were vulnerable to the potential for spuriously opening all eleven SRVs. The team determined that this issue is related to associated circuits. As described in NRC IP 71111.05, Fire Protection, inspection of associated circuits is temporarily limited. Consequently, the team did not pursue the cable routing or circuit analysis that would be necessary to evaluate the possibility, risk, or potential safety significance of Group B and C SRVs spuriously opening due to fire damage to the instrument cables. The team did, however, perform a circuit analysis of Group A SRVs for which the licensee takes credit during a fire in Fire Area 2104 (see Section 1R21.01.b)

Analysis: The team determined that this finding was associated with the protection against external factors attribute. It affected the objective of the mitigating system cornerstone to ensure the availability of systems that respond to initiating events and is therefore greater than minor. The team determined that the finding had potential safety significance greater than very low safety significance because failure to prevent spurious operation of the SRVs could result in them opening in certain fire scenarios, thereby complicating the post-fire recovery actions. However, the finding remains unresolved pending completion of the SDP.

Enforcement: 10 CFR 50, Appendix R, Section III.G.2, requires that where cables or equipment, including associated non-safety circuits that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of the primary containment, one of the following means of ensuring that one or the redundant trains is free of fire damage shall be provided: 1) a fire barrier with a 3-hour rating; 2) separation of cables by a horizontal distance of more than 20 feet with no intervening combustibles and with fire detectors and automatic fire suppression; or 3) a fire barrier with a 1-hour rating with fire detectors and automatic suppression.

The licensee had not provided physical protection against fire damage for the two instrument cables by one of the prescribed methods. Instead, the licensee had relied on local manual operator actions to prevent the spurious opening of all eleven SRVs. Licensee personnel stated that fire damage to two cables was outside of the Hatch licensing basis and, consequently, there was no requirement to protect the instrument cables. However, the licensee could not provide evidence to support that position.

This potential issue will remain unresolved pending the completion of a significance determination by the NRC. This issue is identified as URI 50-366/03-06-02, Untimely and Unapproved Manual Operator Action for Post-Fire SSD.

2. Local Manual Operator Action was Too Difficult and Physically Unsafe

Introduction: A finding of very low safety significance was identified in that a local manual operator action to operate SSD equipment was too difficult and was also physically unsafe. The team judged that some operators would not be able to perform the action. This finding involved a violation of NRC requirements.

Description: The team observed that Steps 4.15.8.1.1 and 9.3.5.1 of the Fire Procedure relied upon local manual operator actions instead of providing physical protection for cables or providing a procedure for cold shutdown repairs. Both steps required the same local manual operator action: "Manually OPEN 2E11-F015A, Inboard LPCI Injection Valve, as required." This action was to be taken in the Unit 2 drywell access, which was a locked high radiation, contaminated, and hot area with temperatures over 100 degrees F.

Valve 2E11-F015A was a large (24-inch diameter) motor-operated gate valve with a three-foot diameter handwheel. The main difficulty with manually opening this valve was lack of an adequate place to stand. An operator showed the team that to perform the action he would have to climb up to, and stand on a small section of pipe lagging (a curved area about four inches wide by 12 inches long), and then reach back and to his right side, to hold the handwheel with his right hand, while reaching forward and to his right to hold the clutch lever for the motor operator with his left hand. The operator would not have good balance while performing the action. The foothold, which was large enough to support only one foot, was well flattened and appeared to have been used in the past to manually operate this valve. The foothold was about six to seven feet above a steel grating, and the team observed that the space available for potential use of a ladder to better access the 2E11-F015A valve handwheel was not good.

Other difficulties with manually opening the valve included the heat; the need to wear full anti-contamination clothing, a hardhat, and safety glasses; and inadequate emergency lighting (see Section 1R05.07). Also, there was no note or step in the procedure to ensure that the RHR pumps were not running before attempting to manually open the 2E11-F015A valve. If an RHR pump were running, it could create a differential pressure across the valve which could make manually opening it much more difficult. If the operator did not have sufficient agility, strength or stamina, he would be unable to complete the action. Also, the team judged that inability to remove sweat from his eyes, due to wearing gloves that could be contaminated, would be a limiting factor for the operator. In addition, if the operator slipped or lost his balance, he could fall and become injured. Considering all of the difficulties, the team judged that this action was physically unsafe and that some operators would not be able to perform it.

The licensee had no operator training JPM for performing this action and an operator stated that he had not performed or received training on this action. One experienced operator, who appeared to be in much better physical condition that an average nuclear plant operator, stated that he had manually operated the valve in the past, but that it had been very difficult for him.

The team judged that, since this action was not required to maintain hot shutdown but only required for cold shutdown following a fire in one of the four selected fire areas, licensee personnel could have time to improve the working conditions after a fire. They could have time to install scaffolding or temporary ventilation, improve the lighting, and assign multiple operators to manually open the valve. They could have time to perform a cold shutdown repair. However, the licensee had not preplanned any cold shutdown repairs for opening this valve.

Analysis: This finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating systems cornerstone. Because the licensee would have time to develop and implement cold shutdown repairs to facilitate accomplishment of the action, this finding did not impact the effectiveness of one or more of the defense in depth elements. Hence, this finding did not have potential safety significance greater than very low safety significance (Green).

Enforcement: 10 CFR 50, Appendix R, Section III.G.1, requires that fire protection features shall be provided for systems important to safe shutdown and shall be capable of limiting fire damage so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control stations can be repaired within 72 hours. In addition, TS 5.4.1 requires that written procedures shall be established, implemented, and maintained covering activities including FPP implementation and including the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33 recommends procedures for combating emergencies including plant fires and procedures for operation and shutdown of safety-related boiling water reactor systems. The fire protection program includes the SSAR which requires that valve 2E11-F015A be opened for SSD following a fire in Fire Area 2104, the Unit 2 east cableway. AOP 34AB-X43-001-2, Fire Procedure, Version 10.8, dated May 28, 2003, implements these requirements in that it provides information and actions necessary to mitigate the consequences of fires and to maintain an operable shutdown train following fire damage to specific fire areas. Also, AOP 34AB-X43-001-2 provides Steps 4.15.8.1.1 and 9.3.5.1 for manually opening valve 2E11-F015A following a fire in Fire Area 2104.

Contrary to the above, the licensee had no procedure for repairing any related fire damage within 72 hours. Instead, the licensee relied on local manual operator actions, as described in Steps 4.15.8.1.1 and 9.3.5.1 of AOP 34AB-X43-001-2. However, those procedure steps were inadequate in that some operators would not be able to perform them because the required actions were too difficult and also were physically unsafe. In response to this issue, the licensee initiated CR 203008202. Because the identified inadequate procedure steps are of very low safety significance and the issue has been entered into the licensee's corrective action program, this violation is being treated as an non-cited violation (NCV), consistent with Section VI.A of the NRC's Enforcement Policy: NCV 50-366/03-06-03, Inadequate Procedure for Local Manual Operator Action for Post-Fire Safe Shutdown Equipment.

3. Unapproved Manual Operator Actions for Post-Fire SSD

Introduction: A finding of very low safety significance was identified in that the licensee relied on some local manual operator actions to operate SSD equipment, instead of providing the required physical protection of cables from fire damage. This finding involved a violation of NRC requirements.

Description: The team observed that AOP 34AB-X43-001-2, Fire Procedure, included some local manual operator actions to achieve and maintain hot shutdown that had not been approved by the NRC. Examples of steps from the procedure included:

- Step 4.15.2.2; ...If a loss of offsite power occurs and emergency busses energize ..."Place Station Service battery chargers 2R42-S026 (2R42-S029), 2R42-S027 (2R42-S030) AND 2R42-S028 (2R42-S031) in service per 34SO-R42-001-2."

- Step 4.15.4.5; ...If HPCI fails to automatically trip on high RPV level... "OPEN the following links to energize 2E41-F124, Trip Solenoid Valve, AND to fail 2E41-F3025 HPCI Governor Valve, in the CLOSED position:
    - TT-75 in panel 2H11-P601
    - TT-76 in panel 2H11-P601"

- Step 4.15.4.6; ...If HPCI fails to automatically trip on high RPV level... "OPEN breaker 25 in panel 2R25-S002 to fail 2E41-F3052, HPCI Governor Valve, in the CLOSED position."

The team walked down these actions using the guidance contained in IP 71111.05T and judged that they could reasonably be accomplished by operators in a timely manner. However, the team determined that these operator actions were being used instead of physically protecting cables from fire damage that could cause a loss of station service battery chargers or a HPCI pump runout.

Analysis: The finding is greater than minor because it affected the availability and reliability objectives as well as the equipment performance attribute of the mitigating systems cornerstone. Since the actions could reasonably be accomplished by operators in a timely manner, this finding did not have potential safety significance greater than very low safety significance.

Enforcement: 10 CFR 50, Appendix R, Section III.G.2, requires that where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of the primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided: 1) a fire barrier with a 3-hour rating; 2) separation of cables by a horizontal distance of more than 20 feet with no intervening combustibles and with fire detectors and automatic fire suppression; or 3) a fire barrier with a 1-hour rating with fire detectors and automatic suppression.

Contrary to the above, the licensee had not provided the required physical protection against fire damage for power to the station service battery chargers or for HPCI electrical control cables. Instead, the licensee relied on local manual operator actions, without NRC approval. In response to this issue, the licensee initiated CR 2003800166. Because the issue had very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC's Enforcement Policy: NCV 50-366/03-06-04, Unapproved Manual Operator Actions for Post-Fire Safe Shutdown.

.06 Communications

a. Inspection Scope

The team reviewed the plant communications systems that would be relied upon to support fire brigade and SSD activities. The team walked down portions of the SSD procedures to verify that adequate communications equipment would be available for personnel performing local manual operator actions. In addition, the team reviewed the adequacy of the radio communication system used by the fire brigade to communicate with the main control room.

b. Findings

No findings of significance were identified.

.07 Emergency Lighting

a. Inspection Scope

The team inspected the licensee's emergency lighting systems to verify that 8-hour emergency lighting coverage was provided as required by 10 CFR 50, Appendix R, Section III.J, to support local manual operator actions that were needed for post-fire operation of SSD equipment. During walkdowns of the post-fire SSD operator actions for fires in the selected fire areas, the team checked if emergency lighting units were installed and if lamp heads were aimed to adequately illuminate the SSD equipment, the equipment identification tags, and the access and egress routes thereto, so that operators would be able to perform the actions without needing to use flashlights.

b. Findings

Inadequate Emergency Lighting for Operation of SSD Equipment

Introduction: A finding with very low safety significance was identified in that emergency lighting was not adequate for some manual operator actions that were needed to support post-fire operation of SSD equipment. This finding involved a violation of NRC requirements.

Description: The team observed that emergency lighting was not adequate for some manual operator actions that were needed to support post-fire operation of SSD equipment. Examples included the following operator actions in procedure 34AB-X43-001-2, Fire Procedure, Version 10.8, dated May 28, 2003:

- Step 4.15.2.2; ...if a loss of offsite power occurs and emergency busses energize ..."Place Station Service battery chargers 2R42-S026 (2R42-S029), 2R42-S027 (2R42-S030) AND 2R42-S028 (2R42-S031) in service per 34SO-R42-001-2."

- Step 4.15.4.5; ...If HPCI fails to automatically trip on high RPV level... "OPEN the following links to energize 2E41-F124, Trip Solenoid Valve, AND to fail 2E41-F3025 HPCI Governor Valve, in the CLOSED position:
  - TT-75 in panel 2H11-P601
  - TT-76 in panel 2H11-P601"

- Step 4.15.5; "IF 2R25-S065, Instrument Bus 2B, is DE-ENERGIZED perform the following manual actions to maintain 2C32-R655, Reactor Water Level Instrument, operable:
  - 4.15.5.1; At panel 2H11-P612, OPEN links AAA-11 and AAA-12.
  - 4.15.5.2; At panel 2H11-P601, CLOSE links HH-48 and HH-49."

- Steps 4.15.8.1.1 and 9.3.5.1; "Manually OPEN 2E11-F015A, Inboard LPCI Injection Valve, as required."

- Steps 4.15.8.1.2 and 9.3.5.2; "Manually CLOSE 2E11-F018A, RHR Pump A Minimum Flow Isolation Valve, as required."

- Step 9.3.2.1; "To prevent all 11 SRVs from opening simultaneously, open links BB-10 in Panel 2H11-P927 and BB-10 in Panel 2H11-P928."

- Step 9.3.3; "At Panel 2H11-P627, open links AA-19, AA-20, AA-21, and AA-22, to prevent spurious actuation of SRVs 2B21-F013D AND 2B21-F013G."

- Step 9.3.6; "OPEN link TB9-21 in Panel 2H11-P700 to open Drywell Pneumatic System Inboard Inlet Isolation, 2P70-F005."

- Step 9.3.7; "OPEN link TB1-12 in Panel 2H11-P700 to open Drywell Pneumatic System Outboard Inlet Isolation, 2P70-F005."

- Step 9.3.9.1; "Confirm OR manually CLOSE RHR Shutdown Cooling Valve 2E11-F006D."

- Step 9.3.9.2; "Manually OPEN Shutdown Cooling Suction Valve 2E11-F008, IF required..."

The team verified that flashlights were readily available and judged that operators would be able to use the flashlights and accomplish the actions, with two exceptions. One exception was the action to open terminal board links in two panels to prevent all eleven SRVs from spuriously opening, which was judged to be untimely (see Section

1R05.04/.05.b.1). The other exception was the action to open 2E11-F015A, which was judged to be too difficult (see Section 1R05.04/.05.b.2). For both of these actions, the lack of adequate emergency lighting could make the actions more difficult to complete in a timely manner and increase the chance of operator error.

Analysis: This finding is greater than minor because it affected the reliability objective and the equipment performance attribute of the mitigating systems cornerstone. Since operators would be able to accomplish the actions with the use of flashlights, this finding did not impact the effectiveness of one or more of the defense in depth elements. Hence, this finding did not have potential safety significance greater than very low safety significance (Green).

Enforcement: 10 CFR 50, Appendix R, Section III.J, requires that emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment, and in access and egress routes thereto.

Contrary to the above, emergency lighting units were not adequately provided in all areas needed for operation of SSD equipment. In response this issue, the licensee initiated CRs 2003008237 and 2003008179. Because the identified lack of emergency lighting is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC's Enforcement Policy: NCV 50-366/03-06-05, Inadequate Emergency Lighting for Operation of Post-Fire Safe Shutdown Equipment.

.08 Cold Shutdown Repairs

The licensee had identified no needed cold shutdown repairs. Also, with the exception of the potential need for a cold shutdown repair to open valve 2E11-F015A (see Section 1R05.05.b.2), the team identified no other need for cold shutdown repairs. Consequently, this section of IP 71111.05 was not performed.

.09 Fire Barriers and Fire Area/Zone/Room Penetration Seals

a. Inspection Scope

The team reviewed the selected fire areas to evaluate the adequacy of the fire resistance of fire area barrier enclosure walls, ceilings, floors, fire barrier mechanical and electrical penetration seals, fire doors, and fire dampers. The team selected several fire barrier features for detailed evaluation and inspection to verify proper installation and qualification. This was accomplished by observing the material condition and configuration of the installed fire barrier features, as well as construction details and supporting fire endurance tests for the installed fire barrier features, to verify the as-built configurations were qualified by appropriate fire endurance tests. The team also reviewed the FHA to verify the fire loading used by the licensee to determine the fire resistance rating of the fire barrier enclosures. The team also reviewed the installation instructions for sliding fire doors, the design details for mechanical and electrical penetrations, the penetration seal database, Generic Letter 86-10 evaluations, and the fire protection penetration seal deviation analysis for the technical basis of fire barrier penetration seals to verify that the fire barrier installations met design requirements and license commitments. In addition, the team reviewed completed surveillance and maintenance procedures for selected fire barrier features to verify the fire barriers were being adequately maintained.

The team evaluated the adequacy of the fire resistance of fire barrier electrical raceway fire barrier system (ERFBS) enclosures for cable protection to satisfy the applicable separation and design requirements of 10 CFR 50, Appendix R, Section III.G.2. Specifically, the team examined the design drawings, construction details, installation records, and supporting fire endurance tests for the ERFBS enclosures installed in Fire Area 2104, the Unit 2 East Cableway. Visual inspections of the enclosures were performed to confirm that the ERFBS installations were consistent with the design drawings and tested configurations.

The team reviewed abnormal operating fire procedures, selected fire fighting pre-plans, fire damper location and detail drawings, and heating ventilation and air conditioning system drawings to verify that access to shutdown equipment and selected operator manual actions would not be inhibited by smoke migration from one area to adjacent plant areas used to accomplish SSD.

b. Findings

No findings of significance were identified.

.10 Fire Protection Systems, Features, and Equipment

a. Inspection Scope

The team reviewed flow diagrams, cable routing information, and operational valve lineup procedures associated with the fire pumps and fire protection water supply system. The review evaluated whether the common fire protection water delivery and supply components could be damaged or inhibited by fire-induced failures of electrical power supplies or control circuits. Using operating and test procedures, the team toured the fire pump house and diesel-driven fire pump fuel storage tanks to observe the system material condition, consistency of as-built configurations with engineering drawings, and determine correct system controls and valve lineups. Additionally, the team reviewed periodic test procedures for the fire pumps to assess whether the surveillance test program was sufficient to verify proper operation of the fire protection water supply system in accordance with the program operating requirements specified in Appendix B of the FHA.

The team reviewed the adequacy of the fire detection systems in the selected plant fire areas in accordance with the design requirements in Appendix R, III.G.1 and III.G. 2. The team walked down accessible portions of the fire detection systems in the selected fire areas to evaluate the engineering design and operation of the installed configurations. The team also reviewed engineering drawings for fire detector types, spacing, locations and the licensee’s technical evaluation of the detector locations for the detection systems for consistency with the licensee’s FHA, engineering evaluations for NFPA code deviations, and NFPA 72E. In addition, the team reviewed surveillance procedures and the detection system operating requirements specified in Appendix B of the FHA to determine the adequacy of fire detection component testing and to ensure that the detection systems could function when needed.

The team performed in-plant walk-downs of the Unit 2 East Cableway automatic wet pipe sprinkler suppression system to verify the proper type, placement and spacing of the sprinkler heads as well as the lack of obstructions for effective functioning. The team examined vendor information, engineering evaluations for NFPA code deviations, and design calculations to verify that the required suppression system water density for the protected area was available. Additionally, the team reviewed the physical configuration of electrical raceways and safe shutdown components in the fire area to determine whether water from a pipe rupture, actuation of the automatic suppression system, or manual fire suppression activities in this area could cause damage that could inhibit the plant's ability to SSD.

The team reviewed the adequacy of the design and installation of the manual carbon dioxide (CO_2) hose reel suppression system for the diesel generator building switchgear rooms 2E and 2F (Fire Areas 2404 and 2408). The team performed in-plant walk-downs of the diesel generator building CO_2 fire suppression system to determine correct system controls and valve lineups to assure accessibility and functionality of the system, as well as associated ventilation system fire dampers. The team also reviewed the licensee's actions to address the potential for CO_2 migration to ensure that fire suppression and post-fire SSD actions would not be impacted. This was accomplished by the review of engineering drawings, schematics, flow diagrams, and evaluations associated with the diesel generator building floor drain system to determine whether systems and operator actions required for SSD would be inhibited by CO_2 migration through the floor drain system.

b. Findings

No findings of significance were identified.

.11 Compensatory Measures

a. Inspection Scope

The team reviewed Appendix B of the FHA and applicable sections of the FPP administrative procedure regarding administrative controls to identify the need for and to implement compensatory measures for out-of-service, degraded, or inoperable fire protection or post-fire SSD equipment, features, and systems. The team reviewed licensee reports for the fire protection status of Unit 1, Unit 2, and of shared structures, systems, and components. The review was performed to verify that the risk associated with removing fire protection and/or post-fire systems or components, was properly assessed and implemented in accordance with the FPP. The team also reviewed CAP CRs generated over the last 18 months for fire protection features that were out of service for long periods of time. The review was conducted to assess the licensee's effectiveness in returning equipment to service in a reasonable period of time.

b. Findings

No findings of significance were identified.

1R21 Safety System Design And Performance Capability

.01 Design Change Request 91-134, SRV Backup Actuation Via Pressure Transmitter Signals

a. Inspection Scope

The team performed an independent design review of plant modification DCR 91-134 in order to evaluate the technical adequacy of the design change package. The scope of the review and circuit analysis performed by the team was limited to the Group A SRVs for which the licensee takes credit in mitigating a fire in the fire areas selected for the inspection.

b. Findings

Introduction:

An inadequate plant modification, DCR 91-134, failed to implement the design input requirements of "one-out-of-two taken twice" logic for the SRV's backup actuation using PT signals.

Description:

DCR 91-134 was implemented in response in to concerns raised in General Electric Report NEDC-3200P, Evaluation of SRV Performance during January-February 1991 Turbine Trip Events for Plant Hatch Units 1 and 2. In order to ensure that individual SRVs will actuate at or near the appropriate set point and within allowable limits, a backup mode of operation for the SRVs was implemented by this DCR. The design was intended to mitigate the effects of corrosion-induced set point drift of the Target Rock SRVs.

Automatically controlled, two stage SRVs are installed on the main steam lines inside containment for the purpose of relieving nuclear boiler pressure either by normal mechanical action or by automatic action of an electro-pneumatic control system. Each SRV can be manually controlled by use of a two position switch located in the main control room. When placed in the “Open” position, the switch energizes the pilot valve of the individual SRV and causes it to go open. When the switch is placed in the “Auto” position, the SRV is opened upon receipt of either an Automatic Depressurization System (ADS), or Low-Low Set (LLS) control logic signal. Either signal will initiate opening of the valve. DCR 91-134 provided a backup mode for initiation of electrical trip of the pilot valve solenoid which was independent of ADS or LLS logic. The backup mode required no operator action to initiate opening of the SRVs and was considered a “blind control loop” to the operators, (i.e., there are no instruments that provide the operators information concerning the open/close status of the SRVs.)

The scope of the plant modification involved the installation of four Rosemount PTs (Model No. 1154GP9RJ), 0-3000 psig, in the 2H21-P404 and -P405 instrument racks at Elevation 158 of the reactor building. Each PT formed part of a 4 to 20 ma current loop and provided the analog trip signal for SRV actuation within the following set point groups:

SRV Group	SRV Identification Tags	SRV Set Point
A	2B21-F013B, D, F, and G	1120 psig
B	2B21-F013A, C, K, and M	1130 psig
C	2B21-F013E, H, and L	1140 psig

Pressure transmitters 2B21-N127A and 2B21-N127C were wired to Analog Transmitter Trip System (ATTS) cabinet 2H11-P927. Pressure transmitter 2B21-N127A instrument loop components consisted of a trip unit master relay K308C and trip unit slave relays K321C and K332C. The loop components for PT 2B21-N127C consisted of a trip unit master relay K335C in addition to trip unit slave relays K336C and K363C. These two instrument loops constituted a “division” of pressure monitoring channels and were intended to provide the "one-out-of-two" logic signal from this division for initiating SRV backup actuation.

Additionally, PTs 2B21-N127B and 2B21-N127D were wired to ATTS cabinet 2H11-P928. Pressure transmitter 2B21-N127B instrument loop components consisted of a trip unit master relay K310D and trip unit slave relays KK312D and K332D. The loop components for PT 2B21-N127D consisted of a trip unit master relay K335D in addition to trip unit slave relays K336D and K363D. These two instrument loops constituted a separate “division” pressure monitoring channels and were intended to provide the "one-out-of-two" logic signal from this division for initiating SRV backup actuation. The design objective of having two instrument channels was to assure compliance with HNP-2-FSAR, Section 15.1.6.1, Application of Single Failure Criteria. This criteria requires for anticipated operational occurrences that the protection sequences within mitigation systems be single component failure proof. A failure of one instrument channel in a division will therefore not eliminate the protection provided by either of the instrument channels.

The following table identifies the division, PT loops and the associated trip unit master and slave relays:

Division	PT Loops	Trip Unit Master Relays	Trip Unit Slave Relays
A	2B21-N127A	K308C	K321C and K332C
	2B21-N127C	K335C	K336C and K363C
B	2B21-N127B	K310D	K312D and K332D
	2B21-N127D	K335D	K336D and K363D

The Group A SRVs were provided logic input signals from the trip unit master relays. The Group B and C SRVs were provided logic input signals from the trip unit slave relays. The 12 relays described above, (6 in ATTS cabinet 2H11-P927 and 6 in ATTS cabinet 2H11-P928), were intended to be wired to provide “one-out-of-two taken twice" logic for actuation of the SRVs. The design objective was to assure that a single relay failure in either division would not cause an inadvertent SRV actuation. Coincident logic input is required from both division instrument loops in order to initiate a SRV backup actuation using the PT signals. This occurs when the circuit, used to energize the individual SRV pilot valve to open the SRV, is enabled by receiving simultaneous logic inputs from either instrument loop in both divisions.

The team performed a circuit analysis of SRV 2B21-F013F (Path 1) and SRV 2B21-F013G (Path 2) in order to verify that the design objectives of implementing a "one-out-of-two taken twice" logic had been achieved. Based on this review the team determined that the design objective of implementing a "one-out-of-two taken twice" logic had not been installed for the SRVs. The logic installed for the SRVs was a "two-out-of-two taken twice" logic in addition to a "one-out-of-two taken twice" logic. The coincident logic implemented using trip unit master relays K310D and K335D could result in spurious actuation of Group A SRVs for a fire in Fire Area 2104. In addition, this spurious actuation defeats the capability to manually control these SRVs. Whenever a SRV lifts, it will remain open until nuclear boiler pressure is reduced to about 85% of its overpressure lift setpoint. However, because the instrument loops have failed high, the trip unit master relays and the trip unit slave relays will continue to energize the pilot valve of the individual SRV and keep the SRV open. As a result, this failure mode prevents the operators from manually controlling the Group A SRVs as is required per the SSAR.

Analysis: This finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating system cornerstone. The team determined that the finding had potential safety significance greater than very low safety significance because it prevented the operators from manually controlling the Group A SRVs which the licensee credited with mitigating a fire in Fire Area 2104. Manual control of the Group A SRVs is required to ensure that the suppression pool temperature will not exceed the heat capacity temperature limit (HCTL) for the suppression pool. Failure to ensure that the suppression pool temperature will not exceed the HCTL could result in loss of net positive suction head for the Core Spray pumps which the licensee credits for mitigating this event. However, the finding remains unresolved pending completion of a significance determination.

Enforcement: 10 CFR 50, Appendix B, Criterion III, requires that design control measures shall provide for verifying or checking the adequacy of design.

DCR 91-134 specified design input requirements for the sensor initiated logic that electrically activates the SRVs to be a "one-out-of-two taken twice" logic scheme. It also identified the potential worst case failure mode of this logic modification as a short in the logic which would result in an inadvertent opening of a SRV. It concluded that the modification was designed so that the actuation logic would not fail to cause inadvertent opening of a SRV nor prevent a SRV from lifting upon ADS/LLS activation. Contrary to the above, the logic implemented by the licensee for DCR 91-134 was different from the specified design input requirements. The independent design verification performed for DCR 91-134 failed to identify this error in the logic scheme. Additionally, the Appendix R Impact Review performed for DCR 91-134 failed to identify the potential failure mode of all eleven SRVs because of fire-induced damage in Fire Area 2104.

Based on the logic input from trip unit master unit relays K310D, and K335D and their associated trip unit slave relays, the plant modification installed for DCR 91-134 failed to correctly implement the "one-out-of-two taken twice" logic that was specified in the SRV backup actuation via PT signals design change package. This failure has created a condition where fire-induced failures of two reactor pressure instrument circuit cables, (within close proximity to each other), could result in spurious actuation of all eleven SRVs with the eleven SRVs subsequently remaining open. Pending completion of a significance determination by the NRC, this item is identified as URI 50-366/03-06-06, Inspector Concerns Associated with Implementation of DCR 91-134.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of licensee audits, self-assessments, and CRs to verify that items related to fire protection and to SSD were appropriately entered into the licensee's CAP in accordance with the Hatch quality assurance program and procedural requirements. The items selected were reviewed for classification and appropriateness of the corrective actions taken or initiated to resolve the issues. In addition, the team reviewed the licensee's applicability evaluations and corrective actions for selected industry experience issues related to fire protection. The operating experience reports were reviewed to verify that the licensee's review and actions were appropriate.

The team reviewed licensee audits and self-assessments of fire protection and safe shutdown to assess the types of findings that were generated and to verify that the findings were appropriately entered into the licensee's corrective action program.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The lead inspector presented the inspection results to licensee management and other members of the licensee's staff at the conclusion of the onsite inspection on July 25, 2003. Subsequent to the onsite inspection, the lead inspector and the Team Leader, Fire Protection, held a follow-up exit by telephone with Mr. S. Tipps and other members of licensee management on September 2, 2003, to update the licensee on changes to the preliminary inspection findings. The licensee acknowledged the findings.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

M. Beard, Acting Engineering Support Supervisor
V. Coleman, Quality Assurance Supervisor
M. Dean, Nuclear Specialist, Fire Protection
R. Dedrickson, Assistant General Manager for Plant hatch
B. Duval, Chemistry Superintendent
M. Googe, Maintenance Manager
J. Hammonds, Operations Manager
D. Javorka, Administrative Assistant, Senior
R. King, Acting Engineering Support Manager
I. Luker, Senior Engineer, Licensing
T. Metzer, Acting Nuclear safety and Compliance Manager
A. Owens, Senior Engineer, Fire Protection
D. Parker, Senior Engineer, Electrical
J. Payne, Senior Engineer, Corrective Action Program
J. Rathod, Bechtel Engineering Group Supervisor
M. Raybon, Summer Intern
K. Rosanski, Oglethorpe Power Corporation Resident Manager
S. Tipps, Nuclear Safety and Compliance Manager
J. Vance, Senior Engineer, Mechanical & Civil
R. Varnadore, Outages and Modifications Manager

NRC personnel:

N. Garret, Senior Resident Inspector
C. Payne, Fire Protection Team Leader

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-366/03-06-01	URI	Concerns Associated with Potential Opening of SRVs (Section 1R05.03.b)
50-366/03-06-02	URI	Untimely and Unapproved Manual Operator Action for Post-Fire SSD (Section 1R.04/05.b.1)
50-366/03-06-06	URI	Inspector Concerns Associated with Implementation of DCR 91-134 (Section 1R21.01.b)

Opened and Closed

50-366/03-06-03	NCV	Inadequate Procedure for Local Manual Operator Action for Post-Fire SSD Equipment (Section 1R.04/05.b.2)

50-366/03-06-04	NCV	Unapproved Manual Operator Actions for Post-Fire SSD (Section 1R.04.05.b.3)
50-366/03-06-05	NCV	Inadequate Emergency Lighting for Operation of Post-Fire SSD Equipment (Section 1R05.07.b)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Procedures

Administrative Procedure 40AC-ENG-008-0S, Fire Protection Program, Rev. 9.2
Administrative Procedure 42FP-FPX-018-0S, Use, Control, and Storage of Flammable/Combustible Materials, Rev. 1.0
Department Instruction DI-FPX-02-0693N, Fire Fighting Equipment Inspection, Rev. 5
Fire Protection Procedure 42FP-FPX-005-0S, Drill Planning, Critiques and Drill Documentation Rev. 1 ED1
Fire Protection Procedure 42FP-FPX-007-0S, Hot Work, Rev. 1.2
Preventive Maintenance Procedure 52PM-MEL-012-0, Low Voltage Switchgear Preventive Maintenance, Rev. 25.0
Preventive Maintenance Procedure 52PM-MEL-014-0, Transformer Maintenance, Rev. 10.1
Surveillance Procedure 42SV-FPX-002-0S, Low Pressure CO_2 System Surveillance, Rev. 7.1
Surveillance Procedure 42SV-FPX-004-0S, Fire Pump Test, Rev. 8.6
Surveillance Procedure 42SV-FPX-006-0S, Fire Damper Surveillance, Rev. 1 ED 1
Surveillance Procedure 42SV-FPX-021-OS, Surveillance of Swinging Fire Doors, Rev. 1.6
Surveillance Procedure 42SV-FPX-024-OS, Fire Hose Stations 31 Day Surveillance, Rev. 1
Surveillance Procedure 42SV-FPX-030-OS, Fire Emergency Self Contained Breathing Apparatus Inspection and Test, Rev. 1
Surveillance Procedure 42SV-FPX-032-0S, Automatic Sliding Fire Door Visual Inspection, Rev. 3.3
Surveillance Procedure 42SV-FPX-036-0S, Annual Fire Pump Capacity Test, Rev. 8.6
Surveillance Procedure 42SV-FPX-037-OS, Fire Detection Instrumentation Surveillance, Rev. 5.1
System Operating Procedure 34SO-X43-001-1, Fire Pumps Operating Procedure, Rev. 4.3
Training Procedure 73TR-TRN-003-0S, Fire Training Program, Rev.4
AOP 34AB-C11-001-2, Loss of CRD System, Version 2.3
AOP 34AB-C71-001-2, Scram Procedure, Version 9.9
AOP 34AB-C71-002-2, Loss of RPS, Version 4.3
AOP 34AB-N61-002-2S, Main Condenser Vacuum Low, Version 0.4
AOP 34AB-P41-001-2, Loss of Plant Service Water, Version 8.1
AOP 34AB-P42-001-2S, Loss of Reactor Building Closed Cooling Water, Version 1.4
AOP 34AB-P51-001-2, Loss of Instrument and Service Air System or Water Intrusion into the Service Air System, Version 3.0
AOP 34AB-R22-001-2, Loss of DC Busses, Version 2.4
AOP 34AB-R22-002-2, Loss of 4160V Emergency Bus, Version 1.4
AOP 34AB-R22-003-2, Station Blackout, Version 2.3
AOP 34AB-R22-004-02, Loss of 4160V Bus 2A, 2B, 2C, or 2D, Version 1.3
AOP 34AB-R23-001-2S, Loss of 600V Emergency Bus, Version 0.4
AOP 34AB-R24-001-2, Loss of Essential AC Distribution Buses, Version 1.3
AOP 34AB-R25-002-02, Loss of Instrument Buses, Version 5.4
AOP 34AB-T47-001-2, Complete Loss of Drywell Cooling, Version 1.8
AOP 34AB-X43-001-2, Fire Procedure, Version 10.8
AOP 34AB-X43-002-0, Fire Protection System Failures, Version 1.3
SOP 34SO-C71-001-2, 120VAC RPS Supply System, Version 10.2

SOP 34SO-N40-001-2, Main Generator Operation, Version 10.8
SOP 34SO-R42-001-2S, 125V DC and 125/250 VDC System, Version 7.1
SOP 34SO-S22-001-2, 500 KV Substation Switching, Version 5.2
31EO-EOP-010-2S, RC RPV Control (Non-ATWS), Rev. 8, Attachment 1
31EO-EOP-012-2S, PC-1 Primary Containment Control, Rev. 4, Attachment 1
31EO-EOP-013-2S, PC-2 Primary Containment Control, Rev. 4, Attachment 1
31EO-EOP-014-2S, SC - Secondary Containment Control, Rev. 6, Attachment 1
31EO-EOP-016-2S, CP-2 RPV Flooding, Rev. 8, Attachment 1
Procedure 34AB-X43-001-2S, Rev.10ED3, "Fire Procedure," dated 5/28/03.
Calibration Procedure 57CP-CAL-097-2, Rosemount 1153 and 1154 transmitters, Revision No. 19.9.

Drawings

H-11814, Fire Hazards Analysis, Control Bldg. El. 130'-0", Rev. 5
H-11821, Fire Hazards Analysis, Turbine Bldg. El. 130'-0", Rev. 0
H-11846, Fire Hazards Analysis, Diesel Generator Bldg., Rev. 2
H-26014, R.H.R. System P&ID Sheet 1, Rev. 49
H-26015, R.H.R. System P&ID Sheet 2, Rev. 46
H-26018, Core Spray System P&ID, Rev. 29
B-10-1326, Rectangular Fire Damper Schedule, Rev. 2
B-10-1329, Rectangular Fire Damper, Rev. 1
H-11033, Fire Protection Pump House Layout, Rev. 47
H-11035, Fire Protection Piping and Instrumentation Diagram, Rev. 22
H-11226, Piping-Diesel Generator Building Drainage, Rev. 6
H-11814, Fire Hazards Analysis Drawing, Control Building, Rev. 5
H-11821, Fire Hazards Analysis Drawing, Turbine Building, Rev. 11
H-11846, Fire Hazards Analysis Drawing, Diesel Generator Building, Rev. 2
H-11894, Fire Detection Equipment Layout-Diesel Generator Building, Rev. 2
H-11915, Fire Detection Equipment Layout-Control Building, Rev. 2
H-13008, Conduit and Grounding, Fire Pump House, Rev. 9
H-13615, Wiring Diagram, Fire Pump House, Rev. 13
H-16054, Control Building HVAC System, Rev. 19
H-41509, Diesel Generator Building CO_2 System-P&ID, Rev. 5
H-43757, Penetration Seals-Type, Number, and as-Built Location, Rev. 3

Calculations, Analyses, and Evaluations

E. I. Hatch Nuclear Plant Units 1 and 2 Safe Shutdown Analysis Report, Rev. 20.
Edwin I. Hatch Nuclear Plant Fire Hazards Analysis and Fire Protection Program, Rev. 20
Calculation SMFP88-001, Hydraulic Analysis of Sprinkler Systems in Control Building East Cableway, dated 03/11/1988
Calculation SMNH94-046, FCF-F10B-006, Fire Resistance of Concrete Block at HNP, dated 09/30/1994
Calculation SMNH94-048, FCF-F10B-006, Cable Tray Combustible Loading Calculation, dated 09/30/1994

Calculation SMNH98-023, HT-98617, Fire Protection Penetration Seal Deviation Analysis, dated 10/28/1998
Calculation SMNH00-011, HT-00606, Hose Nozzle Pressure Drop Analysis, dated 09/08/2000
Evaluation HT-91722, Fire Protection Code Deviation Resolution, dated 04/22/1992
Hatch Response to NRC IN 1999-005, dated 05/04/1999
Hatch Response to NRC IN 2002-024, dated 09/20/2002
Calculation SENH 98-003, Rev. 0, plot K, protective relay settings 4kV bus 2E
Calculation 85082MP, Plot 29, 600V Switchgear 2C
Calculation SENH 94-004, Attachment A, Sheets 7&8, 600/208 Reactor Building MCC 2C
Calculation SENH 91-011, Attachment P, Sheet 6, Reactor Building DC MCC 2A
Calculation SENH 94-013, Sheets 28 and 29, 600V Reactor Building MCC 2E-B
Calculation SENH 91-011, Attachment P, Sheet 16, Reactor Building 250VDC MCC 2B

Audits and Self-Assessments

Audit No. 01-FP-1, Audit of the Fire Protection Program, dated April 12, 2001
Audit No. 02-FP-1, Audit of the Fire Protection Program, dated February 28, 2002
Audit No. 03-FP-1, Audit of Fire Protection, dated April 21, 2003
1999-001106, Lighting in Fire Equipment Building
2002-000629, Inordinate Number of Buried Piping Leaks
2002-002127, Inadequate Bunker Gear
2002-002129, Health Physics Support and Participation for Fire Brigade
2003-000735, Impact on Cold Weather on Operating Units
Audit Report 01-FP-1, Audit of Fire Protection Program, dated 04/12/2001
Audit Report 02-FP-1, Audit of Fire Protection Program, dated 02/28/2002
Audit Report 03-FP-1, Audit of Fire Protection Program, dated 04/21/2003

CRs Reviewed

CR 2000007119, Fire Procedure 34AB-X43-001-1S Needs to be Enhanced
CR 2001002032, Fire Procedure 34AB-X43-001-2S Needs Actions for Diesel Fuel Oil Pumps
CR 2003004377, Fire Procedure 34AB-X43-001-1 Enhancements
CR 2003004379, Fire Procedure 34AB-X43-001-2 Enhancements
CR 2003004382, SSAR Discrepancies

CRs Generated During this Inspection

CR 2003007129, No Fire Procedure Actions for a Fire in the 2C Switchgear Room
CR 2003007719, Use of Link Wrench
CR 2003007978, Fire Damper Corrective Action
CR 2003008141, Breaker Maintenance Handle
CR 2003008165, SSAR Section 2.100
CR 2003008179, Drywell Access Emergency Lights
CR 2003008181, Link Labeling
CR 2003008202, Manually Opening MOV 2E11-F015A
CR 2003008203, SRV Manual Action Steps in Fire Procedure
CR 2003008237, Emergency Lights and Component Labeling for Manual Actions

CR 2003008238, CO2 Migration Through Floor Drains
CR 2003800132, SSAR Error for Position of 2E11-F004A
CR 2003800151, Instruments for Manual Actions
CR 2003800152, Sliding Links in SSAR
CR 2003800153, Promat Test Report
CR 2003008250, Communications for Post-Fire SSD
CR 2003800166, Review Fire Procedure Step 34AB-X43-001-2 Steps to Verify Compliance with Appendix R.

Design Criteria and Standards

Design Philosophy for Fire Detectors at E. I. Hatch Nuclear Plants, Rev. 2

Completed Surveillance Procedures and Test Records

42SV-FPX-021-OS, Surveillance of Swinging Fire Doors, Task # 1-3367-1 (completed on 01/09/2003)
42SV-FPX-024-OS, Fire Hose Stations, Task # 1-3359-1 (completed on 06/27/2003)
42SV-FPX-030-OS, Fire Emergency Self Contained Breathing Apparatus Inspection and Test, Task # 1-4200-3 (completed on 07/07/2003)
42SV-FPX-032-OS, Automatic Sliding Fire Door Surveillance, Task # 1-3361-2 (completed on 08/13/2002
Promatec Technologies Installation Inspection Report for Fire Area 2104, MWO 2-98-00881, Record 09367-2289, dated 09/03/1998

Technical Manuals/Vendor Information

Dow Corning Fire Endurance Test on Penetration Seal Systems in Precast Concrete F Using Silicone Elastomers, dated 10/28/1975
Dow Corning 561 Silicone Transformer Fluid Technical Manual,10-453-97, dated 1997
S-80393, Mesker Instructions for Installing d&H “Pyromatic” Automatic Sliding Fire Door Closer
S-27874B, General Electric Instruction Book GEK-26501, Liquid-Filled Secondary Unit Substation Transformers, Rev. 2
S-52429A, Bisco, Fire Rated Penetration Seal Qualification Data, dated 08/16/1990
S-52480, Factory Mutual, Fire Rated Penetration Seal Qualification Data-Chemtrol Design FC-225, dated 08/31/1990
S-54875B, Promatec, Fire Barriers-Unit 2 East Cableway, Rev. 2
Omega Point Laboratories, SR90-005, Three Hour Wall Test, dated 06/06/1990
Promatec Technologies Inc., PSI-001, Issue 1, General Construction Details, dated 07/21/1998
Promatec Technologies Inc., IP-2031, Installation Inspection for Promat’s Three Hour Solid Wall/Ceiling Protection System, Issue C, dated 06/16/1998
System Information Document No. SI-LP-01401-03, Main Steam and Low Low Set System, dated 4/3/2000

Applicable Codes and Standards

ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants
NFPA 12, Standard for Carbon Dioxide Systems, 1973 Edition.
NFPA 13, Standard for the Installation of Sprinkler Systems, 1976 Edition.
NFPA 14, Standard for the Installation of Standpipe and Hose Systems, 1974 Edition.
NFPA 20, Standard for the Installation of Centrifugal Fire Pumps, 1973 Edition.
NFPA 72D, Standard for the Installation, Maintenance, and Use of Proprietary Protection Signaling Systems, 1975 Edition.
NFPA 72E, Standard on Automatic Fire Detectors, 1974 Edition
NFPA 80, Standard on Fire Doors and Windows, 1975 Edition.
NUREG-1552, Supplement 1, Fire Barrier Penetration Seals in Nuclear Power Plants, dated January 1999
OSHA Standard 29 CFR 1910, Occupational Safety and Health Standards,
Underwriters Laboratory, Fire Resistance Directory, January 1998

Other Documents

Design Change Package 91-009, Retrofill Dielectric Fluid on Unit 2 Transformers, Rev. 1
Fire Protection Inspection Reports for the period 2001-2002
Fire Service Qualification Training, FP-LP-10003, Fire Fighter Safety, dated 01/14/2002
Fire Service Qualification Training, FP-LP-10004, Fire Fighter Personal Protective Equipment, dated 01/14/2002
Fire Service Qualification Training, FP-LP-10014, Fire Streams, dated 01/22/2002
Fire Service Qualification Training, FP-LP-10018, Fire Fighting Principles and Practices, dated 01/22/2002
Hatch Response to NRC Information Notice 1999-05, Inadvertent Discharge of Carbon Dioxide Fire Protection System and Gas Migration, dated 05/04/1999
Hatch Response to NRC Information Notice 2002-24, Potential Problems with Heat Collectors on Fire Protection Sprinklers, dated 09/20/2002
10CFR21-001, ELECTRAK Corporation, Software Error within TRAK2000 Cable Management and Appendix R Analysis System, dated 03/07/2003
U. S. Consumer Product Safety Commission, Invensys Building Systems Announce Recall of Siebe Actuators in Building Fire/Smoke Dampers, dated 10/02/2002
Pre-fire Plan A-43965, Power-Block Areas Methodology, Rev. 0
Pre-fire Plan A-43966, Fire Area 2404, Diesel Generator Building Switchgear Room 2E, Rev. 2
Pre-fire Plan A-43966, Fire Area 2408, Diesel Generator Building Switchgear Room 2F, Rev. 2
Pre-fire Plan A-43965, Fire Area 2016, W 600V Switchgear Room 2C, Rev. 4

License Basis Documents

Hatch UFSAR Section 3.4, Water Level Flood Design, Rev. 20
Hatch UFSAR Section 9.1-A, Fire Protection Plan, Rev. 18C
Hatch UFSAR Section 17.2, Quality Assurance During the Operations Phase, Rev. 20B
Hatch Fire Hazards Analysis, Appendix B, Fire Protection Equipment Operating and Surveillance Requirements, Rev. 12B

Hatch Fire Hazards Analysis, Appendix H, Application of National Fire Protection Association Codes, Rev. 12B
Hatch SER dated April 18, 1994
Safe Shutdown Analysis Report for E.I. Hatch Nuclear Plant Units 1 and 2, Rev. 26
Fire Hazards Analysis for E. I. Hatch Nuclear Plant Units 1 and 2, Rev.18C, dated 7/00.
NRC Safety Evaluation Report dated 01/02/1987; Re: Exemption from the requirements of Appendix R to 10 CFR Part 50 for Hatch Units 1 and 2 (response to letter dated May 16, 1986).
Letter dated 05/16/86, From L. T. Guewa (Georgia Power) to D. Muller, NRC/NRR; Re: Edwin I Hatch Nuclear Plant Units 1 and 2 10 CFR 50.48 and Appendix R Exemption Requests

Design Change Request Documents

DCR No. 91-134, SRV Backup Actuation via PT Signals, Revision 0.
Drawing No. H-26000, Nuclear Boiler System P&ID, Sheet 1, Revision 39
Drawing No. H-27403, Automatic Depressurization System 2B21C Elementary Diagram, Sheet 6 of 6, Revision 2
Drawing No. H-27472, Automatic Depressurization System 2B21C Elementary Diagram, Sheet 3 of 6, Revision 2
Drawing No. H-27473, Automatic Depressurization System 2B21C Elementary Diagram, Sheet 4 of 6, Revision 2
Drawing No. H-24427, Elementary Diagram, ATTS System 2A70 Sheet 27 of 35, Revision 3
Drawing No. H-24428, Elementary Diagram, ATTS System 2A70 Sheet 28 of 35, Revision 3
Drawing No. H-24429, Elementary Diagram, ATTS System 2A70 Sheet 29 of 35, Revision 5
Drawing No. H-24430, Elementary Diagram, ATTS System 2A70 Sheet 30 of 35, Revision 3
Drawing No. H-24431, Elementary Diagram, ATTS System 2A70 Sheet 31 of 35, Revision 3
Drawing No. H-24432, Elementary Diagram, ATTS System 2A70 Sheet 32 of 35, Revision 6

LIST OF ACRONYMS

ADS	Automatic Depressurization System
AOP	Abnormal Operating Procedure
APCSB	Auxiliary and Power Conversion System Branch
ATTS	Analog Transmitter Trip System
BTP	Branch Technical Position
CAP	Corrective Action Program
CO_2	Carbon Dioxide
CRs	Condition Reports
CST	Condensate Storage Tank
DCR	Design Change Request
ERFBS	Electrical Raceway Fire Barrier System
FHA	Fire Hazards Analysis
FPP	Fire Protection Program
HCTL	Heat Capacity Temperature Limit
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IP	Inspection Procedure
JPM	Job Performance Measure
LLS	Low-Low Set
LOCA	Loss of Coolant Accident
ma	Milli-amp
MOVs	Motor Operated Valves
NCV	Non-Cited Violations
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
OSHA	Occupational Safety and Health Administration
PT	Pressure Transmitter
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SCBAs	Self-Contained Breathing Apparatuses
SDP	Significance Determination Process
SERs	Safety Evaluation Reports
SRVs	Safety Relief Valves
SSAR	Safe Shutdown Analysis Report
SSD	Safe Shutdown
TS	Technical Specification
UFSAR	Updated Final Safety Evaluation Reports
URI	Unresolved Item
XLPE	Cross-Linked Polyethylene

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-321, 50-366
License Nos.:	DPR-57, NPF-5
Report No.:	05000321/2003006 and 05000366/2003006
Licensee:	Southern Nuclear Operating Company
Facility:	E. I. Hatch Nuclear Plant
Location:	P. O. Box 2010 Baxley, GA. 31513
Dates:	July 7-11, 2003 (Week 1) July 21-25, 2003 (Week 2)
Inspectors:	C. Smith, P. E., Senior Reactor Inspector, (Lead Inspector) R. Schin, Senior Reactor Inspector G. Wiseman, Fire Protection Inspector K. Sullivan, Consultant, Brookhaven National Laboratory
Accompanying Personnel:	S. Belcher, Nuclear Safety Intern, Week 1
Approved by:	Charles R. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure Attachment

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000321/2003-006, 05000366/2003-006; 7/7-11/2003 and 7/21-25/2003; E. I. Hatch Nuclear Plant, Units 1 and 2; Triennial Fire Protection

The report covered an announced two-week period of inspection by three regional inspectors and a consultant from Brookhaven National Laboratory. Three Green non-cited violations (NCVs) and two unresolved items with potential safety significance greater than Green were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

- TBD. The team identified an unresolved item in that a local manual operator action, to prevent spurious opening of all eleven safety relief valves (SRVs) during a fire event, would not be performed in sufficient time to be effective. Also, licensee reliance on this manual action for hot shutdown during a fire, instead of physically protecting cables from fire damage, had not been approved by the NRC.

  This finding is unresolved pending completion of a significance determination. The finding is greater than minor because it affects the objective of the mitigating system cornerstone. Also, the finding has potential safety significance greater than very low safety significance because failure to prevent spurious operation of the SRVs could result in them opening during certain fire scenarios, thereby complicating the post-fire recovery actions. (Section 1R05.04/.05.b.1)

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix R, Section III.G.1 and Technical Specification 5.4.1 because a local manual operator action to operate safe shutdown equipment was too difficult and was also physically unsafe. The licensee had relied on this action instead of providing physical protection of cables from fire damage or preplanning cold shutdown repairs. However, the team determined that some operators would not be able to perform the action.

  The finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating systems cornerstone. This finding is of very low safety significance because the licensee would have time to develop and implement cold shutdown repairs to facilitate accomplishment of the action. (Section 1R05.04/.05.b.2)

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix R, Section III.G.2 in that the licensee relied on some manual operator actions to operate safe shutdown equipment, instead of providing the required physical protection of cables from fire damage without NRC approval.

  The finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating systems cornerstone. Since the actions could reasonably be accomplished by operators in a timely manner, this finding did not have potential safety significance greater than very low safety significance. (Section 1R05.04/.05.b.3)

- Green. The team identified a non-cited violation 10 CFR 50, Appendix R, Section III.J because emergency lighting was not adequate for some manual operator actions that were needed to support post-fire operation of safe shutdown equipment.

  The finding is greater than minor because it affected the reliability objective and the equipment performance attribute of the mitigating systems cornerstone. Since operators would be able to accomplish the actions with the use of flashlights, this finding did not have potential safety significance greater than very low safety significance. (Section 1R05.07.b)

- TBD: The team identified a violation of 10 CFR 50, Appendix B in connection with the implementation of Design Change Request 91-134, SRV Backup Actuation via Pressure Transmitter Signals. The installed plant modification failed to implement the "one-out-of-two taken twice" logic that was specified as a design input requirement in the design change package. Additionally, implementation of a "two-out-of-two coincidence taken twice" logic has introduced a potential common cause failure of all eleven SRVs as a result of the potential for fire-induced damage to two reactor pressure instrumentation circuit cables in close proximity to each other.

  This finding is unresolved pending completion of a significance determination. This finding is greater than minor because it impacts the mitigating system cornerstone. This finding has the potential for defeating manual control of Group A SRVs that are required for ensuring that the suppression pool temperature will not exceed the heat capacity temperature limit for the suppression pool and therefore has a potential safety significance greater than very low safety significance. (Section 1R21.01.b)

B. Licensee-Identified Violations

None

REPORT DETAILS

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

1R05 Fire Protection

The purpose of this inspection was to review the Hatch Nuclear Plant fire protection program (FPP) for selected risk-significant fire areas. Emphasis was placed on verification that the post-fire safe shutdown (SSD) capability and the fire protection features provided for ensuring that at least one redundant train of safe shutdown systems is maintained free of fire damage. The inspection was performed in accordance with the Nuclear Regulatory Commission (NRC) Reactor Oversight Program using a risk-informed approach for selecting the fire areas and attributes to be inspected. The team used the licensee's Individual Plant Examination for External Events and in-plant tours to choose four risk-significant fire areas for detailed inspection and review. The fire areas chosen for review during this inspection were:

- Fire Area 2016, West 600 V Switchgear Room, Control Building, Elevation 130 feet.
- Fire Area 2104, East Cableway, Turbine Building, Elevation 130 feet.
- Fire Area 2404, Switchgear Room 2E, Diesel Generator Building, Elevation 130 feet.
- Fire Area 2408, Switchgear Room 2F, Diesel Generator Building, Elevation 130 feet.

The team evaluated the licensee's FPP against applicable requirements, including Operating License Condition 2.C.(3)(a), Fire Protection; Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), Appendix R; 10 CFR 50.48; Appendix A of Branch Technical Position (BTP) Auxiliary and Power Conversion Systems Branch (APCSB) 9.5-1; related NRC Safety Evaluation Reports (SERs); the Hatch Nuclear Plant Updated Final Safety Analysis Report (UFSAR); and plant Technical Specification (TS). The team evaluated all areas of this inspection, as documented below, against these requirements.

Documents reviewed by the team are listed in the attachment.

.01 Systems Required to Achieve and Maintain Post-Fire Safe Shutdown

a. Inspection Scope

The licensee's Safe Shutdown Analysis Report (SSAR) was reviewed to determine the components and systems necessary to achieve and maintain SSD conditions in the event of fire in each of the selected fire areas. The objectives of this evaluation were as follows:

- Verify that the licensee's shutdown methodology has correctly identified the components and systems necessary to achieve and maintain a SSD condition.
- Confirm the adequacy of the systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring and support system functions.
- Verify that a SSD can be achieved and maintained without off-site power, when it can be confirmed that a postulated fire in any of the selected fire areas could cause the loss of off-site power.
- Verify that local manual operator actions are consistent with the plant's fire protection licensing basis.

b. Findings

The team identified a potential concern in that the licensee used manual actions to disconnect terminal board sliding links in order to isolate two 4 to 20 milli-amp (ma) instrumentation loop control circuits in order to prevent the spurious actuation of eleven safety relief valves (SRVs). This issue is discussed in Section 1R05.03.b of the report. No other findings of significance were identified.

.02 Fire Protection of Safe Shutdown Capability

a. Inspection Scope

For the selected fire areas, the team evaluated the frequency of fires or the potential for fires, the combustible fire load characteristics and potential fire severity, the separation of systems necessary to achieve SSD, and the separation of electrical components and circuits located within the same fire area to ensure that at least one SSD path was free of fire damage. The team also inspected the fire protection features to confirm they were installed in accordance with the codes of record to satisfy the applicable separation and design requirements of 10 CFR 50, Appendix R, Section III.G, and Appendix A of BTP APCSB 9.5-1. The team reviewed the following documents, which established the controls and practices to prevent fires and to control combustible fire loads and ignition sources, to verify that the objectives established by the NRC-approved FPP were satisfied:

- UFSAR Section 9.1-A, Fire Protection Plan
- Administrative Procedure 40AC-ENG-008-0S, Fire Protection Program
- Administrative Procedure 42FP-FPX-018-0S, Use, Control, and Storage of Flammable/Combustible Materials
- Preventive Maintenance Procedure 52PM-MEL-012-0, Low Voltage Switchgear Preventive Maintenance

The team toured the selected plant fire areas to observe whether the licensee had properly evaluated in-situ fire loads and limited transient fire hazards in a manner consistent with the fire prevention and combustible hazards control procedures. In addition, the team reviewed the licensee's fire safety inspection reports and corrective action program (CAP) condition reports (CRs) resulting from fire, smoke, sparks, arcing, and overheating incidents for the years 2000-2002 to assess the effectiveness of the fire prevention program and to identify any maintenance or material condition problems related to fire incidents.

The team reviewed fire brigade response, fire brigade qualification training, and drill program procedures; fire brigade drill critiques; and drill records for the operating shifts from January 1999 - December 2002. The reviews were performed to determine whether fire brigade drills had been conducted in high fire risk plant areas and whether fire brigade personnel qualifications, drill response, and performance met the requirements of the licensee's approved FPP.

The team walked down the fire brigade equipment storage areas and dress-out locker areas in the fire equipment building and the turbine building to assess the condition of fire fighting and smoke control equipment. Fire brigade personal protective equipment located at both of the fire brigade dress-out areas and fire fighting equipment storage area in the turbine building were reviewed to evaluate equipment accessibility and functionality. Additionally, the team observed whether emergency exit lighting was provided for personnel evacuation pathways to the outside exits as identified in the National Fire Protection Association (NFPA) 101, Life Safety Code, and the Occupational Safety and Health Administration (OSHA) Part 1910, Occupational Safety and Health Standards. This review also included examination of whether backup emergency lighting was provided for access pathways to and within the fire brigade equipment storage areas and dress-out locker areas in support of fire brigade operations should power fail during a fire emergency. The fire brigade self-contained breathing apparatuses (SCBAs) were reviewed for adequacy as well as the availability of supplemental breathing air tanks and their refill capability.

The team reviewed fire fighting pre-fire plans for the selected areas to determine if appropriate information was provided to fire brigade members and plant operators to facilitate suppression of a fire that could impact SSD. Team members also walked down the selected fire areas to compare the associated pre-fire plans and drawings with as-built plant conditions. This was done to verify that fire fighting pre-fire plans and drawings were consistent with the fire protection features and potential fire conditions described in the Fire Hazards Analysis (FHA).

The team reviewed the adequacy of the design, installation, and operation of the manual suppression standpipe and fire hose system for the control building. This was accomplished by reviewing the FHA, pre-fire plans and drawings, engineering mechanical equipment drawings, design flow and pressure calculations, and NFPA 14 for hose station location, water flow requirements and effective reach capability. Team members also walked down the selected fire areas in the control building to ensure that hose stations were not blocked and to verify that the required fire hose lengths to reach the safe shutdown equipment in each of the selected areas were available. Additionally, the team observed placement of the fire hoses and extinguishers to assess consistency with the fire fighting pre-fire plans and drawings.

b. Findings

No findings of significance were identified.

.03 Post-Fire Safe Shutdown Capability

a. Inspection Scope

On a sample basis, the inspectors evaluated whether the systems and equipment identified in the licensee's SSAR as being required to achieve and maintain hot shutdown conditions would remain free of fire damage in the event of fire in the selected fire areas. The evaluation included a review of cable routing data depicting the location of power and control cables associated with SSD Path 1 and Path 2 components of the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems. Additionally, on a sample basis, the team reviewed the licensee's analysis of electrical protective device (e.g., circuit breaker, fuse, relay) coordination. The following motor operated valves (MOVs) and other components were reviewed:

Component ID	Description
2E51-F029	RCIC Pump Suction from Suppression Pool Valve
2E51-F010	RCIC Pump Suction Valve from Condensate Storage Tank (CST)
2P41-C001A	Plant Service Water Pump 2A
2E11-F011A	Residual Heat Removal (RHR) Heat Exchanger A Drain to Suppression Pool Valve
2P41-C001B	Plant Service Water Pump 2B
2E41-F001	HPCI Turbine Steam Supply Valve
2E41-F002	HPCI Turbine Steam Supply Inboard Containment Isolation Valve
2E41-F006	HPCI Pump Inboard Discharge Valve
2E41-F008	HPCI Pump Discharge Bypass Test Valve to CST

b. Findings

The team identified a potential concern in that the licensee used manual actions to isolate two 4 to 20 ma instrumentation loop control circuits associated with eleven SRVs in lieu of providing physical protection. This did not appear to be consistent with the plant's licensing basis nor 10 CFR 50, Appendix R. Spurious action of these SRVs could impact the licensee's fire mitigation strategy. In addition, the licensee provided no objective evidence that post-fire safe shutdown equipment could mitigate this event.

The SSAR stated that a fire in Fire Area 2104 could cause all eleven SRVs to spuriously actuate as a result of fire damage to two cables located in close proximity in this area. The specific circuits that could cause this event were identified by the licensee as circuits ABE019C08 and ABE019C09. Each circuit separately provides a 4 to 20 ma instrumentation signal from an SRV high-pressure actuation transmitter 2B21-N127B or 2B21-N127D to its respective master trip unit (2B21-N697B or 2B21-N697D). The purpose of this circuitry was to provide an electrical backup to the mechanical trip capability of the individual SRVs. In the event of high reactor pressure, the circuits would provide a signal to the master trip units which would cause all eleven SRVs to actuate (open). The pressure signal from each transmitter would be conveyed to its respective master trip unit through a two-conductor, instrument cable that was routed through this fire area (two separate cables). Each cable consisted of a single twisted pair of insulated conductors, an uninsulated drain wire that was wound around the twisted pair of conductors, and a foil shield. In Fire Area 2104, the two cables were located in close proximity in the same cable tray. Actuation of the SRV electrical backup is completely "blind" to the operators. That is, unlike ADS, it does not provide any pre-actuation indication (e.g., actuation of the ADS timer) or an inhibit capability (e.g., ADS inhibit switch). Because the operators typically would not initiate a manual scram until fire damage significantly interfered with control of the plant, it is possible that all eleven SRVs could open at 100% power, prior to scramming the reactor. This event could place the plant in an unanalyzed condition.

Unlike a typical control circuit, a direct short or "hot short" between conductors of a 4 to 20 ma instrument circuit may not be necessary to initiate an undesired (false high) signal. For cables that transmit low-level instrument signals, degradation of the insulation of the individual twisted conductors due to fire damage may be sufficient to cause leakage current to be generated between the two conductors. Such leakage current would appear as a false high pressure signal to the master trip units. If both cables were damaged as a result of fire, false signals generated as a result of leakage current in each cable, could actuate the SRV electrical backup scheme which would cause all eleven SRVs to open. The conductor insulation and jacket material of each cable was cross-linked polyethylene (XLPE). Because both cables were in the same tray and exposed to the same heating rate, there would be a reasonable likelihood that both instrumentation cables could suffer insulation damage at the same time and both circuits could fail high simultaneously.

The licensee's SSAR recognized the potential safety significance of this event and described methods that have been developed to prevent its occurrence and/or to mitigate its impact on the plant's post-fire SSD capability (should it occur). To prevent this event, the licensee developed procedural guidance which directs operators to open link BB-10 in panel 2H11-P927 and link BB-10 in panel 2H11-P928. These panels are located in the main control room. Opening of these links would prevent actuation of the SRV trip units by removing the 4 to 20 ma signal fed by the pressure transmitters (PT) to the master trip units. In the event the SRVs were to open prior to the operators completing this action, the SSAR credits core spray loop A to mitigate the event.

The inspection team had several concerns regarding the licensee's approach to this potential spurious actuation of the SRVs. Specific concerns identified by the team include:

1. The links may not be opened in time to preclude inadvertent actuation of the SRVs.

2. The use of links to avoid inadvertent actuation of the SRVs did not appear to be consistent with the current licensing basis.
3. No objective evidence existed to demonstrate that the post-fire SSD equipment could adequately mitigate a fire in Fire Area 2104, if the SRVs were to open.
4. The operations staff would be unable to manually control the Group A SRVs, which are credited for mitigating a fire in Fire Area 2104, should they spuriously actuate as a result of fire-induced damage.

With regard to the timing of operator actions to prevent fire damage from causing all SRVs to open, the licensee performed an evaluation during the inspection which estimated that approximately thirty minutes would pass from the time of fire detection to the time an operator would implement procedural actions to open the links. The inspectors independently arrived at a similar time estimate based on their review of the procedure. In response to inspector's concerns that this interval may be too lengthy to preclude fire damage to the cables of interest and subsequent actuation of the SRVs, the licensee agreed to enhance its existing procedures so that the action would be taken immediately following confirmation of fire in areas where the spurious actuation could occur. This issue is discussed in Section 1R05.04/.05.b.1 of this report.

The team also determined that the opening of terminal board links was not in compliance with the plant's licensing basis. Current licensing basis documents, specifically Georgia Power request for exemption dated May 16, 1986, and a subsequent NRC Safety Evaluation Report (SER) dated January 2, 1987, characterized the opening of links as a repair activity that is not permitted as a means of complying with 10 CFR 50, Appendix R, Section III.G. The inspectors concluded that, the opening of links was considered a repair by both the licensee and the NRC staff in 1987. The licensee could not provide any evidence to justify why these actions should not be characterized as a repair activity in its current SSAR.

Additionally, because there is a potential for all SRVs to spuriously actuate as a result of fire in Fire Area 2104 at a time when RHR is not available, the SSAR credits the use of core spray loop A to accomplish the reactor coolant makeup function. During the inspection, the licensee performed a simulator exercise of an event which caused all 11 SRVs to open. During this exercise, simulator RPV level instruments indicated that core spray would be capable of maintaining level above the top of active fuel. However, the licensee did not provide any objective evidence (e.g., specific calculation or analysis) which demonstrated that, assuming worst-case fire damage in Fire Area 2104, the limited set of equipment available would be capable of mitigating the event in a manner that satisfied the shutdown performance goals specified in 10 CFR 50, Appendix R, Section III.L.1.e.

Finally, the logic that was installed by design change request (DCR) 91-134 for the SRVs was a "two-out-of-two coincidence taken twice" logic in addition to a "one-out-of-two coincidence taken twice" logic. The team determined that the "two-out-of-two" coincidence logic input from trip unit master relays K310D and K335D represented a common cause failure for Group A SRVs for a fire in Fire Area 2104. Specifically, cable ABE019C08 associated with PT 2B21-N127B current loop, and cable ABE019C09 associated with PT 2B21-N127D current loop, were routed in close proximity to each other in the same cable tray in Fire Area 2104. Both shielded twisted pair instrument cables were unprotected from the effects of a fire in this fire area. Fire-induced insulation damage to both cables could result in leakage currents and cause the instrument loops to fail high. This failure mode would simulate a high nuclear boiler pressure condition and would initiate SRV backup actuation of all the Group A SRVs. Whenever a SRV lifted, it would remain open until pressure reduced to about 85% of its overpressure lift setpoint However, the instrument loops, having failed high, would ensure that the trip unit master relays and the trip unit slave relays continued to energize the pilot valve of the individual SRV and keep the SRV open. This issue is discussed in more detail in Section 1R21.01. Ultimately, this failure mode would prevent the operators from manually controlling the Group A SRVs as required per the SSAR.

In response, the licensee initiated CR 2003800152, dated July 24, 2003, to evaluate actions to open links to determine if they are necessary to achieve hot shutdown, and if an exemption from Appendix R is required. Pending additional review by the NRC, this issue is identified as Unresolved Item (URI) 50-366/03-06-01, Concerns Associated with Potential Opening of SRVs.

.04/.05 Alternative Shutdown Capability/Operational Implementation of Alternative Shutdown Capability

a. Inspection Scope

The selected fire areas that were the focus of this inspection all involved reactor shutdown from the control room. None involved abandoning the control room and alternative SSD from outside of the control room. Thus, alternative shutdown capability was not reviewed during this inspection. However, the licensee’s plans for SSD following a fire in the selected areas involved many local manual operator actions that would be performed outside of the control area of the control room. This section of the inspection focused on those local manual operator actions.

The team reviewed the operational implementation of the SSD capability for a fire in the selected fire areas to determine if: (1) the procedures were consistent with the SSAR; (2) the procedures were written so that the operator actions could be correctly performed within the times that were necessary for the actions to be effective; (3) the training program for operators included SSD capability; (4) personnel required to achieve and maintain the plant in hot standby could be provided from the normal onsite staff, exclusive of the fire brigade; and (5) the licensee periodically performed operability testing of the SSD equipment.

The team walked down SSD manual operator actions that were to be performed outside of the control area of the main control room for a fire in the selected fire areas and discussed them with operators. These actions were documented in Abnormal Operating Procedure (AOP) 34AB-X43-001-2, Version 10.8, dated May 28, 2003. The team evaluated whether the local manual operator actions could reasonably be performed, using the criteria outlined in NRC Inspection Procedure (IP) 71111.05, Enclosure 2. The team also reviewed applicable operator training lesson plans and job performance measures (JPMs) and discussed them with operators. In addition, the team reviewed records of actual operator staffing on selected days.

b. Findings

1. Untimely and Unapproved Manual Operator Action for Fire SSD

Introduction: The team found that a local manual operator action to prevent spurious opening of all eleven SRVs would not be performed in sufficient time to be effective. Licensee reliance on this manual action for hot shutdown during a fire, instead of physically protecting cables from fire damage, had not been approved by the NRC.

Description: The team noted that Step 9.3.2.1 of AOP 34AB-X43-001-2, Fire Procedure, Version 10.8, dated May 28, 2003, stated: “To prevent all eleven SRVs from opening simultaneously, open links BB-10 in Panel 2H11-P927 and BB-10 in Panel 2H11-P928.” The team noted that spurious opening of all eleven SRVs should be considered a large loss of coolant accident (LOCA), and that a LOCA should be prevented from occurring during a fire event to comply with 10 CFR 50, Appendix R, Section III.L. Section III.L requires that, during a post-fire shutdown, the reactor coolant system process variables (e.g., reactor vessel pressure and water level) shall be maintained within those predicted for a loss of normal alternating current power. Having all eleven SRVs opened during a fire would challenge this. Additionally, the team observed that this step was sufficiently far back in the procedure that it may not be completed in time to prevent potential fire damage to cables from causing all eleven SRVs to spuriously open.

The licensee had no preplanned estimate of how long it would take operators to complete this step during a fire event. There was no event time line or operator training JPM on this step. The team noted that, during a fire, operators could be using many other procedures concurrent with the Fire Procedure. For example, they could be using other procedures to communicate with the fire brigade about the fire, respond to a reactor trip, deal with a loss of offsite power, and provide emergency classifications and offsite notifications of the fire event. During the inspection, licensee operators estimated that, during a fire event, it could take about 30 minutes before operators would accomplish Step 9.3.2.1. The team concurred with that time estimate which the team had previously determined independently. However, NRC fire models indicated that fires could potentially cause damage to cables in as short a period as five to ten minutes. Consequently, the team concluded that during a fire event, the licensee’s procedures would not ensure that Step 9.3.2.1 would be accomplished in time to prevent potential spurious opening of all eleven SRVs.

The team also identified other issues with Step 9.3.2.1. There was no emergency lighting inside the panels, hence, if the fire caused a loss of normal lighting (e.g., by causing a loss of offsite power), operators would need to use flashlights to perform the actions inside the panels. Consequently, the team considered the emergency lighting for Step 9.3.2.1 to be inadequate (see Section 1R05.07.b). In addition, labeling of the links inside the panels was so poor that operators stated that they would not fully rely on the labeling. Also, the tool that operators would use to loosen and slide the links inside the energized panels was made of steel and was not professionally, electrically insulated. Further, licensee reliance on this operator action, instead of physically protecting the cables as required by 10 CFR 50, Appendix R, Section III.G.2, had not been approved by the NRC.

The licensee stated that cable damage to two reactor pressure instrument cables would be needed to spuriously open all eleven SRVs. Because the licensee stated that the two cables were in the same cable tray in Fire Area 2104, the team considered that a fire in that area could potentially cause all eleven SRVs to spuriously open (see Section 1R21.01.b).

In response to this issue, the licensee initiated CR 2003008203 and promptly revised the Fire Procedure before the end of the inspection, moving the actions of Step 9.3.2.1 to the beginning of the procedure. The procedure change enabled the actions to be accomplished much sooner during a fire in the Unit 2 east cableway or in other fire areas that were vulnerable to the potential for spuriously opening all eleven SRVs. The team determined that this issue is related to associated circuits. As described in NRC IP 71111.05, Fire Protection, inspection of associated circuits is temporarily limited. Consequently, the team did not pursue the cable routing or circuit analysis that would be necessary to evaluate the possibility, risk, or potential safety significance of Group B and C SRVs spuriously opening due to fire damage to the instrument cables. The team did, however, perform a circuit analysis of Group A SRVs for which the licensee takes credit during a fire in Fire Area 2104 (see Section 1R21.01.b)

Analysis: The team determined that this finding was associated with the protection against external factors attribute. It affected the objective of the mitigating system cornerstone to ensure the availability of systems that respond to initiating events and is therefore greater than minor. The team determined that the finding had potential safety significance greater than very low safety significance because failure to prevent spurious operation of the SRVs could result in them opening in certain fire scenarios, thereby complicating the post-fire recovery actions. However, the finding remains unresolved pending completion of the SDP.

Enforcement: 10 CFR 50, Appendix R, Section III.G.2, requires that where cables or equipment, including associated non-safety circuits that could prevent operation or cause mal-operation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of the primary containment, one of the following means of ensuring that one or the redundant trains is free of fire damage shall be provided: 1) a fire barrier with a 3-hour rating; 2) separation of cables by a horizontal distance of more than 20 feet with no intervening combustibles and with fire detectors and automatic fire suppression; or 3) a fire barrier with a 1-hour rating with fire detectors and automatic suppression.

The licensee had not provided physical protection against fire damage for the two instrument cables by one of the prescribed methods. Instead, the licensee had relied on local manual operator actions to prevent the spurious opening of all eleven SRVs. Licensee personnel stated that fire damage to two cables was outside of the Hatch licensing basis and, consequently, there was no requirement to protect the instrument cables. However, the licensee could not provide evidence to support that position.

This potential issue will remain unresolved pending the completion of a significance determination by the NRC. This issue is identified as URI 50-366/03-06-02, Untimely and Unapproved Manual Operator Action for Post-Fire SSD.

2. Local Manual Operator Action was Too Difficult and Physically Unsafe

Introduction: A finding of very low safety significance was identified in that a local manual operator action to operate SSD equipment was too difficult and was also physically unsafe. The team judged that some operators would not be able to perform the action. This finding involved a violation of NRC requirements.

Description: The team observed that Steps 4.15.8.1.1 and 9.3.5.1 of the Fire Procedure relied upon local manual operator actions instead of providing physical protection for cables or providing a procedure for cold shutdown repairs. Both steps required the same local manual operator action: "Manually OPEN 2E11-F015A, Inboard LPCI Injection Valve, as required." This action was to be taken in the Unit 2 drywell access, which was a locked high radiation, contaminated, and hot area with temperatures over 100 degrees F.

Valve 2E11-F015A was a large (24-inch diameter) motor-operated gate valve with a three-foot diameter handwheel. The main difficulty with manually opening this valve was lack of an adequate place to stand. An operator showed the team that to perform the action he would have to climb up to, and stand on a small section of pipe lagging (a curved area about four inches wide by 12 inches long), and then reach back and to his right side, to hold the handwheel with his right hand, while reaching forward and to his right to hold the clutch lever for the motor operator with his left hand. The operator would not have good balance while performing the action. The foothold, which was large enough to support only one foot, was well flattened and appeared to have been used in the past to manually operate this valve. The foothold was about six to seven feet above a steel grating, and the team observed that the space available for potential use of a ladder to better access the 2E11-F015A valve handwheel was not good.

Other difficulties with manually opening the valve included the heat; the need to wear full anti-contamination clothing, a hardhat, and safety glasses; and inadequate emergency lighting (see Section 1R05.07). Also, there was no note or step in the procedure to ensure that the RHR pumps were not running before attempting to manually open the 2E11-F015A valve. If an RHR pump were running, it could create a differential pressure across the valve which could make manually opening it much more difficult. If the operator did not have sufficient agility, strength or stamina, he would be unable to complete the action. Also, the team judged that inability to remove sweat from his eyes, due to wearing gloves that could be contaminated, would be a limiting factor for the operator. In addition, if the operator slipped or lost his balance, he could fall and become injured. Considering all of the difficulties, the team judged that this action was physically unsafe and that some operators would not be able to perform it.

The licensee had no operator training JPM for performing this action and an operator stated that he had not performed or received training on this action. One experienced operator, who appeared to be in much better physical condition that an average nuclear plant operator, stated that he had manually operated the valve in the past, but that it had been very difficult for him.

The team judged that, since this action was not required to maintain hot shutdown but only required for cold shutdown following a fire in one of the four selected fire areas, licensee personnel could have time to improve the working conditions after a fire. They could have time to install scaffolding or temporary ventilation, improve the lighting, and assign multiple operators to manually open the valve. They could have time to perform a cold shutdown repair. However, the licensee had not preplanned any cold shutdown repairs for opening this valve.

Analysis: This finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating systems cornerstone. Because the licensee would have time to develop and implement cold shutdown repairs to facilitate accomplishment of the action, this finding did not impact the effectiveness of one or more of the defense in depth elements. Hence, this finding did not have potential safety significance greater than very low safety significance (Green).

Enforcement: 10 CFR 50, Appendix R, Section III.G.1, requires that fire protection features shall be provided for systems important to safe shutdown and shall be capable of limiting fire damage so that systems necessary to achieve and maintain cold shutdown from either the control room or emergency control stations can be repaired within 72 hours. In addition, TS 5.4.1 requires that written procedures shall be established, implemented, and maintained covering activities including FPP implementation and including the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33 recommends procedures for combating emergencies including plant fires and procedures for operation and shutdown of safety-related boiling water reactor systems. The fire protection program includes the SSAR which requires that valve 2E11-F015A be opened for SSD following a fire in Fire Area 2104, the Unit 2 east cableway. AOP 34AB-X43-001-2, Fire Procedure, Version 10.8, dated May 28, 2003, implements these requirements in that it provides information and actions necessary to mitigate the consequences of fires and to maintain an operable shutdown train following fire damage to specific fire areas. Also, AOP 34AB-X43-001-2 provides Steps 4.15.8.1.1 and 9.3.5.1 for manually opening valve 2E11-F015A following a fire in Fire Area 2104.

Contrary to the above, the licensee had no procedure for repairing any related fire damage within 72 hours. Instead, the licensee relied on local manual operator actions, as described in Steps 4.15.8.1.1 and 9.3.5.1 of AOP 34AB-X43-001-2. However, those procedure steps were inadequate in that some operators would not be able to perform them because the required actions were too difficult and also were physically unsafe. In response to this issue, the licensee initiated CR 203008202. Because the identified inadequate procedure steps are of very low safety significance and the issue has been entered into the licensee's corrective action program, this violation is being treated as an non-cited violation (NCV), consistent with Section VI.A of the NRC's Enforcement Policy: NCV 50-366/03-06-03, Inadequate Procedure for Local Manual Operator Action for Post-Fire Safe Shutdown Equipment.

3. Unapproved Manual Operator Actions for Post-Fire SSD

Introduction: A finding of very low safety significance was identified in that the licensee relied on some local manual operator actions to operate SSD equipment, instead of providing the required physical protection of cables from fire damage. This finding involved a violation of NRC requirements.

Description: The team observed that AOP 34AB-X43-001-2, Fire Procedure, included some local manual operator actions to achieve and maintain hot shutdown that had not been approved by the NRC. Examples of steps from the procedure included:

- Step 4.15.2.2; ...If a loss of offsite power occurs and emergency busses energize ..."Place Station Service battery chargers 2R42-S026 (2R42-S029), 2R42-S027 (2R42-S030) AND 2R42-S028 (2R42-S031) in service per 34SO-R42-001-2."

- Step 4.15.4.5; ...If HPCI fails to automatically trip on high RPV level... "OPEN the following links to energize 2E41-F124, Trip Solenoid Valve, AND to fail 2E41-F3025 HPCI Governor Valve, in the CLOSED position:
  - TT-75 in panel 2H11-P601
  - TT-76 in panel 2H11-P601"

- Step 4.15.4.6; ...If HPCI fails to automatically trip on high RPV level... "OPEN breaker 25 in panel 2R25-S002 to fail 2E41-F3052, HPCI Governor Valve, in the CLOSED position."

The team walked down these actions using the guidance contained in IP 71111.05T and judged that they could reasonably be accomplished by operators in a timely manner. However, the team determined that these operator actions were being used instead of physically protecting cables from fire damage that could cause a loss of station service battery chargers or a HPCI pump runout.

Analysis: The finding is greater than minor because it affected the availability and reliability objectives as well as the equipment performance attribute of the mitigating systems cornerstone. Since the actions could reasonably be accomplished by operators in a timely manner, this finding did not have potential safety significance greater than very low safety significance.

Enforcement: 10 CFR 50, Appendix R, Section III.G.2, requires that where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of the primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided: 1) a fire barrier with a 3-hour rating; 2) separation of cables by a horizontal distance of more than 20 feet with no intervening combustibles and with fire detectors and automatic fire suppression; or 3) a fire barrier with a 1-hour rating with fire detectors and automatic suppression.

Contrary to the above, the licensee had not provided the required physical protection against fire damage for power to the station service battery chargers or for HPCI electrical control cables. Instead, the licensee relied on local manual operator actions, without NRC approval. In response to this issue, the licensee initiated CR 2003800166. Because the issue had very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC's Enforcement Policy: NCV 50-366/03-06-04, Unapproved Manual Operator Actions for Post-Fire Safe Shutdown.

.06 Communications

a. Inspection Scope

The team reviewed the plant communications systems that would be relied upon to support fire brigade and SSD activities. The team walked down portions of the SSD procedures to verify that adequate communications equipment would be available for personnel performing local manual operator actions. In addition, the team reviewed the adequacy of the radio communication system used by the fire brigade to communicate with the main control room.

b. Findings

No findings of significance were identified.

.07 Emergency Lighting

a. Inspection Scope

The team inspected the licensee's emergency lighting systems to verify that 8-hour emergency lighting coverage was provided as required by 10 CFR 50, Appendix R, Section III.J, to support local manual operator actions that were needed for post-fire operation of SSD equipment. During walkdowns of the post-fire SSD operator actions for fires in the selected fire areas, the team checked if emergency lighting units were installed and if lamp heads were aimed to adequately illuminate the SSD equipment, the equipment identification tags, and the access and egress routes thereto, so that operators would be able to perform the actions without needing to use flashlights.

b. Findings

Inadequate Emergency Lighting for Operation of SSD Equipment

Introduction: A finding with very low safety significance was identified in that emergency lighting was not adequate for some manual operator actions that were needed to support post-fire operation of SSD equipment. This finding involved a violation of NRC requirements.

Description: The team observed that emergency lighting was not adequate for some manual operator actions that were needed to support post-fire operation of SSD equipment. Examples included the following operator actions in procedure 34AB-X43-001-2, Fire Procedure, Version 10.8, dated May 28, 2003:

- Step 4.15.2.2; ...if a loss of offsite power occurs and emergency busses energize ..."Place Station Service battery chargers 2R42-S026 (2R42-S029), 2R42-S027 (2R42-S030) AND 2R42-S028 (2R42-S031) in service per 34SO-R42-001-2."

- Step 4.15.4.5; ...If HPCI fails to automatically trip on high RPV level... "OPEN the following links to energize 2E41-F124, Trip Solenoid Valve, AND to fail 2E41-F3025 HPCI Governor Valve, in the CLOSED position:
  - TT-75 in panel 2H11-P601
  - TT-76 in panel 2H11-P601"

- Step 4.15.5; "IF 2R25-S065, Instrument Bus 2B, is DE-ENERGIZED perform the following manual actions to maintain 2C32-R655, Reactor Water Level Instrument, operable:
  - 4.15.5.1; At panel 2H11-P612, OPEN links AAA-11 and AAA-12.
  - 4.15.5.2; At panel 2H11-P601, CLOSE links HH-48 and HH-49."

- Steps 4.15.8.1.1 and 9.3.5.1; "Manually OPEN 2E11-F015A, Inboard LPCI Injection Valve, as required."

- Steps 4.15.8.1.2 and 9.3.5.2; "Manually CLOSE 2E11-F018A, RHR Pump A Minimum Flow Isolation Valve, as required."

- Step 9.3.2.1; "To prevent all 11 SRVs from opening simultaneously, open links BB-10 in Panel 2H11-P927 and BB-10 in Panel 2H11-P928."

- Step 9.3.3; "At Panel 2H11-P627, open links AA-19, AA-20, AA-21, and AA-22, to prevent spurious actuation of SRVs 2B21-F013D AND 2B21-F013G."

- Step 9.3.6; "OPEN link TB9-21 in Panel 2H11-P700 to open Drywell Pneumatic System Inboard Inlet Isolation, 2P70-F005."

- Step 9.3.7; "OPEN link TB1-12 in Panel 2H11-P700 to open Drywell Pneumatic System Outboard Inlet Isolation, 2P70-F005."

- Step 9.3.9.1; "Confirm OR manually CLOSE RHR Shutdown Cooling Valve 2E11-F006D."

- Step 9.3.9.2; "Manually OPEN Shutdown Cooling Suction Valve 2E11-F008, IF required..."

The team verified that flashlights were readily available and judged that operators would be able to use the flashlights and accomplish the actions, with two exceptions. One exception was the action to open terminal board links in two panels to prevent all eleven SRVs from spuriously opening, which was judged to be untimely (see Section 1R05.04/.05.b.1). The other exception was the action to open 2E11-F015A, which was judged to be too difficult (see Section 1R05.04/.05.b.2). For both of these actions, the lack of adequate emergency lighting could make the actions more difficult to complete in a timely manner and increase the chance of operator error.

Analysis: This finding is greater than minor because it affected the reliability objective and the equipment performance attribute of the mitigating systems cornerstone. Since operators would be able to accomplish the actions with the use of flashlights, this finding did not impact the effectiveness of one or more of the defense in depth elements. Hence, this finding did not have potential safety significance greater than very low safety significance (Green).

Enforcement: 10 CFR 50, Appendix R, Section III.J, requires that emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment, and in access and egress routes thereto.

Contrary to the above, emergency lighting units were not adequately provided in all areas needed for operation of SSD equipment. In response this issue, the licensee initiated CRs 2003008237 and 2003008179. Because the identified lack of emergency lighting is of very low safety significance and has been entered into the licensee’s corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC’s Enforcement Policy: NCV 50-366/03-06-05, Inadequate Emergency Lighting for Operation of Post-Fire Safe Shutdown Equipment.

.08 Cold Shutdown Repairs

The licensee had identified no needed cold shutdown repairs. Also, with the exception of the potential need for a cold shutdown repair to open valve 2E11-F015A (see Section 1R05.05.b.2), the team identified no other need for cold shutdown repairs. Consequently, this section of IP 71111.05 was not performed.

.09 Fire Barriers and Fire Area/Zone/Room Penetration Seals

a. Inspection Scope

The team reviewed the selected fire areas to evaluate the adequacy of the fire resistance of fire area barrier enclosure walls, ceilings, floors, fire barrier mechanical and electrical penetration seals, fire doors, and fire dampers. The team selected several fire barrier features for detailed evaluation and inspection to verify proper installation and qualification. This was accomplished by observing the material condition and configuration of the installed fire barrier features, as well as construction details and supporting fire endurance tests for the installed fire barrier features, to verify the as-built configurations were qualified by appropriate fire endurance tests. The team also reviewed the FHA to verify the fire loading used by the licensee to determine the fire resistance rating of the fire barrier enclosures. The team also reviewed the installation instructions for sliding fire doors, the design details for mechanical and electrical penetrations, the penetration seal database, Generic Letter 86-10 evaluations, and the fire protection penetration seal deviation analysis for the technical basis of fire barrier penetration seals to verify that the fire barrier installations met design requirements and license commitments. In addition, the team reviewed completed surveillance and maintenance procedures for selected fire barrier features to verify the fire barriers were being adequately maintained.

The team evaluated the adequacy of the fire resistance of fire barrier electrical raceway fire barrier system (ERFBS) enclosures for cable protection to satisfy the applicable separation and design requirements of 10 CFR 50, Appendix R, Section III.G.2. Specifically, the team examined the design drawings, construction details, installation records, and supporting fire endurance tests for the ERFBS enclosures installed in Fire Area 2104, the Unit 2 East Cableway. Visual inspections of the enclosures were performed to confirm that the ERFBS installations were consistent with the design drawings and tested configurations.

The team reviewed abnormal operating fire procedures, selected fire fighting pre-plans, fire damper location and detail drawings, and heating ventilation and air conditioning system drawings to verify that access to shutdown equipment and selected operator manual actions would not be inhibited by smoke migration from one area to adjacent plant areas used to accomplish SSD.

b. Findings

No findings of significance were identified.

.10 Fire Protection Systems, Features, and Equipment

a. Inspection Scope

The team reviewed flow diagrams, cable routing information, and operational valve lineup procedures associated with the fire pumps and fire protection water supply system. The review evaluated whether the common fire protection water delivery and supply components could be damaged or inhibited by fire-induced failures of electrical power supplies or control circuits. Using operating and test procedures, the team toured the fire pump house and diesel-driven fire pump fuel storage tanks to observe the system material condition, consistency of as-built configurations with engineering drawings, and determine correct system controls and valve lineups. Additionally, the team reviewed periodic test procedures for the fire pumps to assess whether the surveillance test program was sufficient to verify proper operation of the fire protection water supply system in accordance with the program operating requirements specified in Appendix B of the FHA.

The team reviewed the adequacy of the fire detection systems in the selected plant fire areas in accordance with the design requirements in Appendix R, III.G.1 and III.G. 2. The team walked down accessible portions of the fire detection systems in the selected fire areas to evaluate the engineering design and operation of the installed configurations. The team also reviewed engineering drawings for fire detector types, spacing, locations and the licensee's technical evaluation of the detector locations for the detection systems for consistency with the licensee's FHA, engineering evaluations for NFPA code deviations, and NFPA 72E. In addition, the team reviewed surveillance procedures and the detection system operating requirements specified in Appendix B of the FHA to determine the adequacy of fire detection component testing and to ensure that the detection systems could function when needed.

The team performed in-plant walk-downs of the Unit 2 East Cableway automatic wet pipe sprinkler suppression system to verify the proper type, placement and spacing of the sprinkler heads as well as the lack of obstructions for effective functioning. The team examined vendor information, engineering evaluations for NFPA code deviations, and design calculations to verify that the required suppression system water density for the protected area was available. Additionally, the team reviewed the physical configuration of electrical raceways and safe shutdown components in the fire area to determine whether water from a pipe rupture, actuation of the automatic suppression system, or manual fire suppression activities in this area could cause damage that could inhibit the plant's ability to SSD.

The team reviewed the adequacy of the design and installation of the manual carbon dioxide (CO_2) hose reel suppression system for the diesel generator building switchgear rooms 2E and 2F (Fire Areas 2404 and 2408). The team performed in-plant walk-downs of the diesel generator building CO_2 fire suppression system to determine correct system controls and valve lineups to assure accessibility and functionality of the system, as well as associated ventilation system fire dampers. The team also reviewed the licensee's actions to address the potential for CO_2 migration to ensure that fire suppression and post-fire SSD actions would not be impacted. This was accomplished by the review of engineering drawings, schematics, flow diagrams, and evaluations associated with the diesel generator building floor drain system to determine whether systems and operator actions required for SSD would be inhibited by CO_2 migration through the floor drain system.

b. Findings

No findings of significance were identified.

.11 Compensatory Measures

a. Inspection Scope

The team reviewed Appendix B of the FHA and applicable sections of the FPP administrative procedure regarding administrative controls to identify the need for and to implement compensatory measures for out-of-service, degraded, or inoperable fire protection or post-fire SSD equipment, features, and systems. The team reviewed licensee reports for the fire protection status of Unit 1, Unit 2, and of shared structures, systems, and components. The review was performed to verify that the risk associated with removing fire protection and/or post-fire systems or components, was properly assessed and implemented in accordance with the FPP. The team also reviewed CAP CRs generated over the last 18 months for fire protection features that were out of service for long periods of time. The review was conducted to assess the licensee's effectiveness in returning equipment to service in a reasonable period of time.

b. Findings

No findings of significance were identified.

1R21 Safety System Design And Performance Capability

.01 Design Change Request 91-134, SRV Backup Actuation Via Pressure Transmitter Signals

a. Inspection Scope

The team performed an independent design review of plant modification DCR 91-134 in order to evaluate the technical adequacy of the design change package. The scope of the review and circuit analysis performed by the team was limited to the Group A SRVs for which the licensee takes credit in mitigating a fire in the fire areas selected for the inspection.

b. Findings

Introduction:

An inadequate plant modification, DCR 91-134, failed to implement the design input requirements of "one-out-of-two taken twice" logic for the SRV's backup actuation using PT signals.

Description:

DCR 91-134 was implemented in response in to concerns raised in General Electric Report NEDC-3200P, Evaluation of SRV Performance during January-February 1991 Turbine Trip Events for Plant Hatch Units 1 and 2. In order to ensure that individual SRVs will actuate at or near the appropriate set point and within allowable limits, a backup mode of operation for the SRVs was implemented by this DCR. The design was intended to mitigate the effects of corrosion-induced set point drift of the Target Rock SRVs.

Automatically controlled, two stage SRVs are installed on the main steam lines inside containment for the purpose of relieving nuclear boiler pressure either by normal mechanical action or by automatic action of an electro-pneumatic control system. Each SRV can be manually controlled by use of a two position switch located in the main control room. When placed in the “Open” position, the switch energizes the pilot valve of the individual SRV and causes it to go open. When the switch is placed in the “Auto” position, the SRV is opened upon receipt of either an Automatic Depressurization System (ADS), or Low-Low Set (LLS) control logic signal. Either signal will initiate opening of the valve. DCR 91-134 provided a backup mode for initiation of electrical trip of the pilot valve solenoid which was independent of ADS or LLS logic. The backup mode required no operator action to initiate opening of the SRVs and was considered a “blind control loop” to the operators, (i.e., there are no instruments that provide the operators information concerning the open/close status of the SRVs.)

The scope of the plant modification involved the installation of four Rosemount PTs (Model No. 1154GP9RJ), 0-3000 psig, in the 2H21-P404 and -P405 instrument racks at Elevation 158 of the reactor building. Each PT formed part of a 4 to 20 ma current loop and provided the analog trip signal for SRV actuation within the following set point groups:

SRV Group	SRV Identification Tags	SRV Set Point
A	2B21-F013B, D, F, and G	1120 psig
B	2B21-F013A, C, K, and M	1130 psig
C	2B21-F013E, H, and L	1140 psig

Pressure transmitters 2B21-N127A and 2B21-N127C were wired to Analog Transmitter Trip System (ATTS) cabinet 2H11-P927. Pressure transmitter 2B21-N127A instrument loop components consisted of a trip unit master relay K308C and trip unit slave relays K321C and K332C. The loop components for PT 2B21-N127C consisted of a trip unit master relay K335C in addition to trip unit slave relays K336C and K363C. These two instrument loops constituted a “division” of pressure monitoring channels and were intended to provide the "one-out-of-two" logic signal from this division for initiating SRV backup actuation.

Additionally, PTs 2B21-N127B and 2B21-N127D were wired to ATTS cabinet 2H11-P928. Pressure transmitter 2B21-N127B instrument loop components consisted of a trip unit master relay K310D and trip unit slave relays KK312D and K332D. The loop components for PT 2B21-N127D consisted of a trip unit master relay K335D in addition to trip unit slave relays K336D and K363D. These two instrument loops constituted a separate “division” pressure monitoring channels and were intended to provide the "one-out-of-two" logic signal from this division for initiating SRV backup actuation. The design objective of having two instrument channels was to assure compliance with HNP-2-FSAR, Section 15.1.6.1, Application of Single Failure Criteria. This criteria requires for anticipated operational occurrences that the protection sequences within mitigation systems be single component failure proof. A failure of one instrument channel in a division will therefore not eliminate the protection provided by either of the instrument channels.

The following table identifies the division, PT loops and the associated trip unit master and slave relays:

Division	PT Loops	Trip Unit Master Relays	Trip Unit Slave Relays
A	2B21-N127A	K308C	K321C and K332C
	2B21-N127C	K335C	K336C and K363C
B	2B21-N127B	K310D	K312D and K332D
	2B21-N127D	K335D	K336D and K363D

The Group A SRVs were provided logic input signals from the trip unit master relays. The Group B and C SRVs were provided logic input signals from the trip unit slave relays. The 12 relays described above, (6 in ATTS cabinet 2H11-P927 and 6 in ATTS cabinet 2H11-P928), were intended to be wired to provide “one-out-of-two taken twice" logic for actuation of the SRVs. The design objective was to assure that a single relay failure in either division would not cause an inadvertent SRV actuation. Coincident logic input is required from both division instrument loops in order to initiate a SRV backup actuation using the PT signals. This occurs when the circuit, used to energize the individual SRV pilot valve to open the SRV, is enabled by receiving simultaneous logic inputs from either instrument loop in both divisions.

The team performed a circuit analysis of SRV 2B21-F013F (Path 1) and SRV 2B21-F013G (Path 2) in order to verify that the design objectives of implementing a "one-out-of-two taken twice" logic had been achieved. Based on this review the team determined that the design objective of implementing a "one-out-of-two taken twice" logic had not been installed for the SRVs. The logic installed for the SRVs was a "two-out-of-two taken twice" logic in addition to a "one-out-of-two taken twice" logic. The coincident logic implemented using trip unit master relays K310D and K335D could result in spurious actuation of Group A SRVs for a fire in Fire Area 2104. In addition, this spurious actuation defeats the capability to manually control these SRVs. Whenever a SRV lifts, it will remain open until nuclear boiler pressure is reduced to about 85% of its overpressure lift setpoint. However, because the instrument loops have failed high, the trip unit master relays and the trip unit slave relays will continue to energize the pilot valve of the individual SRV and keep the SRV open. As a result, this failure mode prevents the operators from manually controlling the Group A SRVs as is required per the SSAR.

Analysis: This finding is greater than minor because it affected the availability and reliability objectives and the equipment performance attribute of the mitigating system cornerstone. The team determined that the finding had potential safety significance greater than very low safety significance because it prevented the operators from manually controlling the Group A SRVs which the licensee credited with mitigating a fire in Fire Area 2104. Manual control of the Group A SRVs is required to ensure that the suppression pool temperature will not exceed the heat capacity temperature limit (HCTL) for the suppression pool. Failure to ensure that the suppression pool temperature will not exceed the HCTL could result in loss of net positive suction head for the Core Spray pumps which the licensee credits for mitigating this event. However, the finding remains unresolved pending completion of a significance determination.

Enforcement: 10 CFR 50, Appendix B, Criterion III, requires that design control measures shall provide for verifying or checking the adequacy of design.

DCR 91-134 specified design input requirements for the sensor initiated logic that electrically activates the SRVs to be a "one-out-of-two taken twice" logic scheme. It also identified the potential worst case failure mode of this logic modification as a short in the logic which would result in an inadvertent opening of a SRV. It concluded that the modification was designed so that the actuation logic would not fail to cause inadvertent opening of a SRV nor prevent a SRV from lifting upon ADS/LLS activation. Contrary to the above, the logic implemented by the licensee for DCR 91-134 was different from the specified design input requirements. The independent design verification performed for DCR 91-134 failed to identify this error in the logic scheme. Additionally, the Appendix R Impact Review performed for DCR 91-134 failed to identify the potential failure mode of all eleven SRVs because of fire-induced damage in Fire Area 2104.

Based on the logic input from trip unit master unit relays K310D, and K335D and their associated trip unit slave relays, the plant modification installed for DCR 91-134 failed to correctly implement the "one-out-of-two taken twice" logic that was specified in the SRV backup actuation via PT signals design change package. This failure has created a condition where fire-induced failures of two reactor pressure instrument circuit cables, (within close proximity to each other), could result in spurious actuation of all eleven SRVs with the eleven SRVs subsequently remaining open. Pending completion of a significance determination by the NRC, this item is identified as URI 50-366/03-06-06, Inspector Concerns Associated with Implementation of DCR 91-134.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of licensee audits, self-assessments, and CRs to verify that items related to fire protection and to SSD were appropriately entered into the licensee's CAP in accordance with the Hatch quality assurance program and procedural requirements. The items selected were reviewed for classification and appropriateness of the corrective actions taken or initiated to resolve the issues. In addition, the team reviewed the licensee's applicability evaluations and corrective actions for selected industry experience issues related to fire protection. The operating experience reports were reviewed to verify that the licensee's review and actions were appropriate.

The team reviewed licensee audits and self-assessments of fire protection and safe shutdown to assess the types of findings that were generated and to verify that the findings were appropriately entered into the licensee's corrective action program.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The lead inspector presented the inspection results to licensee management and other members of the licensee's staff at the conclusion of the onsite inspection on July 25, 2003. Subsequent to the onsite inspection, the lead inspector and the Team Leader, Fire Protection, held a follow-up exit by telephone with Mr. S. Tipps and other members of licensee management on September 2, 2003, to update the licensee on changes to the preliminary inspection findings. The licensee acknowledged the findings.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

M. Beard, Acting Engineering Support Supervisor
V. Coleman, Quality Assurance Supervisor
M. Dean, Nuclear Specialist, Fire Protection
R. Dedrickson, Assistant General Manager for Plant hatch
B. Duval, Chemistry Superintendent
M. Googe, Maintenance Manager
J. Hammonds, Operations Manager
D. Javorka, Administrative Assistant, Senior
R. King, Acting Engineering Support Manager
I. Luker, Senior Engineer, Licensing
T. Metzer, Acting Nuclear safety and Compliance Manager
A. Owens, Senior Engineer, Fire Protection
D. Parker, Senior Engineer, Electrical
J. Payne, Senior Engineer, Corrective Action Program
J. Rathod, Bechtel Engineering Group Supervisor
M. Raybon, Summer Intern
K. Rosanski, Oglethorpe Power Corporation Resident Manager
S. Tipps, Nuclear Safety and Compliance Manager
J. Vance, Senior Engineer, Mechanical & Civil
R. Varnadore, Outages and Modifications Manager

NRC personnel:

N. Garret, Senior Resident Inspector
C. Payne, Fire Protection Team Leader

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-366/03-06-01	URI	Concerns Associated with Potential Opening of SRVs (Section 1R05.03.b)
50-366/03-06-02	URI	Untimely and Unapproved Manual Operator Action for Post-Fire SSD (Section 1R.04/05.b.1)
50-366/03-06-06	URI	Inspector Concerns Associated with Implementation of DCR 91-134 (Section 1R21.01.b)

Opened and Closed

50-366/03-06-03	NCV	Inadequate Procedure for Local Manual Operator Action for Post-Fire SSD Equipment (Section 1R.04/05.b.2)

50-366/03-06-04	NCV	Unapproved Manual Operator Actions for Post-Fire SSD (Section 1R.04.05.b.3)
50-366/03-06-05	NCV	Inadequate Emergency Lighting for Operation of Post-Fire SSD Equipment (Section 1R05.07.b)

Discussed

None

LIST OF DOCUMENTS REVIEWED

Procedures

Administrative Procedure 40AC-ENG-008-0S, Fire Protection Program, Rev. 9.2
Administrative Procedure 42FP-FPX-018-0S, Use, Control, and Storage of Flammable/Combustible Materials, Rev. 1.0
Department Instruction DI-FPX-02-0693N, Fire Fighting Equipment Inspection, Rev. 5
Fire Protection Procedure 42FP-FPX-005-0S, Drill Planning, Critiques and Drill Documentation Rev. 1 ED1
Fire Protection Procedure 42FP-FPX-007-0S, Hot Work, Rev. 1.2
Preventive Maintenance Procedure 52PM-MEL-012-0, Low Voltage Switchgear Preventive Maintenance, Rev. 25.0
Preventive Maintenance Procedure 52PM-MEL-014-0, Transformer Maintenance, Rev. 10.1
Surveillance Procedure 42SV-FPX-002-0S, Low Pressure CO_2 System Surveillance, Rev. 7.1
Surveillance Procedure 42SV-FPX-004-0S, Fire Pump Test, Rev. 8.6
Surveillance Procedure 42SV-FPX-006-0S, Fire Damper Surveillance, Rev. 1 ED 1
Surveillance Procedure 42SV-FPX-021-OS, Surveillance of Swinging Fire Doors, Rev. 1.6
Surveillance Procedure 42SV-FPX-024-OS, Fire Hose Stations 31 Day Surveillance, Rev. 1
Surveillance Procedure 42SV-FPX-030-OS, Fire Emergency Self Contained Breathing Apparatus Inspection and Test, Rev. 1
Surveillance Procedure 42SV-FPX-032-0S, Automatic Sliding Fire Door Visual Inspection, Rev. 3.3
Surveillance Procedure 42SV-FPX-036-0S, Annual Fire Pump Capacity Test, Rev. 8.6
Surveillance Procedure 42SV-FPX-037-OS, Fire Detection Instrumentation Surveillance, Rev. 5.1
System Operating Procedure 34SO-X43-001-1, Fire Pumps Operating Procedure, Rev. 4.3
Training Procedure 73TR-TRN-003-0S, Fire Training Program, Rev.4
AOP 34AB-C11-001-2, Loss of CRD System, Version 2.3
AOP 34AB-C71-001-2, Scram Procedure, Version 9.9
AOP 34AB-C71-002-2, Loss of RPS, Version 4.3
AOP 34AB-N61-002-2S, Main Condenser Vacuum Low, Version 0.4
AOP 34AB-P41-001-2, Loss of Plant Service Water, Version 8.1
AOP 34AB-P42-001-2S, Loss of Reactor Building Closed Cooling Water, Version 1.4
AOP 34AB-P51-001-2, Loss of Instrument and Service Air System or Water Intrusion into the Service Air System, Version 3.0
AOP 34AB-R22-001-2, Loss of DC Busses, Version 2.4
AOP 34AB-R22-002-2, Loss of 4160V Emergency Bus, Version 1.4
AOP 34AB-R22-003-2, Station Blackout, Version 2.3
AOP 34AB-R22-004-02, Loss of 4160V Bus 2A, 2B, 2C, or 2D, Version 1.3
AOP 34AB-R23-001-2S, Loss of 600V Emergency Bus, Version 0.4
AOP 34AB-R24-001-2, Loss of Essential AC Distribution Buses, Version 1.3
AOP 34AB-R25-002-02, Loss of Instrument Buses, Version 5.4
AOP 34AB-T47-001-2, Complete Loss of Drywell Cooling, Version 1.8
AOP 34AB-X43-001-2, Fire Procedure, Version 10.8
AOP 34AB-X43-002-0, Fire Protection System Failures, Version 1.3
SOP 34SO-C71-001-2, 120VAC RPS Supply System, Version 10.2

SOP 34SO-N40-001-2, Main Generator Operation, Version 10.8
SOP 34SO-R42-001-2S, 125V DC and 125/250 VDC System, Version 7.1
SOP 34SO-S22-001-2, 500 KV Substation Switching, Version 5.2
31EO-EOP-010-2S, RC RPV Control (Non-ATWS), Rev. 8, Attachment 1
31EO-EOP-012-2S, PC-1 Primary Containment Control, Rev. 4, Attachment 1
31EO-EOP-013-2S, PC-2 Primary Containment Control, Rev. 4, Attachment 1
31EO-EOP-014-2S, SC - Secondary Containment Control, Rev. 6, Attachment 1
31EO-EOP-016-2S, CP-2 RPV Flooding, Rev. 8, Attachment 1
Procedure 34AB-X43-001-2S, Rev.10ED3, "Fire Procedure," dated 5/28/03.
Calibration Procedure 57CP-CAL-097-2, Rosemount 1153 and 1154 transmitters, Revision No. 19.9.

Drawings

H-11814, Fire Hazards Analysis, Control Bldg. El. 130'-0", Rev. 5
H-11821, Fire Hazards Analysis, Turbine Bldg. El. 130'-0", Rev. 0
H-11846, Fire Hazards Analysis, Diesel Generator Bldg., Rev. 2
H-26014, R.H.R. System P&ID Sheet 1, Rev. 49
H-26015, R.H.R. System P&ID Sheet 2, Rev. 46
H-26018, Core Spray System P&ID, Rev. 29
B-10-1326, Rectangular Fire Damper Schedule, Rev. 2
B-10-1329, Rectangular Fire Damper, Rev. 1
H-11033, Fire Protection Pump House Layout, Rev. 47
H-11035, Fire Protection Piping and Instrumentation Diagram, Rev. 22
H-11226, Piping-Diesel Generator Building Drainage, Rev. 6
H-11814, Fire Hazards Analysis Drawing, Control Building, Rev. 5
H-11821, Fire Hazards Analysis Drawing, Turbine Building, Rev. 11
H-11846, Fire Hazards Analysis Drawing, Diesel Generator Building, Rev. 2
H-11894, Fire Detection Equipment Layout-Diesel Generator Building, Rev. 2
H-11915, Fire Detection Equipment Layout-Control Building, Rev. 2
H-13008, Conduit and Grounding, Fire Pump House, Rev. 9
H-13615, Wiring Diagram, Fire Pump House, Rev. 13
H-16054, Control Building HVAC System, Rev. 19
H-41509, Diesel Generator Building CO_2 System-P&ID, Rev. 5
H-43757, Penetration Seals-Type, Number, and as-Built Location, Rev. 3

Calculations, Analyses, and Evaluations

E. I. Hatch Nuclear Plant Units 1 and 2 Safe Shutdown Analysis Report, Rev. 20.
Edwin I. Hatch Nuclear Plant Fire Hazards Analysis and Fire Protection Program, Rev. 20
Calculation SMFP88-001, Hydraulic Analysis of Sprinkler Systems in Control Building East Cableway, dated 03/11/1988
Calculation SMNH94-046, FCF-F10B-006, Fire Resistance of Concrete Block at HNP, dated 09/30/1994
Calculation SMNH94-048, FCF-F10B-006, Cable Tray Combustible Loading Calculation, dated 09/30/1994

Calculation SMNH98-023, HT-98617, Fire Protection Penetration Seal Deviation Analysis, dated 10/28/1998
Calculation SMNH00-011, HT-00606, Hose Nozzle Pressure Drop Analysis, dated 09/08/2000
Evaluation HT-91722, Fire Protection Code Deviation Resolution, dated 04/22/1992
Hatch Response to NRC IN 1999-005, dated 05/04/1999
Hatch Response to NRC IN 2002-024, dated 09/20/2002
Calculation SENH 98-003, Rev. 0, plot K, protective relay settings 4kV bus 2E
Calculation 85082MP, Plot 29, 600V Switchgear 2C
Calculation SENH 94-004, Attachment A, Sheets 7&8, 600/208 Reactor Building MCC 2C
Calculation SENH 91-011, Attachment P, Sheet 6, Reactor Building DC MCC 2A
Calculation SENH 94-013, Sheets 28 and 29, 600V Reactor Building MCC 2E-B
Calculation SENH 91-011, Attachment P, Sheet 16, Reactor Building 250VDC MCC 2B

Audits and Self-Assessments

Audit No. 01-FP-1, Audit of the Fire Protection Program, dated April 12, 2001
Audit No. 02-FP-1, Audit of the Fire Protection Program, dated February 28, 2002
Audit No. 03-FP-1, Audit of Fire Protection, dated April 21, 2003
1999-001106, Lighting in Fire Equipment Building
2002-000629, Inordinate Number of Buried Piping Leaks
2002-002127, Inadequate Bunker Gear
2002-002129, Health Physics Support and Participation for Fire Brigade
2003-000735, Impact on Cold Weather on Operating Units
Audit Report 01-FP-1, Audit of Fire Protection Program, dated 04/12/2001
Audit Report 02-FP-1, Audit of Fire Protection Program, dated 02/28/2002
Audit Report 03-FP-1, Audit of Fire Protection Program, dated 04/21/2003

CRs Reviewed

CR 2000007119, Fire Procedure 34AB-X43-001-1S Needs to be Enhanced
CR 2001002032, Fire Procedure 34AB-X43-001-2S Needs Actions for Diesel Fuel Oil Pumps
CR 2003004377, Fire Procedure 34AB-X43-001-1 Enhancements
CR 2003004379, Fire Procedure 34AB-X43-001-2 Enhancements
CR 2003004382, SSAR Discrepancies

CRs Generated During this Inspection

CR 2003007129, No Fire Procedure Actions for a Fire in the 2C Switchgear Room
CR 2003007719, Use of Link Wrench
CR 2003007978, Fire Damper Corrective Action
CR 2003008141, Breaker Maintenance Handle
CR 2003008165, SSAR Section 2.100
CR 2003008179, Drywell Access Emergency Lights
CR 2003008181, Link Labeling
CR 2003008202, Manually Opening MOV 2E11-F015A
CR 2003008203, SRV Manual Action Steps in Fire Procedure
CR 2003008237, Emergency Lights and Component Labeling for Manual Actions

CR 2003008238, CO2 Migration Through Floor Drains
CR 2003800132, SSAR Error for Position of 2E11-F004A
CR 2003800151, Instruments for Manual Actions
CR 2003800152, Sliding Links in SSAR
CR 2003800153, Promat Test Report
CR 2003008250, Communications for Post-Fire SSD
CR 2003800166, Review Fire Procedure Step 34AB-X43-001-2 Steps to Verify Compliance with Appendix R.

Design Criteria and Standards

Design Philosophy for Fire Detectors at E. I. Hatch Nuclear Plants, Rev. 2

Completed Surveillance Procedures and Test Records

42SV-FPX-021-OS, Surveillance of Swinging Fire Doors, Task # 1-3367-1 (completed on 01/09/2003)
42SV-FPX-024-OS, Fire Hose Stations, Task # 1-3359-1 (completed on 06/27/2003)
42SV-FPX-030-OS, Fire Emergency Self Contained Breathing Apparatus Inspection and Test, Task # 1-4200-3 (completed on 07/07/2003)
42SV-FPX-032-OS, Automatic Sliding Fire Door Surveillance, Task # 1-3361-2 (completed on 08/13/2002
Promatec Technologies Installation Inspection Report for Fire Area 2104, MWO 2-98-00881, Record 09367-2289, dated 09/03/1998

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Dow Corning Fire Endurance Test on Penetration Seal Systems in Precast Concrete F Using Silicone Elastomers, dated 10/28/1975
Dow Corning 561 Silicone Transformer Fluid Technical Manual,10-453-97, dated 1997
S-80393, Mesker Instructions for Installing d&H “Pyromatic” Automatic Sliding Fire Door Closer
S-27874B, General Electric Instruction Book GEK-26501, Liquid-Filled Secondary Unit Substation Transformers, Rev. 2
S-52429A, Bisco, Fire Rated Penetration Seal Qualification Data, dated 08/16/1990
S-52480, Factory Mutual, Fire Rated Penetration Seal Qualification Data-Chemtrol Design FC-225, dated 08/31/1990
S-54875B, Promatec, Fire Barriers-Unit 2 East Cableway, Rev. 2
Omega Point Laboratories, SR90-005, Three Hour Wall Test, dated 06/06/1990
Promatec Technologies Inc., PSI-001, Issue 1, General Construction Details, dated 07/21/1998
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System Information Document No. SI-LP-01401-03, Main Steam and Low Low Set System, dated 4/3/2000

Applicable Codes and Standards

ANSI N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants
NFPA 12, Standard for Carbon Dioxide Systems, 1973 Edition.
NFPA 13, Standard for the Installation of Sprinkler Systems, 1976 Edition.
NFPA 14, Standard for the Installation of Standpipe and Hose Systems, 1974 Edition.
NFPA 20, Standard for the Installation of Centrifugal Fire Pumps, 1973 Edition.
NFPA 72D, Standard for the Installation, Maintenance, and Use of Proprietary Protection Signaling Systems, 1975 Edition.
NFPA 72E, Standard on Automatic Fire Detectors, 1974 Edition
NFPA 80, Standard on Fire Doors and Windows, 1975 Edition.
NUREG-1552, Supplement 1, Fire Barrier Penetration Seals in Nuclear Power Plants, dated January 1999
OSHA Standard 29 CFR 1910, Occupational Safety and Health Standards,
Underwriters Laboratory, Fire Resistance Directory, January 1998

Other Documents

Design Change Package 91-009, Retrofill Dielectric Fluid on Unit 2 Transformers, Rev. 1
Fire Protection Inspection Reports for the period 2001-2002
Fire Service Qualification Training, FP-LP-10003, Fire Fighter Safety, dated 01/14/2002
Fire Service Qualification Training, FP-LP-10004, Fire Fighter Personal Protective Equipment, dated 01/14/2002
Fire Service Qualification Training, FP-LP-10014, Fire Streams, dated 01/22/2002
Fire Service Qualification Training, FP-LP-10018, Fire Fighting Principles and Practices, dated 01/22/2002
Hatch Response to NRC Information Notice 1999-05, Inadvertent Discharge of Carbon Dioxide Fire Protection System and Gas Migration, dated 05/04/1999
Hatch Response to NRC Information Notice 2002-24, Potential Problems with Heat Collectors on Fire Protection Sprinklers, dated 09/20/2002
10CFR21-001, ELECTRAK Corporation, Software Error within TRAK2000 Cable Management and Appendix R Analysis System, dated 03/07/2003
U. S. Consumer Product Safety Commission, Invensys Building Systems Announce Recall of Siebe Actuators in Building Fire/Smoke Dampers, dated 10/02/2002
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Pre-fire Plan A-43966, Fire Area 2404, Diesel Generator Building Switchgear Room 2E, Rev. 2
Pre-fire Plan A-43966, Fire Area 2408, Diesel Generator Building Switchgear Room 2F, Rev. 2
Pre-fire Plan A-43965, Fire Area 2016, W 600V Switchgear Room 2C, Rev. 4

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Hatch UFSAR Section 3.4, Water Level Flood Design, Rev. 20
Hatch UFSAR Section 9.1-A, Fire Protection Plan, Rev. 18C
Hatch UFSAR Section 17.2, Quality Assurance During the Operations Phase, Rev. 20B
Hatch Fire Hazards Analysis, Appendix B, Fire Protection Equipment Operating and Surveillance Requirements, Rev. 12B

Hatch Fire Hazards Analysis, Appendix H, Application of National Fire Protection Association Codes, Rev. 12B
Hatch SER dated April 18, 1994
Safe Shutdown Analysis Report for E.I. Hatch Nuclear Plant Units 1 and 2, Rev. 26
Fire Hazards Analysis for E. I. Hatch Nuclear Plant Units 1 and 2, Rev.18C, dated 7/00.
NRC Safety Evaluation Report dated 01/02/1987; Re: Exemption from the requirements of Appendix R to 10 CFR Part 50 for Hatch Units 1 and 2 (response to letter dated May 16, 1986).
Letter dated 05/16/86, From L. T. Guewa (Georgia Power) to D. Muller, NRC/NRR; Re: Edwin I Hatch Nuclear Plant Units 1 and 2 10 CFR 50.48 and Appendix R Exemption Requests

Design Change Request Documents

DCR No. 91-134, SRV Backup Actuation via PT Signals, Revision 0.
Drawing No. H-26000, Nuclear Boiler System P&ID, Sheet 1, Revision 39
Drawing No. H-27403, Automatic Depressurization System 2B21C Elementary Diagram, Sheet 6 of 6, Revision 2
Drawing No. H-27472, Automatic Depressurization System 2B21C Elementary Diagram, Sheet 3 of 6, Revision 2
Drawing No. H-27473, Automatic Depressurization System 2B21C Elementary Diagram, Sheet 4 of 6, Revision 2
Drawing No. H-24427, Elementary Diagram, ATTS System 2A70 Sheet 27 of 35, Revision 3
Drawing No. H-24428, Elementary Diagram, ATTS System 2A70 Sheet 28 of 35, Revision 3
Drawing No. H-24429, Elementary Diagram, ATTS System 2A70 Sheet 29 of 35, Revision 5
Drawing No. H-24430, Elementary Diagram, ATTS System 2A70 Sheet 30 of 35, Revision 3
Drawing No. H-24431, Elementary Diagram, ATTS System 2A70 Sheet 31 of 35, Revision 3
Drawing No. H-24432, Elementary Diagram, ATTS System 2A70 Sheet 32 of 35, Revision 6

LIST OF ACRONYMS

ADS	Automatic Depressurization System
AOP	Abnormal Operating Procedure
APCSB	Auxiliary and Power Conversion System Branch
ATTS	Analog Transmitter Trip System
BTP	Branch Technical Position
CAP	Corrective Action Program
CO_2	Carbon Dioxide
CRs	Condition Reports
CST	Condensate Storage Tank
DCR	Design Change Request
ERFBS	Electrical Raceway Fire Barrier System
FHA	Fire Hazards Analysis
FPP	Fire Protection Program
HCTL	Heat Capacity Temperature Limit
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IP	Inspection Procedure
JPM	Job Performance Measure
LLS	Low-Low Set
LOCA	Loss of Coolant Accident
ma	Milli-amp
MOVs	Motor Operated Valves
NCV	Non-Cited Violations
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
OSHA	Occupational Safety and Health Administration
PT	Pressure Transmitter
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SCBAs	Self-Contained Breathing Apparatuses
SDP	Significance Determination Process
SERs	Safety Evaluation Reports
SRVs	Safety Relief Valves
SSAR	Safe Shutdown Analysis Report
SSD	Safe Shutdown
TS	Technical Specification
UFSAR	Updated Final Safety Evaluation Reports
URI	Unresolved Item
XLPE	Cross-Linked Polyethylene