(19)

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-321, 50-366
License Nos.:	DPR-57, NPF-5
Report No.:	05000321/200306 and 05000366/200306
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,
Approved by:	Charles R. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

KK-28

Enclosure

CONTENTS

SUMMARY OF FINDINGS

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

The report covered a two-week period of inspection by three regional inspectors and a contractor from Brookhaven National Laboratory. Three Green non-cited violations (NCVs) and four 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 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: Mitigating Systems

- URI. 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. In response to this potential issue, the licensee promptly moved the manual action step to the front of the Fire Procedure to enable operators to accomplish the action much sooner during a fire event. This finding was determined to have potential safety significance greater than very low significance because of the use of manual actions in lieu of physical protection as required by 10 CFR 50 Appendix R, Section III.G.2. (Section 1R05.05.b.1)

  URI. The team identified an unresolved item in that a fire in Fire Area 2104 could cause all eleven SRVs to open at a time when residual heat removal (RHR) system may not be available. To mitigate this event, the licensee's safe shutdown analysis report (SSAR) credits the use of Core Spray Loop A to provide reactor coolant makeup. 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 satisfies the shutdown performance goals specified in Appendix R, section L.1.e to 10CFR 50.

This finding is unresolved pending completion of the NRC review of a calculation of record which demonstrates the capability of the Core Spray system to mitigate the above event. This finding was determined to have potential safety significance greater than very low significance because of a lack of a calculation of record and documentation of the limited set of equipment that would be credited for safe shutdown under these conditions. (Section 1R.05.03.b)

URI: The team identified an unresolved item in that the licensee's current fire protection licensing basis characterizes the opening of terminal board links in control panels as a repair activity which is not permitted to achieve and maintain hot shutdown conditions. The licensee could not provide any evidence to justify why these actions were not characterized as a repair activity in its current SSAR. In response to this inspection finding , the licensee initiated a Condition Report (CR 2003800152, dated 7/24/03) to evaluate actions to open links, in order to determine if they are necessary to achieve hot shutdown, and if an exemption from Appendix R is required.

This finding is unresolved pending completion of a significance determination. This finding is greater than minor because it impacts the mitigating system cornerstone and has the potential for the operator not successfully completing the action because of adverse human factor conditions. (Section 1R.05.01.b)

URI: The team identified an unresolved item in connection with the implementation of design change request (DCR) 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 design input requirements in the design change package. Additionally, implementation of a two-out-of-two coincident taken twice logic , has introduced a potential common cause failure of all eleven SRVs because of fire induced damage to two 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 (HCTL) for the suppression pool. (Section 1R21.01.b)

- Green. The team identified a finding with very low safety significance in that a local manual operator action to operate safe shutdown equipment was too difficult and was also 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 judged that some operators would not be able to perform the action.

This finding involved a violation of 10 CFR 50, Appendix R, Section III.G.1 and Technical Specification 5.4.1. 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 licensee could have time to develop and implement cold shutdown repairs to facilitate accomplishment of the action, this finding did not have potential safety significance greater than very low safety significance. (Section 1R05.05.b.2)

- Green. The team identified a finding with very low safety significance 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, and without NRC approval.

  This finding involved a violation of 10 CFR 50, Appendix R, Section III.G.2. 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 1R05.05.b.3)

- Green. The team identified a finding with very low safety significance in that emergency lighting was not adequate for some manual operator actions that were needed to support post-fire operation of safe shutdown equipment.

  This finding involved a violation of 10 CFR 50, Appendix R, Section III.J. 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)

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.D, 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 Specifications (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 safe shutdown conditions in the event of fire in each of the selected fire areas. The objectives of this evaluation were as follows:

(a) Verify that the licensee's shutdown methodology has correctly identified the components and systems necessary to achieve and maintain a safe shutdown condition.

(b) Confirm the adequacy of the systems selected for reactivity control, reactor coolant makeup, reactor heat removal, process monitoring and support system functions.

(c) Verify that a safe shutdown can be achieved and maintained without off-site power, when it can be confirm[illegible] in any of the selected fire areas could cause th[illegible]

(d) Verify that local manual operator[illegible] fire protection licensing basis.

← State that one finding was identified & is discussed in Section 02.

b. ~~Issues and~~ Findings

Licensing Basis for Repair Activities (Opening/ Safe Shutdown Condition.

Introduction: The team identified a potential concern where the [illegible] actions to disconnect terminal board sliding links in order to isolate two 4-20 ma instrumentation control loop circuits in order to prevent the spurious actuation of eleven SRVs.

Description: The licensee has identified the systems required to perform the shutdown functions of reactor shutdown, over pressure protection, maintenance of coolant inventory, and decay heat removal for SSD paths 1 and 2. The reactor shutdown function is provided by the reactor protection system (RPS) for all paths.

Path 1 utilizes reactor core isolation cooling (RCIC), two group "A" SRVs, and the residual heat removal (RHR) system in the alternate shutdown cooling mode of operation to provide inventory makeup, decay heat removal, and depressurization. RCIC would be used until approximately 4 hours into the event, at which time the reactor pressure will be within the low-pressure coolant injection (LPCI) operability range (approximately 135 psig). To mitigate the impact of a spurious actuation of the automatic depressurization system (ADS) at a time when RHR system may not be available due to fire damage, the licensee has assured that Core Spray (CS) would be available.

Path 2 utilizes the High Pressure Coolant Injection (HPCI), two group "A" SRVs, and the RHR system in the alternate shutdown cooling mode of operation. The HPCI system and one SRV are utilized during the first 4 hours of a fire event to maintain the reactor water level and pressure within acceptable limits. After approximately 4 hours, the RHR system is started in the alternate shutdown cooling mode of operation.

For the fire areas evaluated, the licensee identified the structures, systems and components needed to achieve and maintain safe shutdown conditions in the event of fire. The team evaluated required manual operator actions in order to verify that they were consistent with the plant's fire protection licensing basis. Based on this evaluation

the team determined that the licensee relies on manual operator actions to open terminal board links as a means of preventing an undesired actuation of all eleven SRVs. (see section 1R21.01b).

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. Additionally, the licensee's current licensing basis documents (Reference: Georgia Power request for exemption dated May 16, 1986 and a subsequent 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 Section III.G of Appendix R.

Enforcement: The licensee's current licensing basis documents (Reference: Georgia Power request for exemption dated May 16, 1986 and a subsequent 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 Section III.G of Appendix R. Based on these documents 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 are not characterized as a repair activity in its current SSAR. In response to this inspection finding , the licensee initiated a Condition Report (CR 2003800152, dated 7/24/03) to evaluate actions to open links, in order to determine if they are necessary to achieve hot shutdown, and if an exemption from Appendix R is required. This issue is combined with URI 50-366/03-06-02, Untimely and Unapproved Manual Operator Action for Post Fire Safe Shutdown discussed in section 1R05.05.b.1of the report. The licensing basis concerns will be disposition upon review and acceptance of additional licensing basis documentation which demonstrates that actions necessary to open links should not be considered a repair necessary to achieve and maintain hot shutdown conditions.

move to Page 8 before analysis.

.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 safe shutdown (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 CAR 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 fire protection program (FPP) were satisfied:

- Updated Final Safety Analysis Report (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

pressure within acceptable limits. After approximately 4 hours, the RHR system is started in the alternate shutdown cooling mode of operation.

For the fire areas evaluated, the licensee identified the structures, systems and components needed to achieve and maintain safe shutdown conditions in the event of fire. The team evaluated required manual operator actions in order to verify that they were consistent with the plant's fire protection licensing basis. Based on this evaluation the team determined that the licensee relies on manual operator actions to open terminal board links as a means of preventing an undesired actuation of all eleven SRVs. (see section 1R21.01).

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. Additionally, human factors problems created the potential for the operator to not successfully complete the task. The above concerns along with the fact that the opening of terminal board links are considered "repairs" causes this finding to have potential safety significance greater than low safety significance.

Enforcement: The licensee's current licensing basis documents (Reference: Georgia Power request for exemption dated May 16, 1986 and a subsequent 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 Section III.G of Appendix R. Based on these documents 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 are not characterized as a repair activity in its current SSAR. In response to this inspection finding , the licensee initiated a Condition Report (CR 2003800152, dated 7/24/03) to evaluate actions to open links, in order to determine if they are necessary to achieve hot shutdown, and if an exemption from Appendix R is required. This issue is identified as URI 50-366/03-06-01, Licensing Basis for Repair Activities (Opening/Closing of Links) to Achieve Safe Shutdown Conditions. This item remains open pending review and acceptance of additional licensing basis documentation which demonstrates that actions necessary to open links should not be considered a repair necessary to achieve and maintain hot shutdown conditions.

.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 safe shutdown (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 CAR 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 fire protection program (FPP) were satisfied:

- Updated Final Safety Analysis Report (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

10 CFR 50.48, "Fire Protection," and Appendix R to 10 CFR 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979" establish specific fire protection features required to satisfy General Design Criterion 3, "Fire Protection" (GDC 3, Appendix A to 10 CFR 50). Section III.G of Appendix R requires fire protection features be provided for equipment important to safe shutdown. An acceptable level of fire protection may be achieved by various combinations of fire protection features (barriers, fire suppression systems, fire detectors, and spatial separation of safety trains) delineated in Section III.G.2. For areas of the plant where compliance with the technical requirements of Section III.G.2 can not be achieved, licensees must either seek an exemption from the specific requirement(s) or provide an alternative shutdown capability in accordance with Sections III.G.3 and III.L of the regulation.

For each selected fire area, the results of the licensee's analysis for compliance with Section III.G of Appendix R is documented in a SSAR . The overall approach of these evaluations was to determine the fire-induced losses for a fire in each fire area and then assess the plant impact given those loses.

On a sample basis, an evaluation was performed to verify that 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 RCIC and 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.

b. Findings

Capability of Equipment Credited in SSAR to Mitigate the Spurious Actuation of the SRVs

Introduction: The licensee's evaluation of th
2.100 of the SSAR) states that a fire in this a
automatic depressurization system (ADS) at
ADS includes seven of the eleven SRVs. To
use of Core Spray Loop A for a fire in this are
(which involves seven SRVs), the SSAR also
cause all eleven SRVs to spuriously actuate a
are located in close proximity in this area.

Description: The SSAR states that a fire in Fire Area 2104 could cause all eleven SRVs to spuriously actuate as a result of fire damage to two cables that are located in close proximity in this area. The specific circuits that could cause this event have been identified by the licensee (circuit nos.: ABE019C08 and ABE019C09). Each of these two circuits provides a 4 to 20 milliamp instrumentation signal from SRV high-pressure actuation transmitters (2B21-N127B and 2B21-N127D) to master trip units 2B21-N697B and 2B21-N697D, respectively. The purpose of this circuitry is 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 trip units which would cause all eleven SRVs to actuate (open). The pressure signal from each transmitter is conveyed to its respective trip unit via a two-conductor, instrument cable that is routed through this fire area (two separate cables). Each cable consists of a single twisted pair of insulated conductors, an uninsulated drain wire that is wound around the twisted pair of conductors, and a foil shield. In Fire Area 2104 the two cables are located in close proximity, in the same cable tray. Actuation of the SRV electrical backup is completely "blind" to the operators. 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). Since the operators typically would not initiate a manual scram until fire damage significantly interfered with control of the plant, its possible that all eleven SRVs could open at 100% power, prior to scramming the reactor. This scenario 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 milliamp instrument circuit may not be necessary to initiate an undesired (false high) signal. For cables that transmit low-level instrument signals, any degradation of the insulation of the individual twisted conductors due to fire damage may be sufficient to cause leakage currents to be generated between the two conductors. Such leakage current would appear as a false high pressure signal to the trip units. If both cables were damaged as a result of fire, false signals gen
in each cable, would actuate the SRV electrical ba
eleven of the SRVs to open. The conductor insulat
is cross-linked polyethylene (XLPE). Since both ca
exposed to the same heating rate, there is a reaso
shorting in the two cables at approximately the san

The licensee's SSAR recognizes the potential safe
describes methods that have been developed to p

its impact on the plant's post-fire safe shutdown capability should it occur. To prevent this scenario, the licensee has developed procedural guidance which directs operators to open link BB-10 in panel 2H11-P927 and link BB-10 in panel 2H11-P928. Opening of these links would prevent actuation of the SRV trip units by removing the 4 to 20 milliamp signal fed by the pressure transmitters. In the event the SRVs were to open prior to operators completing this action, the SSAR credits Core Spray loop A to mitigate the event. However, the inspection team had several concerns regarding the effectiveness of the licensee's approach. Specific concerns identified by the team included:

1. The timing of operator actions necessary to prevent the event (the time from fire detection to the time the two links would be opened);

2. Whether the operator actions (opening of links) were consistent ... plant's current fire protection licensing ... needed to achieve and maintain hot st

3. The capability of the limited set of syste ... SSAR for accomplishing post-fire safe ... of fire in Fire Area 2104 to mitigate the ... the shutdown performance goals specif...

4. Defeat having 2 SRV

With regard to the timing of operator actions to prevent ... SRVs to open, during the inspection the licensee perfo... estimated that approximately thirty minutes would pass ... the time an operator would implement procedural actions to prevent its occurrence (opening of links). The licensee concurred with the insp... team's concern that this time (30 minutes) may be too long to provide an effectiv... actuation. To improve the effectiveness of this action th... its existing procedures so that the action would be take... confirmation of fire in areas where the spurious actuatic...

Analysis: This finding is greater than minor because i... reliability objectives and the equipment performance at... cornerstone. In order to achieve safe shutdown conditi... areas chosen for review, manual control of two SRVs i... 2B21-F013B and 2B21-F013F are required to remain manually operable... for Path 2 SRVs 2B21-F013D and 2B21-F013G are required to remain manually operable. These actions are necessary to ensure that the suppression pool temperature will not exceed the heat capacity temperature limit (HCT... for the suppression pool. One SRV (per Path) is opened to manually control dep... and a half hours after event initiation in order to maintai... HCTL. The second SRV is opened approximately four ... alternate shutdown cooling mode of operation. The ina... manually control the listed SRVs because of spurious a... have potential safety significance greater than low safe...

I think you have 4 concerns

Simula.

need to add stuff here on 2) licensing basis 3) Capability of the limited set of syste... 4)

This problem is not discussed in the 3 problems above. ?

Enforcement: 10 CFR 50 Appendix R, section L.1.e states that during the post fire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal AC power. Because there is a potential for all SRVs to spuriously actuate as a result of fire in Fire Area 2104 at a available, the SSAR credits the use of Core Spray Loop A to coolant makeup function. During the inspection, on 7/24/03, performed a simulator exercise of an event which caused all this exercise, simulator RPV level instruments indicated that capable of maintaining level above the top of active fuel. Ho provide any objective evidence (e.g., specific calculation or demonstrated that, assuming worst-case fire damage in Fire of equipment available would be capable of mitigating the e satisfies the shutdown performance goals specified in Appe 10CFR 50. Pending review and acceptance of objective ev this capability, this issue is identified as URI 50-366/03-06-02, Capability of Equipment Credited in the SSAR to Mitigate the Spurious Actuation of Eleven SRVs.

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 safe shutdown from outside of the control room. 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 Appendix R safe shutdown analysis (SSA); (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 provide 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 Safe Shutdown

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 would be considered a large loss of coolant accident (LOCA), and that a LOCA must be prevented from occurring during a fire event. 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 event, 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. However, NRC fire models indicated that fires could potentially cause damage to cables in as little as about 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, so that 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 poor so 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 instrument cables, for reactor pressure signals, would be needed to spuriously open all eleven SRVs. Since the licensee stated that the two cables were in the same cable tray in fire area 2104, the Unit 2 east cableway, the team considered that a fire in that area could potentially cause all eleven SRVs to spuriously open (see section 1R21.01).

In response to this potential 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.

Analysis: The team determined that this potential issue is related to associated circuits. As described in NRC Inspection Procedure (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 for a fire in fire area 2104. (see section 1R21.01).

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 manual operator actions to prevent the spurious opening of all eleven SRVs. Licensee personnel contended that fire damage to two cables was outside of the Hatch licensing basis and consequently that there was no requi
However, the licensee could provide no eviden

This potential issue will remain unresolved pend
~~associated circuits and~~ completion of a significa
identified as URI 50-366/03-06(03), Untimely ar
for Fire Safe Shutdown.

Why are we waiting on associated circuits guidance.

2. Local Manual Operator Action was Too Difficult and 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 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 were relied on 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. He 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 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; required wearing of 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 or 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 unsafe and that some operators would not be able to perform it.

The licensee had no operator training job performance measure (JPM) for performing this action and could not demonstrate that all operators could perform the 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 and was 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

Where is the SDP logic

What makes this a green finding

cornerstone. Since the licensee could have time t
shutdown repairs to facilitate accomplishment of t
potential safety significance greater than very low

Enforcement: 10 CFR 50, Appendix R, Section II
features shall be provided for systems important
of limiting fire damage so that systems necessar
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 Fire Protection Program 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 BWR 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

Can't the licensee implement manual actions for cold S/D repair. This implies → No physical protection is a problem?

Contrary to the above, the licensee had not provided
damage, for electrical operation valve 2E11-F015A
procedure for repairing any related fire damage with
relied on local manual operator actions, as describe
AOP 34AB-X43-001-2. However, those procedure s
operators would not be able to perform them becaus
difficult and also were unsafe. In response to this is
203008202. Because the identified inadequate ope
significance and the issue has been entered into the
program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC's Enforcement Policy: NCV 50-366/03-06-03, Local Manual Operator Action for Post-Fire Safe Shutdown Equipment was Too Difficult and Unsafe.

3. Unapproved Manual Operator Actions for Post-Fire Safe Shutdown

Introduction: A finding of very low safety significance was identified in that the licensee relied on some manual operator actions to operate SSD equipment, instead of providing the required physical protection of cables from fire damage, and without NRC approval. 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 included:

Smaller

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 [illegible] CLOSED position."

The team walked down these actions and judged tl
accomplished by operators in a timely manner. In
these operator actions were being used instead of
damage that could cause a loss of station service b
runaway.

Modify to include safe reference to the criteria of the imp. procedure.

Analysis: The finding is greater than minor becaus
reliability objectives and the equipment performanc
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 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.

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 CR2003800166 dated 7/25/2003. 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.1 of the NRC's Enforcement Policy: NCV 50-366/03-06-05, 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 safe shutdown activities. The team walked down portions of the safe shutdown 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 Safe Shutdown 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 to 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 1R05.05.b.1). The other exception was the action to open 2E11-F015A, which was judged to be too difficult (see section 1R05.05.b.2). For all 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 have potential safety significance greater than very low safety significance.

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 safe shutdown 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.1 of the NRC's Enforcement Policy: NCV 50-366/03-06-06, Inadequate Emergency Lighting for Operation of 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 (GL) 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 (HVAC) 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

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 safely shutdown.

The team reviewed the adequacy of the design and installation of the manual 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 safe shutdown 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 fire protection program 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 safe shutdown 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 approved fire protection program. The team also reviewed Corrective Action Program Condition Reports 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. DCR 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 and its associated 10 CFR 50.59 Evaluation. 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

Inadequate Plant Modification Results in Common Cause Failure of Safety Relief Valves (SRVs).

Introduction

Design Change Request (DCR) 91-134 was implem
raised in General Electric Report NEDC-3200P, "E
January-February 1991 Turbine Trip Events for Pla
ensure that individual SRV(s) will actuate at or neai
allowable limits, a backup mode of operation was ir
design was intended to mitigate the effects of corrc
Target Rock SRVs.

1 or 2 sentences to describe what the problem is. Something like ... Inadequate plant mod. and 50.59 review

Description:

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 sw[illegible] the pilot valve of the individual SRV and causes it to go open. When the [illegible] position the SRV is opened upon receipt of either an Autc (ADS), or Low-Low Set (LLS) control logic signal. Either the valve. DCR 91-134 provided a backup mode for initia valve solenoid, which was independent of ADS or LLS lo no operator action to initiate opening of the SRVs and wa loop to the operators.

Explain again what is meant by a third control loop

The scope of the plant modification involved the installati transmitters (Model No. 1154GP9RJ), 0-3000 psig, in th instrument racks at EL. 158 of the Reactor Building. Eac[illegible] part of a 4-20 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 D	1140 psig

Pressure transmitters 2B21-N127A and 2B21-N1[illegible] were wired to ATTS cabinets 2H11-P927. Pressure transmitter 2B21-N127A ir of a trip unit master relay K308C and trip unit sla loop components for pressure transmitter 2B21-N relay K335C in addition to trip unit slave relays K instrument loops constituted a "Division" pressure intended to provide the one-out of two logic signa backup actuation. The design objective of having assure compliance with the single failure criterior

include what GDC. How is this applicable.

Additionally, pressure transmitters 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 pressure transmitter 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 was 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 the single failure criterion of 10 CFR 50 Appendix A.

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 total of 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 via the pressure tran...

Analysis: The licensee in their SSAR takes cr... in order to achieve and maintain safe shutdow... SRVs are required for a fire in the fire areas s...

The team performed a circuit analysis of SRV ... F013G (Path 2) in order to verify that the desig... of-two taken twice logic had been achieved. B... that the design objective of implementing a one... been installed for the SRVs. The logic installed ... coincident taken twice logic in addition to a one-out-of-two coincident taken twice logic. The team also determined that the two-out-of-two coincident logic input from trip unit master relays K310D and K335D represented a common cause failure for both SRVs for a fire in fire area 2104. Specifically, cable ABE019C08 associated with pressure transmitter 2B21-N127B current loop, and cable ABE019C09 associated with pressure transmitter 2B21-N127D current loop, are both routed in the same cable tray in fire area 2104. Both shielded twisted pair instrument cables are unprotected from the effects of a fire in this fire area. Fire induced insulation damage to both cables could result in leakage currents which causes the instrument loops to fail high. This failure mode simulates a high nuclear boiler pressure condition which would initiate SRV backup actuation of SRV 2B21-F013F and 2B21-F013G. Spurious actuation of both SRVs for a fire in fire area 2104 defeats the capability to manually control these SRVs as is required per the SSAR.

Enforcement 10 CFR 50, Appendix B, ... measures shall provide for verifying or ch... industry standard, ANSI N45.2.11-1974, ... for relating the final design back to the so...

The logic implemented by the licensee for... design input requirements. The plant inst...

out-of-two taken twice logic that was specified for the SRV backup actuation via pressure transmitter signals design change package. This failure has created a condition where fire induced failures of two instrument circuit cables, (within close proximity to each other), could result in spurious actuation of all eleven SRVs based on the logic input from trip master unit relays K310D, and K335D and their associated trip unit slave relays. The 10 CFR 50.59 Evaluation performed for the plant modification failed to identify this failure mode. Additionally, the 10 CFR 50.59 Evaluation was inadequate in that it did not provide an adequate technical basis that an Unreviewed Safety Question (USQ) had not been created by implementation of the plant modification. Pending additional review by the NRC, this item is identified as URI 50-366/03-06-07, Implementation of DCR 91-134 Results in Spurious Actuation of Eleven SRVs because of Fire Induced Faults.

This inspection finding may be a " Potentially Generic Issue" by having implications for other licensees who have implemented a plant modification similar to DCR 91-134 for a BWR having a Mark 1 containment.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of licensee audits, self-assessments, and condition reports (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 (OE) 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 team presented the inspection results to Mr. R. Dedrickson, Assistant General Manager, and other members of your staff at the conclusion of the inspection on July 25, 2003 The licensee acknowledged the findings presented. Proprietary information is not included in the inspection report.

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
B. Duval	Chemistry Superintendent
R. Dedrickson	Assistant General Manager for Plant hatch
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
J. Payne	Senior Engineer, Corrective Action Program
D. Parker	Senior Engineer, Electrical
J. Rathod	Bechtel Engineering Group Supervisor
K. Rosanski	Oglethorpe Power Corporation Resident Manager
M. Raybon	Summer Intern
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

Attachment

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-366/03-06-01,	URI	Licensing Basis for Repair Activities (Opening/Closing of Links) to Achieve Safe Shutdown Conditions (Section 1R05.01.b)
50-366/03-06-02,	URI	Capability of Equipment Credited in the SSAR to Mitigate the Spurious Actuation of Eleven SRVs (Section 1R05.03.b)
50-366/03-06-03	URI	Untimely and Unapproved Manual Operator Action for Post-Fire Safe Shutdown (Section 1R05.05.b.1)
50-366/03-06-07,	URI	Implementation of DCR 91-134 Results in Spurious Actuation of Eleven SRVs because of Fire Induced Faults (Section 1R21.01.b)

Opened and Closed

50-366/03-06-04	NCV	Local Manual Operator Action for Post-Fire Safe Shutdown Equipment was Too Difficult and Unsafe (Section 1R05.05.b.2)
50-366/03-06-05	NCV	Unapproved Manual Operator Actions for Post-Fire Safe Shutdown (Section 1R05.05.b.3)
50-366/03-06-06	NCV	Inadequate Emergency Lighting for Operation of Post-Fire Safe Shutdown Equipment (Section 1R05.07.b)

Discussed

None

Attachment

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

Pre-fire Plan A-43965, Power-Block Areas Methodology, Rev. 0

Pre-fire Plan A-43965, Power-Block Areas Methodology, Rev. 0

Pre-fire Plan A-43965, Power-Block Areas Methodology, Rev. 0

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 Pressure Transmitter 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

Attachment