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by nrcgwia.nrc.gov; Thu, 31 Jul 2003 15:57:13 -0400
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by smtp-gateway ESMTPœ id h6VJqNpO028196
for <CFS1@nrc.gov>; Thu, 31 Jul 2003 15:52:25 -0400 (EDT)
Received: from exchange02.bnl.gov ([130.199.74.18])
by smtpgw2.bnl.gov with esmtp (Exim 3.36 #1)
id 19iJTw-0002RJ-00
for <CFS1@nrc.gov>; Thu, 31 Jul 2003 15:53:00 -0400
Received: by exchange02.bnl.gov with Internet Mail Service (5.5.2656.59)
id <QAMS3N2Z>; Thu, 31 Jul 2003 15:43:28 -0400
Message-ID: <1655641A03CED411BDCB0002B30945240857309A@exchange02.bnl.gov>
From: "Sullivan, Kenneth" <ks@bnl.gov>
To: "'Caswell Smith'" <CFS1@nrc.gov>
Cc: "Higgins, James C" <higgins@bnl.gov>
Subject: BNL Inspection Report Input - Hatch
Date: Thu, 31 Jul 2003 15:43:24 -0400
Return-Receipt-To: "Sullivan, Kenneth" <ks@bnl.gov>
MIME-Version: 1.0
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Caswell -

Attached file contains the BNL inspection report input for Hatch - A "formal" / "final" hardcopy to follow via regular mail. -- Let me know if you have any questions/comments.

P.S. -- Thanks for the kind words in your de-brief notes to management. It was a pleasure to work with you and the other team members again.

Thanks-

Ken Sullivan
(631)344-7915

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T-16

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filename="BNLhatch03RPT.wpd"
Click to view Base64 Encoded File

Energy Sciences and Technology Department

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July 31, 2003

Mr. Caswell Smith
U.S. Nuclear Regulatory Commission
Region II
61 Forsyth Street SW
Suite 23T85
Atlanta, GA 30303-8931

Reference: Triennial Fire Protection Inspection, Edwin I. Hatch Nuclear Plant; NRC Inspection Report Number: 50-321,366/03-06.

Dear Mr. Smith:

The enclosed technical letter report (TLR) describes the results of my activities during the Triennial Fire Protection Inspection performed at the Edwin I. Hatch Nuclear Plant. As described in the Inspection Plan dated June 30, 2003, my evaluation focused on a review of the systems required to achieve and maintain safe shutdown and the adequacy of separation provided for power, control and instrumentation cables of required shutdown equipment, as documented in the licensee's safe shutdown analysis report, implementing procedures and supporting documents. This inspection focused on the following four fire areas of Unit 2: 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; and Fire Area 2408, Switchgear Room 2F, Diesel Generator Building, Elevation 130 feet.

As described in the attached report, two unresolved findings resulted from this review. These include: (1) the licensing basis for repair activities (opening /closing of links) needed to achieve and maintain hot shutdown conditions; and, (2) objective evidence (e.g., calculation or analysis) which demonstrates the capability of equipment credited in the Safe Shutdown Analysis Report to mitigate the spurious actuation of all (11) SRVs in a manner that satisfies the safe shutdown performance goals specified in Appendix R to 10 CFR 50.

It was a pleasure to work with you and other members of the inspection team. Please do not hesitate to contact me at 631-344-7915 if you have any additional questions.

Sincerely,

Kenneth Sullivan,
Nuclear Energy & Infrastructure System Division
Energy Sciences & Technology Department

cc: J. Higgins
D. Norkin (NRC)

D. Diamond, w/o attachment
W. Horak, w/o attachment

BROOKHAVEN NATIONAL LABORATORY
Energy Sciences & Technology Department

Report Input to U.S. Nuclear Regulatory Commission
Region II

Fire Protection Baseline Inspection
of
Edwin I. Hatch Nuclear Plant
Units 1&2
(JCN: J-2843 Task 16)

NRC Inspection Report No:50-321,366/2003-06

Licensee:	Southern Company	
Facility:	Edwin I. Hatch Nuclear Plant, Units 1&2	
Inspection Conducted:	July 7 - 11, 2003 and July 21 - 25, 2003	
NRC Inspectors:	C. Smith	Region II (Team Leader)
	J. Wiseman	Region II
	R. Schin	Region II

BNL Technical Specialist:

K. Sullivan Date
(Electrical Systems)

Objective / Scope of Review

The objective of this triennial fire protection inspection was to perform a risk-informed inspection of defense-in-depth mitigating elements provided to ensure the successful accomplishment of safe shutdown conditions in the event of fire at the Hatch Nuclear Plant. The scope of this review included an evaluation of plant-specific design features, systems, equipment and operating procedures. In accordance with NRC Inspection Module 71111.05, dated 3/06/03, the evaluation did not include a comprehensive review of the potential impact of fire-induced failures in associated circuits of concern to post-fire safe shutdown.

From a review of licensee documents (e.g., Individual Plant Examination of External Events, Safe Shutdown Analysis Report, and the Fire Hazards Analysis), and observations noted during observations of facility conditions (i.e., plant walk-downs), the inspection team determined that a fire in the following fire areas presents a significant contribution to overall plant risk and conditional core damage probability:

1. Fire Area 2016, West 600 V Switchgear Room, Control Building, Elevation 130 feet. A fire in this area would involve shutdown from the Main Control Room using Safe Shutdown Path 2.

2. Fire Area 2104, East Cableway, Turbine Building, Elevation 130 feet. A fire in this area would involve shutdown from the Main Control Room using Safe Shutdown Path 1.

3. Fire Area 2404, Switchgear Room 2E, Diesel Generator Building, Elevation 130 feet. A fire in this area would involve shutdown from the Main Control Room using Safe Shutdown Path 2.

4. Fire Area 2408, Switchgear Room 2F, Diesel Generator Building, Elevation 130 feet. A fire in this area would involve shutdown from the Main Control Room using Safe Shutdown Path 2.

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

a. <u>Inspection Scope</u>

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. As described in the Inspection Plan, the objectives of this evaluation were to:

(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 confirmed that a postulated fire in any of the selected Fire Areas could cause the loss of off-site power.

(d) Verify that local manual operator actions are consistent with the plant's fire protection licensing basis.

b. Issues and Findings

The licensee's SSAR is based on assuring that a minimum set of systems and equipment, that are capable of satisfying the shutdown performance goals of Appendix R would be available in the event of fire in any plant location (fire area). This minimum set of systems and equipment is referred to in the licensee's SSAR as a shutdown path. Three specific paths for safe shutdown of the plant were developed. Paths 1 or 2 would be used in the event of fire in areas that meet the fire protection requirements of Appendix R Section III.G.2. Path 3 is an alternative shutdown capability and is used in the event of a significant fire in the control room, computer room, or cable spreading room which forces operators to abandon the main control room due to fire damage or environmental (i.e., control room habitability) concerns. Remote shutdown panels would be utilized for Path 3 shutdown. Since none of the fire areas selected for review during this inspection required this capability, shutdown Path 3, was not reviewed during this inspection.

Systems required to perform the shutdown functions of reactor shutdown, overpressure protection, maintenance of coolant inventory, and decay heat removal have been identified for each path. The reactor shutdown function is provided by the reactor protection system (RPS) for all paths.
Path 1 utilizes reactor core isolation cooling (RCIC), SRVs, and the RHR system in the alternate shutdown cooling mode 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 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), 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 appears to have properly identified the structures, systems and components needed to achieve and maintain safe shutdown conditions in the event of fire. However, as stated in the Inspection Plan, one objective of this inspection was to verify that manual operator actions are consistent with the plant's fire protection licensing basis. As described in Section 3 below, the licensee relies on manual operator actions to open terminal board links as a means of preventing an undesired actuation of all 11 SRVs. However, the inspection team noted that 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) characterize the opening of links as a repair activity that is not permitted as a means of complying with Section III.G of Appendix R. From these documents it appears that although the opening of links was considered a repair by both the licensee and the 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 observation, the licensee initiated a Condition Report (CR 2003800152, dated 7/24/03) 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 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, it is recommended that this finding remain unresolved.

2. Fire Protection of Safe Shutdown Systems

(Reviewed by other inspection team members)

3. 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 Safe Shutdown Analysis Report (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 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

The licensee's evaluation of the effects of fire in Fire Area 2104 (Section 2.100 of the SSAR) states that a fire in this area could cause a spurious actuation of the automatic depressurization system (ADS) at a time when RHR may not be available. ADS includes seven of the eleven SRVs. To mitigate this event, the SSAR credits the use of Core Spray Loop A for a fire in this area.

In addition to a spurious ADS actuation (which involves 7 SRVs), the SSAR also states that a fire in Fire Area 2104 could cause all 11 SRVs 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 11 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 11 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 generated as a result of leakage current in each cable, would actuate the SRV electrical backup scheme which would cause all eleven of the SRVs to open. The conductor insulation and jacket material of each cable is cross-linked polyethylene (XLPE). Since both cables are in the same tray and exposed to the same heating rate, there is a reasonable likelihood of each twisted pair shorting in the two cables at approximately the same time.

The licensee's SSAR recognizes the potential safety significance of this event and describes methods that have been developed to prevent its occurrence and/or mitigate 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 BNL Technical Specialist include:

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 with the plants current fire protection licensing basis with respect to repairs needed to achieve and maintain hot shutdown conditions (see Section 2 above); and,

3. The capability of the limited set of systems and equipment credited in the SSAR for accomplishing post-fire safe shutdown conditions in the event of fire in Fire Area 2104 to mitigate the event in a manner that satisfies the shutdown performance goals specified in Appendix R to 10 CFR 50.

With regard to the timing of operator actions to prevent fire damage from causing all SRVs to open, during the inspection the licensee performed an evaluation which estimated that approximately thirty minutes would pass from the time of fire detection to the time an operator would implement procedural actions to prevent its occurrence (opening of links). The licensee concurred with the inspection team's concern that this time (30 minutes) may be too long to provide an effective means of preventing the actuation. To improve the effectiveness of this action 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. (Note: a detailed review of this concern and the adequacy of the licensee's corrective action was performed by the Mechanical Systems reviewer).

In the event of fire in Fire Area 2104 (East Cableway), RHR Loop A may not be immediately available. Since there is a potential for all SRVs were to spuriously actuate as a result of fire in this area 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, on 7/24/03, the licensee voluntarily 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 could not provide any objective evidence (e.g., specific calculation or analysis) which demonstrates 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 to 10CFR 50. Pending review and acceptance of objective evidence which demonstrates this capability, it is recommended that this finding remain unresolved.

List of Persons Contacted During the Inspection

D. Parker	Southern Company	Engineering
C. Tully	Southern Company	Licensing
M. Dean	Southern Company	Engineering

List of Documents and Drawings Reviewed during Inspection

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.
Procedure 34AB-X43-001-2S, Rev.10ED3, "Fire Procedure," dated 5/28/03.
Hatch Plant P&IDs for the HPCI, RCIC, CS and RHR systems.
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
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

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