(21)

① Paragraph numbering of IR05 does not match the module

August xx, 2003

Header

SUBJECT: EDWIN I. HATCH NUCLEAR POWER PLANT - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT ~~50-321/03-06~~ AND ~~50-366/03-06~~

~~05000321/2003006~~ ~~05000366/2003006~~

Dear Mr. Sumner:

On July 25, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Hatch Nuclear Plant Units 1 and 2. The enclosed inspection report documents the inspection findings, which were discussed on that date with Mr. R. Dedrickson and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

indeterminate procedural

This report documents four findings that have potential safety significance greater than very low significance, however a safety significance determination has not been completed. One issue involving a did present an immediate safety concern ~~and condition report (CR) 2003008203 was written to revise the Fire Protection procedure to resolve the concern identified by the inspectors. The procedure was revised prior to the end of the inspection~~. The other three issues did not present an immediate safety concern.

In addition, the report documents three NRC-identified findings of very low safety significance (Green), all of which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these three findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Hatch Nuclear Power Plant.

however, you staff revised the procedure prior to the end of the inspection

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 50-321, 366/03-06

U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-321, 50-366
License Nos.:	DPR-57, NPF-5
Report No.:	~~50-321/03-06 and 50-366/03-06~~
Licensee:	Southern Nuclear Operating Company
Facility:	E. I. Hatch Nuclear Plant
Location:	P. O. Box 2010 Baxley, GA. 31513
Dates:	July 7-11, 2003 (Week 1) July 21-25, 2003 (Week 2)
Inspectors:	C. Smith, P.E., Senior Reactor Inspector, (Lead Inspector) R. Schin, Senior Reactor Inspector G. Wiseman, Fire Protection Inspector K. Sullivan, Consultant, Brookhaven National Laboratory
Accompanying Personnel:	S. Belcher, Nuclear Safety Intern, (Week 1) ~~Region II~~
Approved by:	Charles R. Ogle, Chief Engineering Branch 1 Division of Reactor Safety

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REPORT DETAILS

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

1R05 FIRE PROTECTION

The purpose 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. 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. The inspection was performed in accordance with the new Nuclear Regulatory Commission (NRC) reactor oversight process using a risk informed approach for selecting the fire areas and attributes to be inspected. The team used Plant Hatch Individual Plant Examination of External Events, to choose several risk significant areas for detailed inspection and review. The fire areas chosen for review during this inspection were:

Use boiler plate I~ NA IR.

- 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.

From a review of licensee documents and observations noted during observations of facility conditions (i.e., plant walk-downs), the inspection team determined that a fire in the selected fire areas presented a significant contribution to overall plant risk and conditional core damage probability.

Documents reviewed by the team are listed in the attachment.

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

No bold

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:

Use the boiler plate in NA 03-06 inspection report

(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

Licensing Basis for Repair Activities (Opening/Closing of Links) Performed to Achieve Safe Shutdown Condition.

Introduction: The licensee's SSAR is based on assuring that a minimum set of systems and equipment, that are capable of satisfying the
of Appendix R, would be available in the event of fire in any
This minimum set of systems and equipment is referred to
shutdown path. Three specific paths for safe shutdown of
Paths 1 or 2 would be used in the event of fire in areas tha
requirements of Appendix R Section III.G.2. Path 3 is an a
and is used in the event of a significant fire in the control r
cable spreading room which forces operators to abandon t
fire damage or environmental (i.e., control room habitabilit
shutdown panels would be utilized for Path 3 shutdown. N
for review during this inspection required this capability, ar
reviewed during this inspection.

Description: Systems required to perform the shutdown functions of reactor shutdown, over pressure 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
removal (RHR) system in the alternate shutdown cooling m
inventory makeup, decay heat removal, and depressurizati
approximately 4 hours into the event, at which time the rea
the low-pressure coolant injection (LPCI) operability range
mitigate the impact of a spurious actuation of the automatic
(ADS) at a time when RHR system may not be available du
licensee has assured that Core Spray (CS) would be availa

Path 2 utilizes the High Pressure Coolant Injection (HPCI), S
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 ... SRVs. (see section 1R21.01).

Analysis: This finding is greater than min... reliability objectives and the equipment pe... cornerstone. Additionally, human factors ... to not successfully complete the task. The ... opening of terminal board links are conside... potential safety significance greater than lo...

The team determined that the manual actions are not timely & reasonably accomplishable. Hence may have a safety significance greater than very low significance.

Licensing basis.

Enforcement: The licensee's current lice... Power request for exemption dated May 16 ... 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 consi... licensee and the NRC staff in 1987. The licensee could n... justify why these actions are not characterized as a repair ... In response to this inspection finding , the licensee initiate... 2003800152, dated 7/24/03) to evaluate actions to open lir... they are necessary to achieve hot shutdown, and if an exe... required. This issue is identified as URI 50-366/03-06-01, ... Activities (Opening/Closing of Links) to Achieve Safe Shut... remains open pending review and acceptance of additional ... documentation which demonstrates that actions necessary ... considered a repair necessary to achieve and maintain hot...

Use the model of MC 03-07.

.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:

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

Make Smaller

TBD

~~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
cause all eleven SRVs to open at a time when r
may not be available. To mitigate this event, the
report (SSAR) credits the use of Core Spray Loc
However, the licensee did not provide any objec
or analysis) which demonstrated that, assuming
2104, the limited set of equipment available wo
a manner that satisfies the shutdown performan
section L.1.e to 10CFR 50.

Need to review.

This finding is unresolved pending completion of [illegible] 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)

TBD ~~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)

TBD ~~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)

make smaller ⊙ 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

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 transmitter signals.

Analysis: The licensee in their SSAR takes credit for manual control of Group "A" SRVs in order to achieve and maintain safe shutdown conditions. Manual control of Group "A" SRVs are required for a fire in the fire areas selected for review.

The team performed a circuit analysis of SRV 2B21-F013F (Path 1) and SRV 2B21-F013G (Path 2) in order to verify that the design objectives of implementing a one-out-of-two taken twice logic had been achieved. Based on this review the team determined that the design objective of implementing a one-out-of-two taken twice logic had not been installed for the SRVs. The logic installed for the SRVs was a two-out-of-two coincident taken twice logic in addition to a one-out-of-two coincident taken twice logic. The logic implemented results in spurious actuation of group "A" SRVs for a fire in fire area 2104 and defeats the capability to manually control these SRVs as is required per the SSAR.

Enforcement 10 CFR 50, Appendix B, Criterion III, requires that design control measures shall provide for verifying or checking the adequacy of design. The accepted industry standard, ANSI N45.2.11-1974, section 4, requires design activities to provide for relating the final design back to the source of design input.

The logic implemented by the licensee for DCR 91-134 was different from the specified design input requirements. The plant installation failed to correctly implement the one-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 with the eleven SRVs assuming a stuck open mode of operation, 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 50-366/03-06-06, 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