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SOUTHERN COMPANY
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Report of Facility Changes, Tests, and Experiments Safety Evaluation Summaries

Ladies and Gentlemen:

Enclosed is the 24 month report of facility changes, tests, and experiments safety evaluation summaries in accordance with the requirements of 10 CFR 50.59(d)(2).

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Lewis Sumner

H. L. Sumner, Jr.

HLS/il/sdl

Enclosure: Report of Facility Changes, Tests, and Experiments Safety Evaluation Summaries

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Enclosure

Edwin I. Hatch Nuclear Plant

NRC Docket Nos.: 50-321 and 50-366

Operating Licenses: DPR-57 and NPF-5

Report of Facility Changes, Tests, and Experiments
Safety Evaluation Summaries

GLOSSARY
ACRONYMS AND ABBREVIATIONS

ABN	as-built notice
AC	alternating current
ADS	automatic depressurization system
AHU	air handling unit
ALARA	as low as reasonably achievable
APLHGR	average power linear heat generation rate
APRM	average power range monitor
ARI	alternate rod insertion
ARM	area radiation monitor
ARTS	average power range monitor, rod block monitor, and Technical Specifications
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram-recirculation pump trip
BHD	bottom head drain
BOP	balance of plant
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CRD	control rod drive
CS	core spray
CST	condensate storage tank
DAS	data acquisition system
DBA	design basis accident
DBE	design basis earthquake
DC	direct current
DCB	double cantilever beam
DCR	design change request
DCS	dry cask storage
DHR	decay heat removal
dP	differential pressure
ECCS	emergency core cooling system
ECP	electrochemical potential
EDG	emergency diesel generator
EFCV	excess flow check valve
EFPD	effective full power days
EFPH	effective full power hours

GLOSSARY
ACRONYMS AND ABBREVIATIONS

EHC	electrohydraulic control
ELI	Equipment Location Index
EMI	electromagnetic interference
EOC-RPT	end of cycle-recirculation pump trip
EOF	Emergency Operations Facility
EPA	Environmental Protection Agency
ERFDS	Emergency Response Facility Display System
ETS	Environmental Technical Specifications
EQ	Environmental Qualification
FHA	Fire Hazards Analysis
FPC	fuel pool cooling
FSAR	Final Safety Analysis Report
GE	General Electric
GL	Generic Letter
GPC	Georgia Power Company
HCU	hydraulic control unit
HNP	Hatch Nuclear Plant
HPCI	high pressure coolant injection
HVAC	heating, ventilation, and air-conditioning
HWC	hydrogen water chemistry
I&C	instrumentation and control
IE	inspection and enforcement
IGSCC	intergranular stress corrosion cracking
ILRT	integrated leak rate test
IRM	intermediate range monitor
ISFSI	independent spent fuel storage installation
ISI	inservice inspection
IST	inservice testing
LAN	local area network
LCO	limiting condition for operation
LDS	leak detection system
LDCR	license document change request
LLRT	local leak rate test
LLS	low-low set
LOCA	loss of coolant accident
LOSP	loss of offsite power

GLOSSARY
ACRONYMS AND ABBREVIATIONS

LPAP	low power alarm point
LPCI	low pressure coolant injection
LPM	loose-parts monitor
LPRM	local power range monitor
LPSP	low power setpoint
MCC	motor control center
MCPR	minimum critical power ratio
MCR	main control room
MCRECS	main control room environmental control system
MDC	minor design change
MG	motor generator
MPC	Multi-Purpose Canister
MOV	motor-operated valve
MPL	master parts list
MSIV	main steam isolation valve
MS SRV	main steam safety relief valve
MSL	main steam line
MSLRM	main steam line radiation monitor
MSR	moisture separator reheater
NMA	noble metals addition
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
ODCM	Offsite Dose Calculation Manual
OPDRV	operations with the potential to drain the reactor vessel
OPRM	oscillation power range monitor
PAM	post accident monitoring
PASS	post accident sampling system
PCIS	primary containment isolation system
PCIV	primary containment isolation valve
P&ID	piping and instrumentation diagram
PLC	programmable logic controller
PPC	plant process computer
PRB	Plant Review Board
PRNM	power range neutron monitor
PSW	plant service water
PSW	plant service water

GLOSSARY
ACRONYMS AND ABBREVIATIONS

QA	quality assurance
RBM	rod block monitor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
REA	Request for Engineering Assistance
RES	Request for Engineering Services
RFI	radio frequency interference
RFP	reactor feed pump
RFPT	reactor feed pump turbine
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RMCS	reactor manual control system
RPS	reactor protection system
RPT	recirculation pump trip
RPV	reactor pressure vessel
RRS	reactor recirculation system
RSCS	rod sequence control system
RWCU or RWC	reactor water cleanup
RWCS	reactor water cleanup system
RWE	rod withdrawal error
RWM	rod worth minimizer
SAER	Safety Audit and Engineering Review
SAT	station auxiliary transformer
SBGT or SGTS or SGT	standby gas treatment
SCM	stress corrosion monitor
SDC	setpoint design change
SED	System Evaluation Document
SJAE	steam jet air ejector
SLMCPR	safety limit minimum critical power ratio
SNC	Southern Nuclear Operating Company
SoRA	Summary of Required Actions
SPDS	Safety Parameter Display System
SRB	Safety Review Board
SR	Surveillance Requirement
SRM	source range monitor
SRV	safety relief valve

GLOSSARY
ACRONYMS AND ABBREVIATIONS

SSAR	safe shutdown analysis report
SSC	system, structure, or component
TBWD	thrust bearing wear detector
TCV	turbine control valve
THV	torus hardened vent
TIL	Technical Information Letter
TIP	traversing incore probe
TLD	thermoluminescent dosimeter
TM	Temporary Modification
TRM	Technical Requirements Manual
TS	Technical Specifications
TSV	turbine stop valve
Ver.	Version

10 CFR 50.59 SUMMARIES

TEMPORARY MODIFICATIONS (TM)

APC 1-04-033

This Temporary Modification via control of an Annunciator and Plant Component sheet (APC) is to disable the "ROD DRIFT" alarm function for rod 26-39 and rod 34-39 via installation of a jumper on the Probe Buffer Card. The FSAR mentions that a drifting rod is "indicated by an alarm and a red light in the MCR. The rod drift condition is also monitored by the process computer." As a result of this APC, the alarm will not come in for the three rods identified. This TM will allow the drift alarm to function for all other control rods for which the alarm is not disabled. Therefore, the proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FSAR.

APC 1-04-089

This Temporary Modification via control of an Annunciator and Plant Component sheet (APC) is to disable the "ROD DRIFT" alarm function for rod 26-39, rod 34-27 and rod 34-39 via installation of a jumper on the Probe Buffer Card. The FSAR mentions that a drifting rod is "indicated by an alarm and a red light in the MCR. The rod drift condition is also monitored by the process computer." As a result of this APC, the alarm will not come in for the three rods identified. This TM will allow the drift alarm to function for all other control rods for which the alarm is not disabled. Therefore, the proposed activity does not increase the probability of occurrence of an accident previously evaluated in the FSAR.

TM 1-04-021, Rev. 0

This activity is for binding closed or "gagging" 1E11-F200B, which is the minimum flow valve for RHRSW pump 1E11-C001B. The FSAR states that the design function of this valve is a low-flow bypass. Since the valve will be bound closed, no low-flow bypass will be available to the affected pump. This constitutes a change to the FSAR.

The valve addressed by this TM cannot form any part of a sequence of events which could cause an accident. Hence, no change to this valve could affect the probability of occurrence of an accident.

DESIGN CHANGE REQUESTS (DCR)

DCR 90-028, Rev. 0

This DCR provides the design for replacing nine Allis Chalmers safety related starters. Siemens starter pan assemblies will be used as replacements. The replacements are functionally equivalent and testing will be performed to assure the replacements are seismically and environmentally qualified for worst case conditions. The margin of safety of the Motor Control Centers will not be affected by the starter assembly change out.

DCR 98-047, Rev. 0

This DCP will replace RMS-9 trip devices on selected (based on load category) frames of 600V Busses 2A, 2B, 2AA and 2BB with MVT+ trip units. All the busses and their respective loads are nonsafety-related. The function of the 600V switchgear will not change. The breaker trip devices are being replaced to enhance 600V distribution system reliability by reducing spurious breaker trips. The analysis of trip unit failure presented in the Discussion section of the safety evaluation addresses the potential impact of EMI/RFI conducted and radiated emissions from the new trip units on any safety-related equipment or system. The conclusion is that no safety-related equipment or system would be adversely affected by the installation of these trip units. Therefore, the probability of an accident previously evaluated in the FSAR will not increase.

DCR 99-050, Rev. 0

This DCR provides the design for the replacement of the Safety Parameter Display System (SPDS) and the Emergency Response Facility Display System (ERFDS). The design functions of the existing SPDS will be retained in the new system, although they will be accomplished using different software and microprocessor-based hardware. The SPDS has no control functions. It is a monitoring system only. The same information, including calculated values, will be displayed to the plant operators as the existing system. This DCR is considered to be a digital upgrade. EMI/RFI testing has demonstrated that the RTP racks are not expected to adversely impact any safety related or important to safety system. All interconnections between Class 1E equipment and non-class 1E components for the new SPDS will be properly isolated.

DCP 1H03-009, Rev. 3

This design change will provide adequate margin for the Unit 1 reactor building crane main hoist to lift and lower the rated 125 ton loads, improve crane

reliability, and bring the crane into compliance with OSHA requirements. The addition of a pulley on the main hoist drum, which is coupled to an auxiliary shaft via a gear belt, introduces a new potential failure. This change may be considered adverse. However, there are no design basis accidents in the FSAR for which the crane is the initiator. The pulley driven overspeed switch provides a backup method for preventing a load drop. The Unit 1 Reactor Building Crane is a single failure proof crane for which a load drop is not considered a credible event. The new switch, belt and pulley assembly is considered to be as rugged as the prior system. Because the potential failures are similar, no new possibility of a malfunction of an SSC important to safety with a different result than previously evaluated in the Updated FSAR is expected by this modification.

DCP 1H03-026T, Rev. 0

This DCR adds a blind plate in place of the 1P41-D166 orifice plate, located in the 1B diesel generator room, and remove a section of the 12" shield piping in the 2G switchgear room and relocate service water vent and drain valves. This design change will not affect the service water supply to any component except the 1B diesel generator. This change will remove the Division I back-up supply. This does not reduce the reliability of any component. It only reduces the redundancy in available back-up cooling water supplies. PRA evaluation shows that there is no increase in the average risk following implementation of this design change.

MDC 03-5009, Rev. 0

This MDC will permanently remove the automatic PSW transfer and isolation logic for the 1Z41-B008B MCR AC unit. This is a change to the plant as described in the Unit 1 FSAR section 10.7.6. This logic automatically transfers PSW supply from Div. I to Div. II in the event of a low flow condition in Div. I in conjunction with a loss of offsite power or a loss of coolant accident. Therefore, this logic is designed to function following a LOCA or LOSP and the subsequent occurrence of a problem causing a low flow in Div. I of the PSW system. This change has no impact upon the probability of a problem occurring. The overall system operation will remain as described in the FSAR. Therefore, the probability of a system failure remains unaffected as well.

DCP 1040113801, Ver. 1

This DCP is to implement setpoint and calibration changes to facilitate implementation of the 10 PSI reactor pressure increase to allow the achievement of 100 percent of the rated power approved under Appendix K uprate. Because of a slight increase in rated containment pressure the License Amendments for REA 00-650/RER 2003-254 (RPV 10 PSI Increase, LDCR-2003-077 (Tech Spec Changes) & LDCR 2004-040 (FSAR Revisions) that allows the increase must be approved prior to implementation of this DCP. LDCR 2004-041 which is a result

of the change in steam density is included with this DCP. This DCP is to implement the setpoint and calibration changes associated with the 10 PSI increase. There are no impacts to the frequency or occurrence of accidents, likelihood of occurrence of a malfunction of a structure, system, or component (SSC), consequences of accidents previously evaluated, consequences of SSC malfunctions, possibility for the creation of an accident of a different type, possibilities for malfunctions of SSC's, impact on the fuel cladding, reactor coolant pressure boundary, or containment, or changes in the method of evaluations due to the changes associated with this package. Were the 10 PSI increase not to occur and these changes were implemented the result would be a reduction in operating margin rather than safety margin.

DCP 2040113901, Ver. 1

This DCP is to implement setpoint and calibration changes to facilitate implementation of the 10 PSI reactor pressure increase to allow the achievement of 100 percent of the rated power approved under Appendix K uprate. Because of a slight increase in rated containment pressure the License Amendments for REA 00-650/RER 2003-254 (RPV 10 PSI Increase, LDCR-2003-077 (Tech Spec Changes) & LDCR 2004-040 (FSAR Revisions) that allows the increase must be approved prior to implementation of this DCP. LDCR 2004-039 which is a result of the change in steam density is included with this DCP. This DCP is to implement the setpoint and calibration changes associated with the 10 PSI increase. There are no impacts to the frequency or occurrence of accidents, likelihood of occurrence of a malfunction of a structure, system, or component (SSC), consequences of accidents previously evaluated, consequences of SSC malfunctions, possibility for the creation of an accident of a different type, possibilities for malfunctions of SSC's, impact on the fuel cladding, reactor coolant pressure boundary, or containment, or changes in the method of evaluations due to the changes associated with this package. Were the 10 PSI increase not to occur and these changes were implemented the result would be a reduction in operating margin rather than safety margin.

LICENSING DOCUMENT CHANGE REQUESTS (LDCR)

LDCR 2003-076, Rev. 0

This proposed change is to revise U1 and U2 TS Bases to remove the discussion of the automatic swap of PSW cooling water to the "B" Control Room AC unit; add a description to the U1 FSAR for the manual action, in place of the automatic action, for providing cooling water to the MCR AC unit 1Z41-B008B from the PSW Division II; and add component "condensing unit," its malfunction and

comments to the MCR HVAC Systems Failure Analysis Table as the result of the manual action. This a change to the plant as described in the U1 FSAR section 10.7.6 and U1 TS Bases B 3.7.5. The automatic transfer function is merely a design feature of the B MCR AC train that is included for additional defense-in-depth. Retention of this feature is not necessary for operability of the MCR AC units.

Probability of failure of the PSW function is unaffected. The overall system operation will remain as described in the FSAR. Therefore, the probability of system failure remains unaffected as well.

LDCR 2004-006, Ver. 1

1. The LDCR addresses Unit 1 TRM TLCO 3.3.10 in that a one-time extension of the completion time for the LCO is being proposed from 24 hours to 7 days.

2. On January 30, 2004, the Unit 1 turbine "A" Master Trip Solenoid failed to function during the performance of TSR 3.3.10.1. This failure placed the unit in a required action to isolate the turbine from the steam supply within 24 hours. The initial apparent cause of the failure is due to sticking of the solenoid.

3. The safety basis for the proposed one-time change is provided as follows:
   - With the master trip solenoid inoperable, electrical overspeed protection for the turbine is not available. However, the mechanical overspeed trip is unaffected by this component failure.
   - The surveillance on the mechanical overspeed trip is up-to-date.
   - The FSAR (HNP-1-FSAR-7.11.3) identifies the mechanical overspeed as the protection feature credited with preventing catastrophic overspeed of the turbine.
   - The backup overspeed trip is credited only when the mechanical overspeed trip is locked out for testing.
   - Significant margin (~50%) exists between the mechanical overspeed trip setpoint and the speed at which the overstress could possibly lead to failure.
   - The incremental probability of occurrence of an overspeed event during the seven days allowed by this change is judged to be very small.

   The following compensatory measures are recommended for the seven-day period:
   - In order to decrease even further the potential for grid-induced events that might result in a generator load rejection, switchyard work involving breakers should be curtailed. No work that would result in a change in EOOS color is permitted.

- Operations personnel should brief at the beginning of each shift that the Master Trip Solenoid Valve (MTSV) is unavailable and prompt operator action may be required to cause turbine trip.
- GENCOM should be apprised of the turbine status (without electrical overspeed protection) and asked to minimize any action that could lead to a load reject.

LDCR 2004-011, Rev. 1

1. Revision 1 addresses Unit 1 TRM TLCO 3.3.10 in that a one-time extension of the completion time for the LCO is being proposed from 24 hours to 8 days.

2. On February 06, 2004, the Unit 1 turbine "B" Master Trip Solenoid failed to function during the performance of TSR 3.3.10.1. This failure placed the unit in a required action to isolate the turbine from the steam supply within 24 hours. The initial apparent cause of the failure is due to sticking of the solenoid.

3. The safety basis for the proposed one-time change is provided as follows:
   - With the master trip solenoid inoperable, electrical overspeed protection for the turbine is not available. However, the mechanical overspeed trip is unaffected by this component failure.
   - The surveillance on the mechanical overspeed trip is up-to-date.
   - The FSAR (HNP-1-FSAR-7.11.3) identifies the mechanical overspeed as the protection feature credited with preventing catastrophic overspeed of the turbine.
   - The backup overspeed trip is credited only when the mechanical overspeed trip is locked out for testing.
   - Significant margin (approximately 50%) exists between the mechanical overspeed trip setpoint and the speed at which the overstress could possibly lead to failure.
   - The incremental probability of occurrence of an overspeed event during the 8 days allowed by this change is judged to be very small.

   The following compensatory measures are recommended for the 8-day period:
   - In order to decrease even further the potential for grid-induced events that might result in a generator load rejection, switchyard work involving breakers should be curtailed. That is, no work that would result in a change in EOOS color is permitted.
   - Operations personnel should brief at the beginning of each shift that the MTSV is unavailable and prompt operator action may be required to cause turbine trip.

- GENCOM should be apprised of the turbine status (without electrical overspeed protection) and asked to minimize any action that could lead to a load reject.

LDCR 2004-014, Rev. 0

This proposed change is to revise the Unit 1 TRM TSR 3.9.3.1 to remove the requirement to perform a hoist limit loaded interlock test for the refueling platform fuel grapple and the auxiliary hoist every 7 days after its initial performance. This is a change to the Unit 1 TRM that relaxes the surveillance frequency requirements on the loaded interlock surveillances for the refuel platform fuel grapple and the auxiliary hoist.

The loaded interlocks setpoint surveillance in the TRM insures that the fuel grapple and the auxiliary hoist are capable of detecting when a fuel bundle has been lifted by the refuel platform. That fuel loaded signal is a part of the refueling interlocks. For example, when the fuel grapple is loaded, a rod withdrawal block will engage if the refuel platform is near the core and the mode switch is in the refuel position. This particular TRM surveillance verifies that the setpoint on the fuel grapple loaded signal is adequate. A separate Technical Specifications surveillance (not affected by this TRM revision) will insure that the integrated signals together provide the necessary signal (in the example above, the rod block).

Furthermore, the nature of the refueling interlocks is such that any problems with the interlocks will be evident to the refuel platform operator. As a result, increased frequencies for their surveillances are of questionable value. For example, when the fuel grapple is loaded with a fuel bundle, a "fuel grapple loaded" annunciator is provided to the operator in the platform cabin. It is therefore likely that any failure of this interlock will be obvious to the operator. This point is noted in the Technical Specifications Bases for the refueling interlock surveillance SR 3.9.1.1: "The 7 day frequency is based on engineering judgment and is considered adequate in view of other indications of refueling interlocks and their associated input status that are available to unit operations personnel."

No other plant systems are involved in the TRM change.

For the above reasons, the likelihood of occurrence of a previously evaluated system or component malfunction is not increased.

LDCR 2004-022, Ver. 2

This is a proposed change to the Unit 1 TRM and FSAR to increase the acceptance criterion on the RCIC AC inboard valve from 20 seconds to 25 seconds.

The safety function of this AC valve, in conjunction with the outboard DC valve, 1E51-F008, is to automatically isolate on a RCIC steam supply line break in the reactor building. During the 2004 Spring Refueling outage, a gearing change was made to the valve operator and, as a result, the as left close stroke time increased to 19.9 seconds. This meets the acceptance criterion of 20 seconds, but obviously leaves very little margin. It is therefore desired to explore the possibility of increasing the stroke time acceptance criterion.

Rupture of the RCIC steam line is one of the high energy line breaks (HELB) analyzed in the FSAR. Other examples include rupture of a main steam line, the High Pressure Coolant Injection (HPCI) steam supply line the RWCU supply line. These breaks are non-limiting with respect to fuel limits and vessel inventory because, unlike the breaks in the primary containment, they isolate. The HELB safety analysis calculates the mass and energy escaping from the break into the reactor building from event initiation until the valve is fully closed. This mass and energy release is used to calculate the peak temperatures and pressure differentials within the reactor building and is also the basis of the environmental qualification (EQ) temperature profiles in the reactor building.

The calculation for the RCIC HELB has been reviewed and it was determined that an increase in the isolation stroke time acceptance criterion from 20 to 25 seconds can be done without affecting the reactor building temperatures, pressures, or the EQ temperature profile in the reactor building.

CAUTION TAGS

1-CA-04-1E11-00144, Rev. 0
1-CA-04-1E11-00147, Rev. 0

This activity is for binding closed or "gagging" 1E11-F200B and 1E11-F200C, which are the minimum flow valves for RHRSW pump 1E11-C001B and 1E11-C001C. The purpose of maintaining these valves in the closed position is to ensure that sufficient flow can be developed from the RHRSW at the worst-case conditions of river level. The FSAR states that the design function of these valves is a low-flow bypass. Since the valves will be bound closed, no low-flow bypass will be available to the affected pump. This constitutes a change to the FSAR.

The valves addressed by this activity cannot form any part of a sequence of events which could cause an accident. Hence, no change to these valves could affect the probability of occurrence of an accident.

2-CA-04-2E11-00059, Rev. 0

This activity is for binding closed or "gagging" 2E11-F200B and 2E11-F200C, which are the minimum flow valves for RHRSW pump 2E11-C001B and 2E11-C001C. The purpose of maintaining these valves in the closed position is to ensure that sufficient flow can be developed from the RHRSW at the worst-case conditions of river level. The FSAR states that the design function of these valves is a low-flow bypass. Since the valves will be bound closed, no low-flow bypass will be available to the affected pump. This constitutes a change to the FSAR.

The valves addressed by this activity cannot form any part of a sequence of events which could cause an accident. Hence, no change to these valves could affect the probability of occurrence of an accident.