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February 29, 1996

Docket Nos. 50-321
50-366

HL-5115

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

Edwin I. Hatch Nuclear Plant
Annual Operating Report for 1995

Gentlemen:

Enclosed is the 1995 Annual Operating Report for Edwin I. Hatch Nuclear Plant Unit 1, Docket No. 50-321, and Unit 2, Docket No. 50-366. This report is submitted in accordance with the requirements of 10 CFR 50.59(b)(2).

Sincerely,

J. T. Beckham, Jr.

SRP/ld

Enclosure: 1995 Annual Operating Report for Plant Hatch Units 1 and 2

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ENCLOSURE

EDWIN L. HATCH NUCLEAR PLANT - UNITS 1 AND 2
NRC Docket Nos. 50-321 and 50-366
Operating Licenses DPR-57 and NPF-5

ANNUAL OPERATING REPORT
1995

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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
APRM	average power range monitor
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram-recirculation pump trip
BOP	balance of plant
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
CFUF	cumulative fatigue usage factor
CITM	contact integrating thermal monitor
CPIS	containment purge and inerting system
CRB	control rod block
CRD	control rod drive
CS	core spray
CST	condensate storage tank
DBA	design basis accident
DBE	design basis earthquake
DC	direct current
DCR	design change request
dP	differential pressure
ECCS	emergency core cooling system
ECP	electrochemical potential
EFPD	effective full power days
EFPH	effective full power hours
EHC	electrohydraulic control
EOC-RPT	end of cycle-recirculation pump trip
FHA	Fire Hazards Analysis
FSAR	final safety analysis report
GE	General Electric
GL	Generic Letter
GPC	Georgia Power Company
HNP	Hatch Nuclear Plant

GLOSSARY

ACRONYMS AND ABBREVIATIONS

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
ISI	inservice inspection
IST	inservice testing
ITS	Improved Technical Specifications
LCO	limiting condition for operation
LDS	leak detection system
LLRT	local leak rate test
LLS	low-low set
LOCA	loss of coolant accident
LOSP	loss of offsite power
LPCI	low pressure coolant injection
LPM	loose-parts monitor
LPRM	local power range monitor
MCC	motor control center
MCPR	minimum critical power ratio
MCR	main control room
MCRECS	main control room environmental control system
MDC	minor design change
MOV	motor-operated valve
MPL	master parts list
MSIV	main steam isolation valve
MSL	main steam line
MSLRM	main steam line radiation monitor
NMA	noble metals addition
NPPM	Nuclear Procurement Policy Manual
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NWS	normal water chemistry
OPDRV	operations with a potential for draining the reactor vessel

GLOSSARY

ACRONYMS AND ABBREVIATIONS

PASS	post accident sampling system
PCIOMR	pre-condition interim operating management recommendations
PCIS	primary containment isolation system
PCIV	primary containment isolation valve
P&ID	pipng and instrumentation diagram
PRB	Plant Review Board
PSW	plant service water
RBM	rod block monitor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
REA	request for engineering assistance
RES	request for engineering services
RFP	reactor feed pump
RFPT	reactor feed pump turbine
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RPS	reactor protection system
RPV	reactor pressure vessel
RRS	reactor recirculation system
RTD	resistance temperature detectors
RWCU or RWC	reactor water cleanup
RWCS	reactor water cleanup system
RWE	rod withdrawal error
RWM	rod worth minimizer
SAT	station auxiliary transformer
SBGT or SGTS or SGT	standby gas treatment
SCM	stress corrosion monitor
SJAE	steam jet air ejector
SNC	Southern Nuclear Operating Company
SRB	Safety Review Board
SR	Surveillance Requirement
SRM	source range monitor
SRV	safety relief valve
SSC	system, structure, or component
TCV	turbine control valve
THV	torus hardened vent

GLOSSARY

ACRONYMS AND ABBREVIATIONS

TIL	Technical Information Letter
TIP	traversing incore probe
TLD	thermoluminescent dosimeter
TOL	thermal overload
TRM	Technical Requirements Manual
TSs	Technical Specifications
TSV	turbine stop valve

INTRODUCTION

The Edwin I Hatch Nuclear Plant is a two-unit facility located approximately 11 miles north of Baxley, Georgia, on U.S. Highway 1. The plant consists of two light water reactors. Unit 1 is licensed to operate at 2436 MWt. Unit 2 was licensed to operate at 2436 MWt through Cycle 12 and is licensed to operate at 2558 MWt for the remainder of the license. The maximum dependable capacity for 1995 on Unit 1 was 741 net MWe. The maximum dependable capacity for 1995 on Unit 2 was 765 net MWe through December 2, after which the maximum dependable capacity was 809 net MWe. General Electric furnished the boiling water reactor, the nuclear steam supply system, the turbine, and the generator for both units. The plant was designed by Southern Company Services, Inc., with assistance provided by Pechtel Power Corporation. The condenser cooling method employs induced-draft cooling towers and recirculating water systems with normal makeup supplies drawn from the Altamaha River.

The plant is a co-owned facility with ownership delegated as follows:

Georgia Power Company	50.1%
Oglethorpe Electric Membership Cooperative	30.0%
Municipal Electrical Authority of Georgia	17.7%
City of Dalton, Georgia	2.2%

Licensing information for the units is as follows:

	<u>Unit 1</u>	<u>Unit 2</u>
Docket Number	50-321	50-366
License Issued	08/06/74 (DPR-57)	06/13/78 (NPF-5)
Initial Criticality	09/12/74	07/04/78
Initial Synchronization	11/11/74	09/22/78
Commercial Operation	12/31/75	09/05/79

Georgia Power Company has sole responsibility for overall planning, design, construction, operation, maintenance, and disposal of the Hatch Nuclear Plant.

SAFETY RELIEF VALVE CHALLENGES FOR 1994

Unit 1DateValves

4/11/95

1B21-F013A,B,C,D,E,F,G,H,J,K,&L

An improperly placed jumper caused the reactor water level signal to decrease to zero. The reactor water level control system responded by increasing flow from the reactor feed pumps. The reactor automatically shut down following an automatic trip of the main turbine on high reactor water level. Both reactor recirculation pumps and both feedwater pumps tripped. Reactor pressure increased, resulting in the opening of all SRVs and the main turbine bypass valves. The SRVs closed at the appropriate pressures as the reactor vessel pressure decreased and the bypass valves automatically controlled pressure.

UNIT 2

No SRV challenges occurred this year.

10 CFR 50.59 SUMMARIES FOR 1995

UNIT 1/COMMON AS-BUILT NOTICES

94-0155, Rev 0

Change the automatic water spray system from automatic to manual for the MCR ventilation filter trains per DCR 87-147. Revise Units 1 and 2 FSARs to reflect this change and correct the setpoint for the temperature control for the electric carbon drying heating coils in the filter trains.

The filter trains are safety-related, but the temperature switches are not listed as safety-related devices. The FSAR states that the continued operation of, or total failure of, the heater has no effect on the safety of the associated equipment. This setpoint is within the scope of the previous evaluation. Equipment function is unchanged.

94-0234, Rev 0

Revise Unit 1 FSAR figure 11.1-1, sh 3 of 3, to add a steam seal system P&ID continuation reference, and figure 11.4-1, sh 3 of 3, to add N33-B001 drain P&ID reference. Make corrections to main turbine steam seal and lube oil systems on other drawings.

The involved components are nonsafety-related and are not required to effect or support the safe shutdown of the reactor or perform in the operation of reactor safety features. The drawings are corrected to bring them into compliance with the original design as supported by other design documents. No system functions, responses, or interfaces with safety-related systems are changed. TSs revision is not required.

94-0242, Rev 0

Move a fire extinguisher from the north wall of the C52 entry hallway in the control building, el 110 ft, to the south wall to provide easier personnel access and egress. Revise FHA drawing H-11815 accordingly.

This modification has no impact on any plant accident or equipment malfunction scenario. The reliability, availability, and accessibility of the extinguisher are unaffected. The extinguisher does not interface with any equipment important to safety. Neither the safety limits nor failure points of the extinguisher or any other equipment or system are affected by the relocation.

UNIT 1/COMMON AS-BUILT NOTICES

ME-0080, Rev 0

Revise Unit 1 FSAR figure 8.5-1, sh 1 of 2, to delete the Division I designation from 125-V dc cabinet 1R25-S003.

This change does not affect any accident precursors nor does it alter, degrade, or prevent actions assumed to occur in previously analyzed accidents. No new system parameters or failure modes are introduced, and system logic, function, and operation remain unchanged. Cabinet 1R25-S003 is not safety related, and all circuits in this cabinet are nondivisional and nonessential.

UNIT 2 AS-BUILT NOTICES

92-0458, Rev 0

Correct the range of the measured radiation levels on the panel instrument label for area radiation monitor recorder 2D21-R600. Revise Unit 2 FSAR table 12.3-1 accordingly.

This instrumentation and its panel are nonsafety related. This change does not increase the personnel allowable radiation exposure limit. The recorder continues to operate in a manner consistent with the FSAR evaluation. No new modes of operation are established, and no LCO is exceeded.

94-0207, Rev 0

Change label color codings and label locations for instruments having like condensing chambers. Revise Unit 2 FSAR figure 7.5-1, shs 4, 5, and 6 of 21, accordingly.

These changes conform to NUREG -0700 and do not affect the operation, reliability or accident response of the involved instruments, the main steam system or any other equipment important to safety. The effect on equipment important to safety is unchanged, and no new accident mechanisms are introduced. This modification does not change any setpoint, safety limit, or failure point.

94-0213, Rev 0

Document that the dissolved oxygen flow chamber does not exist as a part of the PASS design. (GPC took exception to measuring dissolved oxygen to meet the requirements of RG 1.97. NUREG-0737 recommends but does not mandate use of an oxygen flow chamber.)

This change is in compliance with NUREG-0737. The dissolved oxygen flow chamber does not perform any function associated with any safety-related system, and its removal does not change the required operational modes of the PASS. The TSs do not specify any detail regarding the dissolved oxygen; therefore, the margin of safety is not reduced.

UNIT 1/COMMON DESIGN CHANGE REQUESTS

87-147, Rev 0

Replace the fire protection automatic deluge feature for carbon filters in filtration units with manually activated deluge systems to eliminate the threat of an accidental discharge of water onto the filter carbon adsorbents.

Due to the characteristics of carbon filter fires, the response time differential has a negligible effect on the propagation and severity of a carbon bed fire within the filtration units. The transition from automatic deluge to manual activation does not constitute an increase in the consequences of a fire. Some filtration units are safety related and have backup filtration systems. Since the affected unit can be isolated and contained, a fire poses no threat to the operation of the backup system, or to any safety-related equipment in the area.

88-0120, Rev 0

Provide a permanent HVAC system with a larger cooling capacity to cool the radwaste control room.

No equipment assumed to operate as a result of an accident and no equipment important to safety are dependent upon this HVAC system for proper function. The original fan and coil unit will be maintained as a backup unit since this system is nonsafety related, its operation will have no effect on the reactor or any other safety-related equipment. This system is not addressed in the TSs.

88-0192, Rev 2

Heat trace the inlet and outlet sample lines for the main stack area radiation monitor to maintain the temperature above the maximum expected condensation temperature. Add a seismically supported control and status panel, aluminum sheathed cable, heating cable power connection boxes, and a permanently mounted power supply test card and line filter to the microprocessor panel.

The additions to the microprocessor panel do not adversely affect the operation of the monitor and will not change, degrade, or prevent actions described or assumed in an accident. The panel's function is nonsafety related, and the panel does not interface with any safety-related equipment. No control actions are initiated by this system, and no new interfaces are introduced.

UNIT 1/COMMON DESIGN CHANGE REQUESTS

91-0165, Rev 0

Install two strainers in parallel on each of the 4-in. PSW headers serving the MCR HVAC units and provide instrumentation that indicates when strainers need to be cleaned.

This modification meets the original design, material, and construction requirements applicable to the safety-related PSW system. The operation and function of the PSW system and the strainers are unchanged. This change allows the strainers in the PSW supply to be cleaned without disabling one of the condensing units.

92-0152, Rev 0

Add strainers in the turbine building PSW inlet to the two condensing units utilized in cooling the health physics and chemistry area.

The health physics and chemistry area HVAC system and the turbine building PSW system serving the condensing units are not safety related. Each strainer is equipped with a blowdown valve and means for isolation for strainer basket removal. No new modes of failure in the safety-related portion of the PSW system are introduced. No operation or function of any equipment important to safety is affected by the change.

93-0040, Rev 0

Provide an additional power circuit from the vital ac panel for the condensate booster pump and reactor feed pump minimum flow control valve flow switches. Replace the alarm modules for these pumps and the differential pressure controller for the SJAE condensate flow with more modern equipment.

This modification decreases the probability of a transient being initiated due to the condensate and feedwater minimum flow valves failing open due to a loss of power. No plant transient responses are adversely affected. The seismic integrity of the MCR panels is not adversely impacted. This modification to nonsafety-related equipment does not alter or potentially affect any safety-related equipment or any equipment required to support the operation of safety-related equipment.

93-0060, Rev 1

Install two new nonsafety-related AHUs (each with a chilled water cooling coil, prefilters, and a supply fan) for the refueling floor dressout area, the reactor building elevator equipment room, and the radio room. Replace the outdated passenger elevator controls with modern solid-state controls with microprocessors.

UNIT 1/COMMON DESIGN CHANGE REQUESTS

The new nonsafety-related equipment does not provide cooling for any equipment required to function during or following a DBA. This design incorporates applicable codes and standards. The cooling equipment mounting, piping, raceways, controls, and ductwork incorporate Seismic Category II/I design considerations. Failure of the new elevator controls will not affect any safety-related systems.

93-5009, Rev 0

Add permanent equipment to inject oxygen into the nonsafety-related feedwater system to maintain the proper concentration of 20 to 50 ppb.

There is no change to any ECCS, reactivity control, or other safety-related equipment assumed to operate in response to an accident. The reliability and operation of equipment are not adversely affected, and the chemical effects are beneficial. The addition of this system has no adverse effect on any acceptance limit or failure point of any system or component addressed in the TSs.

94-0024, Rev 0

Add a helper mechanical draft cooling tower and include cooling tower site preparation, basin construction, and interface tie-in work with the existing cooling towers.

All activities associated with this change are nonsafety-related. No equipment important to safety is located in the vicinity of this modification. No accident response of any equipment or system is adversely affected by this change. The new cooling tower is as reliable as the existing cooling towers and has no negative impact on the reliability of the towers, any associated equipment, or any equipment important to safety.

94-0027, Rev 0

Add a signal processor to the reactor building exhaust flow recorder instrument loop to alleviate flow oscillations. The signal processor continuously self averages each flow input to provide a better indication of flow depicted on the recorder and provides indication of gross channel failure.

This portion of the reactor zone HVAC system is not safety-related. Equipment required to operate in response to an accident is not affected by this modification. The seismic integrity of the involved MCR panel is not affected, and the power supply at the recorder is adequate for the additional load of the new signal processor. The function of the instrument loop is unchanged.

UNIT 1/COMMON DESIGN CHANGE REQUESTS

95-0006, Rev 0

Change the SGTS fan logic such that the SGTS fan with low flow will shut down, but remain available, and automatically initiate should the operating fan trip or experience low flow. Change heater logic to energize the heater whenever a fan is running, regardless of flow and contains a high temperature cutoff.

The change to the safety-related SGTS increases reliability by preventing excessive operation of both fans and possible damage due to fan operation in the unstable portion of the flow curves. Allowing the heater to be on whenever the fan is running prevents inadequately treated effluents discharging to the stack. Adequate operation is ensured because the non-running filtration train will remain on standby and automatically initiate should the other train trip or experience low flow. No new loads or operational requirements are added, and the capability of the system to meet its TSs requirements is not reduced.

UNIT 2 DESIGN CHANGE REQUESTS

88-0355, Rev 0

Replace starter breakers in MCCs 2R24-S011 and 2R24-S012, and a feeder breaker in 125-V dc cabinet 2R25-S003, because the existing breakers do not provide the required short circuit protection to the motor circuits.

The replacement breakers are Class 1E and qualified to the same requirements as the existing ones. Replacement of the breakers provides the required coordinated short circuit protection to the safety-related motor circuits and nonsafety-related annunciator circuit. Replacement of the breakers does not increase the probability of circuit failures. System function and response are unaffected by this design change. The existing power cables are adequately sized for the loads.

90-0128, Rev 0

Modify the RFP seal water injection system by replacing seal water injection controllers with programmable logic controllers, increasing pump water bearing brackets vent area, and adding breathing devices on bearing housings and oil leakoff connections on loop seals.

The RFP seal water injection system does not serve any safety function or impact the performance of any safety-related systems. This change creates no new accident scenarios or failure mechanisms. This modification enhances RFP seal water injection system performance and provides better control of RFP bearing temperature. The TSs do not address the RFP seal water injection system.

90-0164, Rev 0

Replace the existing RFPT mechanical hydraulic control system with an electronic control system. Install new wiring and conduit runs.

The electronic controller is a fault-tolerant design that reduces the likelihood of a controller failure. The consequences of transients involving the feedwater system are unchanged by this modification, and system failure modes remain the same. The feedwater system and its RFPT controls are nonsafety related. This modification does not impact the seismic integrity of the MCR panels.

91-0009, Rev 0

Replace the TF-1 liquid in electrical transformers with silicon to remove Poly-chlorinated Biphenol. Derate the transformers by 15%, because the silicone cooling capability is less. Part of the switchgear served by the transformers is safety related. A suitable alternate power source supplying affected loads is used when this modification is made. Because

UNIT 2 DESIGN CHANGE REQUESTS

the transformers can still handle their present loads, no system or equipment operation, response, or reliability is affected. The thermal, dielectric, and fire hazard characteristics of silicone are acceptable.

91-0030, Rev 0

Install decontamination taps on the reactor recirculation suction piping to allow for chemical decontamination.

The reactor recirculation piping forms the reactor coolant pressure boundary and, therefore, is safety related. The addition of the connections does not impose any adverse loads on the piping, and the failure modes are unchanged. The radiological consequences of a LOCA due to failure of a decontamination connection are bounded by the existing LOCA analysis. The modification does not affect the operating characteristics or accident response of the reactor recirculation system.

91-0139, Rev 1

Replace the existing offgas hydrogen analyzers 1/4-in. sample line tubing with 1/2-in. tubing to improve the performance and reliability of the analyzers by minimizing moisture-induced trip.

No setpoints, safety limits, plant responses, failure points, or safety equipment are impacted by this modification. The operation or failure of the nonsafety-related offgas hydrogen analyzers does not impact any safety-related system, structure or component. No new seismic hazards are created. The existing system logic and operation are unchanged, and no new failure modes are introduced.

92-0003, Rev 1

Relocate the low flow alarms for the refueling floor ventilation system supply fans to a different flow element to correct erroneous alarms. Lower the setpoint of the flow switches for the exhaust fan low flow alarms and replace the existing obsolete recorder.

The refueling floor ventilation system does not perform any safety-related function other than secondary containment isolation, and isolation damper function is not impacted by this change. System flows still comply with the original design requirements. The modification does not impact the performance of any safety-related system. The seismic qualification of the panel is not adversely affected.

UNIT 2 DESIGN CHANGE REQUESTS

92-0091, Rev 0

Replace the HPCI system GEMAC flow controller with a Yokogawa microprocessor based programmable controller. Replace five GE indicators with Dixon indicators. The flow controllers are safety-related, but the indicators are nonsafety-related. Response times and other system performance characteristics are not adversely affected, and interlocks, trips, and isolation functions expected to occur in the event of an accident are unaffected. Because HPCI is a single-loop system, the original common-mode and single-failure criteria are not affected. The operating system software was verified and validated consistent with industry standards to assure that no malfunctions of a different type are introduced.

92-0093, Rev 0

Replace the RCIC system GEMAC flow controllers with Yokogawa microprocessor based programmable controllers. Replace six GE indicators with Dixon indicators.

The controllers and one indicator are safety-related. Response times and other system performance characteristics are not adversely impacted, and interlocks, trips, and isolation functions expected to occur in an accident are unaffected. The operating system software and the control algorithm were validated and verified consistent with industry standards to assure that no malfunctions of a different type are introduced.

93-0050, Rev 0

Convert the PSW system high flow isolation signal to a high flow annunciation signal. Provide new lower setpoints closer to the expected peak pressure differentials for normal plant operation, but sufficiently high to preclude nuisance annunciations.

The dP switches are safety related, while the annunciators are nonsafety related. Deletion of the high dP signals to the isolation valves does not cause the PSW system to be operated outside its design limits, nor does it adversely affect the safety-related portions of the PSW system. This modification does not change, degrade, or prevent actions described or assumed in an accident, since the nonsafety-related portions of the PSW system can still be isolated after a break. Although this isolation may be manual, adequate time exists to take appropriate action. The automatic closure signals for LOCA, LOSP, and turbine building (condenser room) flooding are not changed.

UNIT 2 DESIGN CHANGE REQUESTS

93-0059, Rev 0

Install an incore SCM system for increased HWC flowrates. The safety-related portion provides ECP sensors in various locations and routes cables via a replacement drywell penetration to the existing data acquisition panel. The nonsafety-related portion includes modification of the HWC verification system to accommodate the SCM system, the HWC injection system to allow increased flowrates of oxygen and hydrogen, and the HWC control, monitoring and sampling systems.

Operation with increased flowrates does not adversely impact fire protection and operational safety features evaluated for HWC activities in compliance with the Guidelines for Permanent HWC Installations. Based on test and operational experience, the modified LPRM assembly and its cover tube meet the applicable codes and standards and perform equally as well as the standard designs. Any broken ECP sensor parts would be sufficiently small that they would not cause unacceptable coolant flow blockage or malfunction of a structure, system or component important to safety. The operability and performance of the RRS, RWCS, and neutron monitoring systems are unchanged. The injection of hydrogen into the reactor coolant does not change reactor operating conditions. The change in margin of safety from the MSL radiation monitor setting is negligible, because for gross fuel failures, the expected response would be many orders of magnitude greater than the increased N-16 background from the increased injected hydrogen.

93-0068, Rev 0

Replace existing RCIC to CST valve 2E51-F022 with a new DRAG valve designed to resist cavitation and increases the stroke time from 15 to 42 seconds. Replace check valve 2E51-F023 with a new nozzle check valve designed to yield a lower pressure drop and relocate it downstream of the DRAG valve.

The function of the RCIC system or of any other safety-related system is not altered by this change. The piping system and supports were reanalyzed to ensure Seismic Category I integrity. The existing failure modes are unchanged. The modification meets the requirements of the necessary codes and standards to preclude the possibility of adversely affecting any other safety-related equipment or introducing any new accidents.

94-0003, Rev 0

Remove all testable control circuitry and equipment associated with CS check valves 2E21-F006A&B, allowing them to function as simple check valves.

UNIT 2 DESIGN CHANGE REQUESTS

Since removal of the test features and the associated indicating lights for the safety-related CS check valves does not affect the mechanical operation, basic function, reliability, or integrity of these valves, no new failure modes or accidents are introduced. The testable features are no longer utilized, since a manual method of operability testing is currently used. Plant response to evaluated accidents does not change. The pressure boundary of the valves is not adversely affected.

94-0004, Rev 0

Remove all testable control circuitry and equipment associated with LPCI check valves 2E11-F050A&B, making them simple check valves

Since only the testable portions of the safety-related valves are removed, the mechanical operation and reliability of the valves remain unchanged, and no new failure modes or accidents are introduced. Plant response to evaluated accidents does not change. No acceptance limits or failure points are affected.

94-0016, Rev 0

Add stiffener plates to one beam located at el 148 ft in the drywell and remove a snubber on MSL D.

The modification to the structural steel brings the stresses within code allowables. The piping analysis which is based on a refined piping analysis methodology developed by GE, was performed with the snubber removed. All support loads and piping are adequate. The changes were seismically analyzed, and safety-related equipment is not adversely affected. This modification does not change any operating parameter or system response, and no new modes of failure are introduced.

94-0025, Rev 0

Add a helper mechanical draft counterflow cooling tower. Install water supply piping from the cooling tower header tie-in to the helper tower and construct a spillway to direct discharge water from the helper tower to the existing cooling tower flume.

The new cooling tower is as reliable as the existing cooling towers and has no negative impact on the function or reliability of the existing cooling towers or any associated equipment. No accident response of any equipment or system is adversely affected by this change. This modification is located in the cooling tower yard well away from any equipment important to safety, and the system being modified does not interact with any equipment considered important to safety. No safety settings or limits, or failure points of any equipment or system addressed in the TSs are affected by this change.

UNIT 2 DESIGN CHANGE REQUESTS

94-0032, Rev 0

Eliminate the steam condensing mode of RHR by cutting out piping and supports, pipe whip restraints and valves associated with the steam condensing mode; installing pipe caps; and removing abandoned piping and instrumentation.

Eliminating the steam condensing mode of RHR does not change the alternatives for decay heat removal, because this mode is not relied upon to provide this function. Using other options does not degrade the ability to cope with any accident or transient. This mode of RHR is not safety related and is not required to bring the reactor to a hot shutdown condition. This modification does not impact the function of any safety-related equipment, and a potential source of failure in a high energy line is eliminated by removing the HPCI to low pressure RHR interface boundary.

94-0034, Rev 0

This GL 89-10 change affects valves and operators in the RHR, CS, HPCI, RCIC, PSW, and recirculation systems by modifying or replacing the actuator, increasing the stroke time, replacing the actuator motor, modifying or replacing the valve, modifying limit switch operation, replacing or bypassing motor thermal overloads, drilling a hole in the flex wedge disc to preclude pressure locking, and deleting one valve.

These modifications improve the capabilities of these safety-related MOVs, thus improving valve reliability. The new and modified components meet the design, material, and construction standards applicable to the system. The valve and system function are unchanged. All response times are within the requirements of all applicable analyses.

94-0035, Rev 0

Replace transmitters, scales, and meters in the nuclear boiler, feedwater control, and turbine EHC systems to increase the maximum operating power level.

This change involves modifications to both safety-related and nonsafety-related components and makes instrumentation compatible with uprated operating parameters. The subject setpoint changes were reviewed and approved under the power uprate program. The seismic and environmental qualifications of the new transmitters remain unchanged. Operation of safety-related equipment and the ability to mitigate design basis events remain unaffected by this modification because the function and logic of the instrumentation are not altered.

UNIT 2 DESIGN CHANGE REQUESTS

94-0036, Rev 0

Revise setpoints and calibration endpoints for the instrumentation affected by power uprate operating parameters.

Operation of safety-related equipment in the nuclear boiler, feedwater control, HPCI, and RCIC systems and the ability to mitigate design basis events are unchanged, because the function and logic of the instrumentation are not altered. The subject setpoint changes and related revisions to the TSs were reviewed and approved under the power uprate program. This modification meets the requirements of applicable Plant Hatch design criteria, and no physical hardware changes are required.

94-0043, Rev 0

Eliminate the need for TSI fire barrier materials by providing minor equipment revisions and making manual action changes in place of cable raceway protection, reroutes cables, and repulls single conductor circuits to preclude spurious actuation, ensuring the capability to accomplish safe shutdown should a fire occur. Make emergency lighting changes to accommodate changes to the manual actions. Reduce the number of SRVs operable from the remote shutdown panel from three to the required two.

This change eliminates the dependency on TSI as a fire barrier to protect safety-related safe shutdown equipment and function without altering or changing the level of compliance. The cable changes do not alter system operation or reliability. The change improves the reliability of safe shutdown components by preventing undesirable spurious actuation. The wiring modifications ensure proper operational modes of safe shutdown components, and the TSI removal reduces the impact of cable aging on cables routed in TSI wrapped raceways.

94-0049, Rev 0

Lower the low water level ATWS-RPT setpoint to aid in preventing thermal stratification and unnecessary thermal cycles resulting from the rapid cooldown of the bottom head region and the reduction in pressure to atmospheric conditions.

No safety-related plant system or component is affected in a manner that could render it more susceptible to failure. GE reviewed the change and determined the results of the ATWS event with the lowered setpoint remain acceptable and indicate that reactor power remains stable with no unacceptable power spikes. The ATWS-RPT aids in maintaining level above the top of the active fuel, and a substantial margin remains after the setpoint reduction. For non-ATWS events, delaying the RPT provides a slight improvement in the present ECCS-LOCA analysis, thereby improving the margin of safety.

UNIT 2 DESIGN CHANGE REQUESTS

94-0052, Rev 0

Install a shroud stabilizer system to structurally replace all nine horizontal shroud welds H1-H8 without credit being taken for actual weld conditions, thereby meeting the requirements of NRC GL 94-03 concerning IGSCC of core shrouds in BWRs.

The stabilizers are designed as safety-related components and assure that the shroud, even if cracked, performs its safety functions. The impacts of the leakage flows through the shroud repair holes and the potential weld cracks in the shroud were evaluated, and the flow is sufficiently small that the steam separation system performance, jet pump performance, core monitoring, fuel thermal margin and fuel cycle length remain adequate. The impact on ECCS performance is insignificant, and hence, the licensing basis fuel peak clad temperature for the normal condition with no shroud leakage is applicable. Seismic loads in the RPV and internal structures are decreased by the shroud stabilizer installation.

94-0054, Rev 0

Replace the existing limit switches for the main turbine stop valves with new seismically qualified, safety-related limit switches less susceptible to contact chatter due to vibrations.

The new switches operate as previously required and do not adversely affect the function or reliability of any other equipment. No new accident mechanisms or failure modes are introduced.

95-0023, Rev 0

Replace the existing fuel grapple head with a new one equipped with a new mast-mounted camera and video monitoring system to improve the ability to handle fuel assemblies and perform verification activities associated with core offload and reload.

The refueling equipment is safety related; however, the mast is classified as nonsafety-related. The permanently mast-mounted camera improves the reliability and performance of the underwater television system. The new television system is seismically installed and treated as safety-related equipment to ensure its reliability. The function of the grapple head remains unchanged, and no new interfaces with safety-related equipment are created.

95-0036, Rev 0

Remove specific safety-related snubbers from reactor recirculation piping, associated portions of RHR piping, and RWCU suction piping to reduce costs and radiation exposure.

UNIT 2 DESIGN CHANGE REQUESTS

All components involved in this modification are safety related. This modification was evaluated and verified using approved programs and methodologies. The reliability of the affected systems and impact to the safety-related function of any system are unchanged. The failure of piping or equipment in these systems would result in the same effects as previously analyzed and thus are bound by existing analyses. All remaining supports affected by this modification were evaluated and found acceptable.

95-0043, Rev 0

Remove the PSW supply and return lines from the seal coolers on the RHR and CS jockey pumps.

Per manufacturer's requirements cooling must be provided for the seal water if the process fluid is above 350°F, but the jockey pump suction does not exceed 200°F. Jockey pump function and operation are not impacted since use of the seal cooler is not necessary. Seal cooling is now provided by the process fluid. The original design criteria of the safety-related RHR, CS, and PSW systems are unchanged. All applicable design codes are met. PSW system capability to provide equipment and area cooling is not adversely impacted.

95-0045, Rev 0

Install larger, slower motors to valves 2E11-F015A&B (which function for LPCI injection, shutdown cooling admission, and reactor and containment isolation) to make gearing changes to the motors and operators; and make spring pack changes to the valves.

This change increases the available thrust and torque to these safety-related valves and assures their function per design. Electrical loading on the LPCI inverter was evaluated and is acceptable, as are the stress and seismic loads associated with the larger motor. The NRC-approved accident analysis and resulting peak clad temperature are not changed. Conformance to 10 CFR 100, 10 CFR 50 App A, and the radiological consequences reported in the FSAR are unchanged.

95-0056, Rev 0

Replace the existing RHR loop B minimum flow bypass valve manufactured by Walworth with a new equivalent valve manufactured by Anchor Darling.

The new safety-related valve meets all the design and code requirements of the existing valve. Therefore, it is equivalent to the existing valve in design requirements, operating characteristics, compatibility with existing equipment, and reliability.

UNIT 1/COMMON REQUESTS FOR ENGINEERING ASSISTANCE/SERVICES

REA 94735, Rev 0

Remove redundant portable fire extinguishers and non-App R emergency lighting units and modify emergency lighting battery surveillance and replacement frequency. Revise FHA to reflect the changes.

The fire protection system is a nonsafety-related, Seismic Category II/I system. Safety-related equipment and systems are designed and/or protected from adverse effects due to operation or malfunction of the automatic fire protection systems. This change does not affect the capability of the affected fire protection components to perform as designed. The replacement cycle of batteries at 6 years before the normal operating life of 7 years enhances system performance and reliability.

RES 92009, Rev 0

Approve the use of the cobalt-free material, NOREM 02 (which was developed as a part of the cobalt reduction program to eliminate highly radioactive Cobalt 60 debris generation), as an equivalent hardfacing material that can be used during routine valve repair without the need for a DCR.

Because the cobalt-free material performs as well as or better than the cobalt-based materials, valve operation and sealing are not impacted. The new alloy has similar physical characteristics and properties and should not impact the MOV sizing or performance results. The material does not adversely affect safety-related equipment or components.

UNIT 2 REQUESTS FOR ENGINEERING ASSISTANCE/SERVICES

REA 94708, Rev 0

Downgrade certain components of the Unit 2 radwaste systems from ASME Code Section III, Class 3, requirements stipulated in RG 1.143. Revise the FSAR to reflect the downgrade.

The radwaste systems equipment and components are nonsafety-related and Seismic Category II. Down grading the systems does not have any adverse effect on the limits of radioactive exposure to the public, or on the function or operation of the systems. All interfaces between the radwaste systems and all safety-related systems were analyzed, and no safety-related components or systems are affected. No new single failure or accident mechanism is added by this modification.

REA 95616, Rev 0

Combine the service and performance tests for the Class 1E batteries into a test controlled by a safety-related site procedure to reduce the time required for testing when both tests are required during the same refueling outage.

The combined test meets the intent of separate tests and yields test results that are conservative compared to the results from separate tests. These safety-related batteries tested by this procedure are capable of performing in an accident situation as well as when tested separately. Since the batteries are out of service when tested, the testing cannot cause a new type of accident. Reliability equals that of the separate tests, and the consequences of a malfunction of equipment important to safety are not increased beyond those resulting from the loss of one redundant dc system as evaluated in the FSAR. The combined test is in compliance with the TSs.

UNIT 1/COMMON MINOR DESIGN CHANGES

93-5051, Rev 0

Install tubing and valves for addition of a second channel analyzer to the reactor coolant sample ion chromatograph, which can be used for online anion (chloride and sulfate) analysis.

The equipment being installed is equivalent to currently installed equipment and meets the design, material, and construction standards applicable to the system. It does not cause the system to be operated outside its normal design or tested limits. The new equipment is not safety related, and its failure would not affect any safety-related equipment. In the event of failure of this equipment and the resulting reactor coolant leakage, PCIVs can be closed and secondary containment would prevent release of radioactive materials.

93-5055, Rev 0

Provide additional work space within the chemistry hot lab by removing one unused fume hood.

The fume hood being removed is not safety related, and its removal does not impact any safety-related equipment. This change does not affect the operation of the other two fume hoods in the lab. All plant effluent monitoring systems and isolation features of the turbine building ventilation system are not affected and will operate as designed.

94-5002, Rev 0

Add small bore control air piping, valves, and supports from the turbine building to the transformer fire protection valve house to replace damaged underground piping and relocate the air supply to the Unit 1 turbine building ventilation controls.

This modification does not degrade any required safety system function or increase challenges to safety systems required to function in the accident mode. This change to nonsafety-related piping maintains the essential instrument air system design characteristics and does not impact any safety system settings or safety limits specified in the TSs. The operation and response of this system are unchanged.

UNIT 1/COMMON MINOR DESIGN CHANGES

94-5041, Rev 0

Install a hardened steel plate between the refueling bridge main hoist sheave pivot arm and the load cell to reduce load cell indicator and controller drift.

This modification does not change any equipment function or operation. While the refueling bridge is safety-related, the controls are not safety-related. This change does not alter the maintenance of any equipment, nor does it change TSs compliance. The addition of the hardened steel plate does not change the setpoints of the load cell.

94-5045, Rev 0

Delete firehose station 1T43-HSR52, which is in the middle of the refueling floor's heavy loads path and relocate firehose station 1T43-HSR01 to a higher, more accessible location within the northwest diagonal.

The affected fire protection equipment is not safety-related. All required compensatory measures are being taken as previously analyzed in the FHA, and the hose station being removed is not required for proper manual suppression coverage. The manual suppression design criteria continue to be met, and the ability to extinguish a fire in a safety-related area remains unchanged.

94-5046, Rev 0

Change the setpoints for the EDGs inboard and outboard bearing temperatures for alarm and shutdown, and for stator temperature alarm and shutdown.

The EDGs are safety related. The new setpoints for the generator inboard and outboard bearing temperatures are provided by the generator manufacturer, and the new stator temperature setpoints are within the range allowed by the EDGs' operating instruction manuals and nameplate data. These changes have no adverse impact on the operation of the EDGs. The setpoints for the EDG temperature switches are not contained in the TSs.

94-5048, Rev 0

Remove the 90° elbow from the discharge end of all four drywell sand cushion drain lines, and the sand from the 2-in. drain line and the sand cushion area to facilitate containment examination.

This change does not affect the containment function or the sand pocket drain system function. The containment system, as analyzed, does not take any credit for the existence of the sand in the sand pocket region. The new configuration of the drain pipe termination in the torus room does not affect the draining capability of the sand pocket region, degrade

UNIT 1/COMMON MINOR DESIGN CHANGES

the concrete shielding structure surrounding the drain pipes, or affect any equipment in the torus room. The containment system continues to meet the ASME Code requirements for the design loadings shown in the FSAR, and no additional loadings or combinations are created.

95-5021, Rev 0

Install two doors on el 185 ft of the Unit 1 reactor building east-west passage to control access when the area is a potential high dose area.

The modification to these nonsafety-related doors does not change the structural integrity of the Unit 1 reactor building or the response of any system assumed to operate in a accident. The doors are being removed from the Unit 1 radwaste area chemical treatment room since controlled access is not required at that location. Doors are not a precursor to any event evaluated in the FSAR. A review of the area revealed no Seismic Category II/I concerns or safety-related equipment which could be impacted in the area of the doors.

95-5041, Rev 0

Isolate the CO₂ hose reel system in the Units 1 and 2 control and turbine buildings and the Unit 2 radwaste building by installing a steel plate prior to the supply valve. Install placards at the CO₂ hose reel stations to indicate they are not operational.

These nonsafety-related CO₂ hose reels are one of several manual fire fighting tools. In the event of a fire in areas and zones where the hose reels are isolated, adequate fire extinguishers and H₂O hose stations are available for manual firefighting.

UNIT 2 MINOR DESIGN CHANGES

93-5011, Rev 0

Add permanent equipment to inject oxygen into the feedwater to maintain the proper concentration of 20 to 50 ppb.

This change is not to the ECCS, reactivity control, or any safety-related equipment assumed to operate in response to an accident. This modification is only to the feedwater system, which is not safety related. The only effect is to maintain the feedwater oxygen level in an approved range beneficial to the piping and not harmful to the reliability of any component. Injection at this point in the feedwater piping does not adversely affect pump or system operation due to the nature and amount of material injected.

94-5019, Rev 0

Terminate several LPM circuits to coax connections in drywell penetrations.

The LPM circuits are nonsafety-related and do not interfere with the safety-related circuits in the penetrations. No safety-related system is impacted by this change. The introduction of nonsafety-related LPM cables into the same penetration with safety-related cables does not reduce the margin of safety, because the LPM cables are extremely low energy circuits and a failure of one LPM circuit would not impact the RPS or essential circuits in the penetrations.

94-5028, Rev 0

Replace the existing wooden roof over the nitrogen storage tank area with a metal roof.

The nitrogen storage tank roof is not a safety-related structure. The new roof structure is more able to withstand extreme wind and live loads than the existing roof.

94-5034, Rev 0

Install tubing and valves on the reactor coolant ion chromatograph for addition of a second channel analyzer, which can be used for on-line anion (chloride and sulfate) analysis.

The nonsafety-related equipment being installed is equivalent to the existing equipment and meets the design, material, and construction standards applicable to the system. This change does not cause the system to be operated outside its normal design or tested limits. The modification does not change, degrade, or prevent the actions described in the FSAR accident evaluations. Operation or failure of this equipment has no effect on the margins of safety.

UNIT 2 MINOR DESIGN CHANGES

94-5035, Rev 0

Remove thermal relief valves 2E11-F3078A&B from the shell side of the RHR heat exchanger and replace them with blind flanges.

Adequate relief protection for the safety-related RHR heat exchangers is provided by the larger 2E11-F055A&B relief valves. The 2E11-F055 valves have the same relieving set pressure and are IST SRVs. The thermal relief valves were supplied with the heat exchanger, but their function is served by the 2E11-F055 valves. This change does not adversely affect RHR system performance and does not have any seismic impact.

94-5040, Rev 0

Change operation of the refueling bridge main hoist normal up light from a geared limit switch to a separate trip-type limit switch.

This modification does not change how the nonsafety-related controls of the main hoist function and does not affect the operation of any other equipment. No new operating modes or methods are introduced. This modification does not change any setpoints involving handling of fuel and does not affect the TSs.

95-5038, Rev 0

Replace the "A" CS pump minimum flow line Walworth swing check valve with a more reliable Enertech flanged nozzle check valve. Relocate the valve to the other side of the elbow in the pump discharge line and into a vertical position away from any flow disturbances.

The safety-related nozzle check valve performs the same function as the swing check valve in that it prevents reversal of flow, thereby ensuring the CS pump is not damaged during operation. The relocation of the valve and its smoother flow characteristics reduce turbulence, thereby reducing erosion of the downstream piping. Because of the equal performance, and improved valve and flow characteristics, the function, operation and reliability of the CS system are not adversely affected.

UNIT 1/COMMON FSAR/FHA CHANGES

13B-001

Revise Unit 1 FSAR section 11.2 and Unit 2 FSAR subsection 10.2.3 to increase the main turbine-generator inservice inspection period to every 8 to 10 years, per GE TIL 1008-3 R1.

The operational durability of a GE-manufactured turbine rated at 1800 rpm was proven by GE to be greater than their declaration. The nondestructive examination used for this nonsafety-related equipment is still in accordance with the manufacturer's recommendations and is, therefore, adequate for inspection purposes. TIL 1008-3R1 upgraded the magnetic particle test to a volumetric (ultrasonic) test as an assured means of confirming turbine integrity.

13B-009

Change section 4.3.4 of the Unit 1 FSAR to increase the limit on heatup and cooldown of the recirculation piping and associated equipment from 100°F/hour to 165°F/hour.

GE evaluated this change against the requirements of ASME Code Section III, NB-3600, and concluded the 165°F/hour cooldown rate is a mild transient for the piping. Code requirements can easily be met for this heatup and cooldown rate. A recirculation pipe break different from a previously analyzed one is not created. The safety-related recirculation system operation is unchanged, and no other safety-related systems or equipment are affected. The heatup and cooldown rates for the recirculation system piping and associated equipment are not addressed in the TSs.

13B-015

Delete Unit 2 FSAR references to the NPPM which originally contained procurement level, requisition review and related instructions applicable to the procurement and material control process that is now contained in corporate and site procedures.

Since the procurement process for plant equipment is not changed, the abilities of plant systems to perform their intended safety functions are not changed. The change does not affect the plant's response to accidents or equipment malfunctions evaluated in the FSAR. The performance of plant equipment and plant operations are not changed. The TSs are not affected.

UNIT 1/COMMON FSAR/FHA CHANGES

13C-015

Revise FSAR paragraph 5.3.3.1 to depict certain valves and dampers in the reactor building ventilation system as being gagged rather than locked closed.

The method of gagging closed the safety-related valves and dampers is equivalent to locking them closed to secure them in the closed position; therefore, the valves and dampers continue to meet the original design requirements. Gagging the valves and dampers does not change the ability of the valves to be isolated for a modified secondary containment. The operation and function of the SGTS are not altered by this change.

13C-022

Revise chapter 4 of the Unit 1 FSAR and chapter 5 of the Unit 2 FSAR to change the method for determining the effects of thermal cycle fatigue on the various RPV components, which can be expressed as a function of the CFUF.

This revision does not represent a design or operation change to any safety-related systems needed for mitigation of accidents. Tracking the stresses and loads on the safety-related RPV in this manner ensures that the RPV will not be operated after having experienced excessive fatigue. This change does not affect any LCO or limiting safety system setting in the TSs. The margin of safety from a fatigue viewpoint is more clear and up-to-date using this methodology.

13D-018

Add the spent fuel shipping canister weighing 700 pounds to the list of items that may be moved over the spent-fuel pool racks in table 10.3-1 of the Unit 1 FSAR.

In full capacity, the safety-related canister is approximately the same weight as a fuel bundle. Therefore, in the unlikely event that the canister is dropped on a rack, the fuel damage sustained would be bounded by the consequences of the fuel handling accident as described in FSAR subsection 14.4.4. The canister is designed per GE specifications and meets the required design criteria for the partial and fully loaded conditions in the spent fuel pool.

UNIT 1/COMMON FSAR/FHA CHANGES

14A-010

Revise the Units 1 and 2 FSARs to make them consistent in referring to the purge and inerting systems as the normal nitrogen supplies for the drywell pneumatic systems.

The supply gas provided to safety-related components in the drywell pneumatic system is identical in quality and pressure to that formerly supplied by compressors, and the purge and inerting system was originally intended to be a safety-related backup supply to the drywell pneumatic system. Safety-related equipment using motive gas from the drywell pneumatic system functions identically. This modification does not impair the operability of any component or system required by the TSs, and the margin of safety is not dependent upon the specific supply of motive gas to the drywell pneumatic system.

F10B-002

Revise FHA App B, tables 1.1-1 and 1.1-2, to accurately reflect the correct status of the doors.

Some doors in App B are shown to be locked but the doors are not required to be locked; therefore the decision was made to maintain them as unlocked. This modification is editorial, since these nonsafety-related doors are not required to be locked. This information is listed in the tables for general information. The TSs do not address the fire protection components associated with this modification.

F10B-004

Revise FHA table 1.4-1 to remove the Unit 1 HPCI room south wall water curtain, which is no longer required for 3-hour separation.

A 10 CFR 50 App R analysis redefined the HPCI room as a zone of the reactor building fire area, and fire rated barriers are not required between zones within the same fire area. This nonsafety-related water curtain does not interface with or affect any equipment important to safety. The TSs do not address the components involved in this change.

F10B-006

Relocate noncommitment-type information contained in the FHA to other plant information sources without affecting commitments or compromising the overall concept of the document; i.e., fire area square footage and combustible loading; appendices containing the fire resistance of concrete block, the combustibility of askarels, and miscellaneous supporting calculations with re-establishment as independent calculations; and combustible loading figures from each area and zone summary.

UNIT 1/COMMON FSAR/FHA CHANGES

The modifications to the FHA are strictly editorial. The information being removed is not required to be documented in the FHA and is found in other site documentation. Although the affected information was removed from the FHA, it has not been altered in the other sources. The guidelines stated in the FHA for the manipulation of this information remain in effect.

F10C-002

Revise FHA App B to identify the spray and sprinkler system valves with the surveillance requirements.

The change is a clarification to ensure the appropriate valves are selected for the surveillance inspections. The system valves are nonsafety-related passive devices and their normal positions are open. The surveillance activity applies to the suppression system isolation valves to ensure the valves remain open to permit water flow in the unlikely event of a fire. The TSs do not address fire protection systems, equipment, or surveillance.

F10C-004

Revise the FHA to adjust the maximum permissible combustible loadings of certain fire zones and areas to provide a reasonable margin from the resident loading to accommodate transient combustibles.

The adjusted loadings were analyzed with their fire area containment capabilities. All modifications were performed within FHA guidelines. The consequences of this change are insignificant, since it is a result of invoking FHA guidelines. This change has no effect on any margin of safety as defined in the TSs.

F10D-002

Delete the reference to TS 6.9.2, which addresses Special Reports and places the reference in the individual FHA sections. Correct a TSs reference to diesel fuel testing acceptance limits, based on the conversion of ITS.

Although different, this fuel oil testing program provides comparable assurances of fuel oil quality to the existing FHA requirements and assures no diminishment in the plant's ability to extinguish a design basis fire. The changes resulting from the TSs conversion have no detrimental effect on plant operation or design and do not reduce compliance with regulatory reporting requirements. The radiological consequences of a previously evaluated accident or malfunction are not increased.

UNIT 2 FSAR/FHA CHANGES

13B-011

Clarify Unit 2 FSAR paragraph 7.6.10.1, which applies to both units, to allow disabling of the RPT until EOC, achieving consistency with the TSs.

No safety-related plant equipment or system is affected by this change, which is only being made to clarify TSs requirements with respect to EOC-RPT. The safety analyses are not changed as a result of this FSAR revision. The EOC-RPT continues to operate as designed and as required per TSs.

13B-022

Correct an error which deleted support 2B21-MSRV-H9 on isometric H-26805 (Unit 2 FSAR figure 3.9-37, sh 2 of 4).

This is a drawing documentation change to correct a drafting error which inadvertently removed a support from the safety-related main steam system. A plant design calculation and an ISI surveillance show the support at the corrected location. No other design documents are affected by this change. No accident analysis previously evaluated in the FSAR is impacted. No new accident scenarios or modes of failure are created.

13D-010

Revise Unit 2 FSAR to reflect that, with the advent of the ITS, many LCOs and other requirements formerly in the TSs now reside in the TRM.

This change to the FSAR is being made only to reference the TRM as a document where many plant requirements are located. No changes are being made to any safety-related systems or procedures. The relocation of existing TSs requirements to the TRM was reviewed and approved by the NRC prior to ITS implementation. The requirements relocated to the TRM are under the control of 10 CFR 50.59.

13D-013

This change adds to the Unit 2 FSAR descriptions and design control measures used for validation for the computer programs, ZENPIPE, CAEPIPE and PS+CAEPIPE.

This revision meets the requirements of 10 CFR 50 App B and GDC 1, as specified in NUREG-0800 and RG 1.70. The use of these programs provides results equally as acceptable as those obtained previously, and designs of new or modified safety-related and nonsafety-related piping systems will not introduce any flaws or nonconservatisms. System performance and integrity are not adversely affected.

UNIT 2 FSAR/FHA CHANGES

13D-016

Clarify the wording in Unit 2 FSAR paragraph 8.3.1.1.3.D to remove the implication that a physical interlock prevents two DGs from being synchronized to offsite power concurrently.

This change does not require a physical modification to the plant or the control logic for the safety-related DG system. It is being made only to clarify an incorrect implication in the FSAR. The function of the DGs to provide emergency power while maintaining redundancy and separation requirements is not altered. The DGs will continue to operate during accident conditions or test conditions as previously designed. There are no modifications to safety limits or margins as contained in the TSs.

13D-022

Revise Unit 2 FSAR subsection 8.2.1 to reflect modification of the 162-MVAR capacitor bank connected to 230-KV switchyard bus 2 to a 117-MVAR bank, and the addition of a 85-MVAR bank to 230-KV switchyard bus 1.

These modifications give more operating flexibility for certain contingencies and improved voltage control during outage conditions. This change to a nonsafety-related switchyard does not increase the likelihood or duration of an LOSP, or decrease the ability to recover from such an event. This change does not adversely affect the function, response, or reliability of any plant equipment. No acceptance limits or failure points of any plant equipment are affected.

14A-001

Revise Unit 2 FSAR paragraph 9.4.10.1.1 to better define the pumps in operation for which ventilation must be maintained by the river intake structure ventilation system does not represent a physical change to the plant or its operation.

The safety-related PSW and RHRSW pumps continue to perform their intended functions with no degradation during prescribed operating conditions. This change is supported by a calculation that defines the number of operating pumps before, during, and after a DBA. This change does not affect any allowable limit or failure point of any safety-related system or equipment.

UNIT 1/COMMON BASES CHANGES

B 3.3.6.1, Primary Containment Isolation Instrumentation

1. The primary containment isolation signals associated with Isolation Group 6 are swapped with those assigned to Isolation Group 11. Reassigning Isolation Group numbers for two Unit 1 Isolation Groups (swapping Groups 6 and 11) is administrative and does not alter the response of any containment barriers to an isolation signal. Initiation of automatic isolation of affected lines penetrating the primary containment whenever monitored variables exceed preselected operational limits is not affected.
2. For consistency between Units 1 and 2, the primary containment Isolation Group numbering scheme adopted for Unit 2 is applied to Unit 1. Due to physical differences between the units, no PCIVs are assigned to Isolation Groups 7 and 12 for Unit 1. Therefore, all references to Isolation Groups 7 and 12 are removed from the Bases. The administrative changes reflect actual plant configuration and do not alter the response of any containment barriers to an isolation signal.
3. Editorial and clarification information is added to the Background section, Applicable Safety Analyses, LCO, and Applicability sections to reflect actual plant configuration and clarify what isolation signals isolate which valves and the applicability of certain time delay relays. The editorial changes reflect actual plant configuration and do not alter the response of any containment barrier to an isolation signal. No unreviewed safety questions were involved.
4. Two isolation valves that are not PCIVs, but receive group isolation signals are added to the Bases for information. Adding the valves does not alter the response of any containment barrier to an isolation signal or alter the physical condition or operation of any PCIV or containment penetration.

B 3.3.3.2, B 3.6.1.3, and B 3.6.4.2, Remote Shutdown System, Primary Containment Isolation Valves, and Secondary Containment Isolation Valves

These editorial changes indicate that certain surveillance tests do not have to be performed during plant outages. The tests can be and have been performed during power operation. The Bases descriptions of the surveillances could be interpreted to require the tests to be performed only during outages. A statement in the Bases description for the surveillances is changed from "The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power." to "The 18 month Frequency is based on consistency with the typical industry refueling cycle." The changes only modify words that could be read as requiring the surveillances to be performed during outages. There is no prohibiting reason the surveillances cannot be performed during power operation. No unreviewed safety questions are involved.

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UNIT 1/COMMON BASES CHANGES

B 3.6, Containment Peak Accident Pressure

These changes are made in support of plant power uprate and are bounding relative to current rated thermal power. The peak accident pressure (Pa) changes from 53.6 psig to 49.6 psig based on analyses made for the power uprate project. The change updates the Bases Section 3.6. The revision does not change any plant equipment or mode of operation. No unreviewed safety questions are involved.

B 3.8.3, Diesel Fuel Oil and Transfer, Lube Oil, and Starting Air, Action F.1

This editorial change clarifies the operational limits for the DG fuel oil, lube oil, and starting air system. The clarifying information does not impact the physical condition or operation of the DGs or the operation of the DG starting air subsystems described in the FSAR.

B 3.8.3, Diesel Fuel Oil and Transfer, Lube Oil, and Starting Air

The change removes redundant and potentially confusing information from the Background section. Two consecutive sentences contain the same information stated in slightly different ways. One of the sentences is deleted. The change does not modify any plant equipment or change any procedures used to operate the equipment. It only removes potentially confusing information and thereby, clarifies the description of the DG lubricating oil system.

B 3.3.3.1, PAM Instrumentation

These editorial changes correct minor typographical errors in the text of the Post Accident Monitoring System LCO and Action E.1. The changes do not alter any plant equipment or procedures.

B 3.6.1.3, SR 3.6.1.3.4, Primary Containment Isolation Valves

This change provides more detailed information regarding the monitoring and surveillance of the TIP shear isolation valves. The change does not modify any plant equipment or operations, or procedures used to operate plant equipment.

B 3.4.7, B 3.4.8, B 3.9.7, and B 3.9.8, RHR Shutdown Cooling System - Hot Shutdown, Cold Shutdown, High Water Level, and Low Water Level

The changes add a paragraph containing clarifying information to more accurately explain TSs LCOs, and differentiate the requirements for RHR shutdown cooling operability and the requirements for forced circulation. The changes do not modify any plant equipment. The changes expand the discussion of the reasons for the LCOs associated with the

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affected TSs and provide for a more accurate understanding of the LCOs. There is no change to the TSs or other documents that direct plant activities.

B 3.8.1, SR 3.8.1.5, AC Sources - Operating

This editorial change clarifies the description of a Note, which modifies the 184-day "fast cold" start surveillance, to indicate that, because of changing bus loads, momentary load transients do not invalidate the surveillance test. The change establishes consistency between the Bases and the SR. No transient or accident analyzed in the FSAR is affected.

B 3.5.1 and B 3.5.2, ECCS - Operating and Shutdown

This editorial change corrects the title of a reference document. Plant equipment is not modified, and plant operations are not affected.

B 3.3.4.1, EOC-RPT Instrumentation

This section discusses TSs Actions A.1 and A.2 EOC-RPT instruments. However, the heading for B 3.3.4.1 indicates it applies only to Action A.1. This editorial change adds Action A.2 to the heading. There is no effect on plant operations.

B 3.5.1, Emergency Core Cooling System - Operating

1. This editorial change corrects a grammatical error associated with the conditions under which SR 3.5.1.6, which requires that the recirc discharge valves be cycled through one complete cycle of travel, must be performed, that is, "...verification during or following MODE 4 > 48 hours and prior to...." instead of "verification during or following each MODE 4 > 48 hours and prior to....". No accident or transient precursors are affected.
2. This change clarifies the description of Action Statements G.1 and G.2 for AC Sources - Operating. The change indicates more clearly when a MODE change is necessary in accordance with the Action Statements. The clarifying information does not alter the intent of TS 3.5.1 or Bases B 3.5.1, nor does it impact the physical condition or operation of any ECCS component.

B 3.8.1, AC Sources - Operating

This change clarifies the requirement of the LCO and eliminates misleading wording. The intent of the requirement is to have, as a minimum, power from both circuits to the F 4-kV ESF bus and power from independent circuits available to the E and G 4-kV buses. The loss of a single circuit still leaves a full ESF division available for safe shutdown. The change clarifies the intent of the offsite power circuits, while eliminating

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wording that requires the plant to be put in jeopardy of a loss of high pressure BOP feed on a turbine trip. The result of the change is a safer plant condition. No unreviewed safety questions were involved.

B 3.3.5.1 and B 3.5.1, ECCS Instrumentation and ECCS - Operating

These changes clarify information relative to the existing number of LPCI Pump Start - Time Delay Relays, the time required for the LPCI to start on demand, and the failure modes for the LPCI Pump Start - Time Delay Relay as presented in the TSs. The changes reflect actual design. They have no affect on either accident initiators or mitigators.

B 3.6.1.8, Suppression Chamber-to-Drywell Vacuum Breakers

This change provides needed clarification concerning which torus/drywell vacuum breakers are required to be functionally tested. It indicates that the vacuum breakers required to be operable are those that are required to be tested. Addition of the clarifying information does not alter the intent of TS 3.6.1.8 or Bases B 3.6.1.8. The change does not impact the physical condition or operation of the vacuum breakers as described in the FSAR.

B 3.5.1, ECCS - Operating

This change provides clarifying information for the Background section relative to the effects of an LOSP on LPCI subsystems. The change is administrative and reflects actual plant configuration. The intent of the Bases is not altered, and the physical condition or operation of the LPCI subsystems is not impacted.

B 3.8.1, AC Sources - Operating

TSs LCO 3.8.1 requires the DG to be capable of starting, accelerating to rated frequency and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals and continue to operate until offsite power is restored. The DGs are required in a variety of initial conditions. A discussion of additional DG capabilities that must be demonstrated to meet required surveillances was inadvertently left out of this LCO, thus the discussion is added. This administrative change does not alter the intent of B 3.8.1, and does not impact the physical condition or operation of the DGs.

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B 3.3.6.1, Primary Containment Isolation Instrumentation

This editorial change revises a reference from "FSAR Chapter 14.4" to "FSAR Chapter 14". The change does not alter the operational condition of any isolation signal.

B 3.3.6.3, Low-Low Set (LLS) Instrumentation

The sequence of the LLS logic was originally written as one reactor pressure transmitter and two trip units. The LLS logic actuates the associated LLS valves only after the arming portion of the LLS logic is satisfied. Arming occurs when one of the 11 SRVs opens. Once armed, each of the LLS valves is opened by a two-out-of-two logic. Accordingly, the Bases description is changed to two reactor pressure transmitters and two trip units to accurately reflect actual plant configuration. The intent of B 3.3.6.3 is not altered and is consistent with NEDE-22224.

B 3.6.1.3, Primary Containment Isolation Valves

These changes add clarifying information to Actions A.1, A.2, B.1, C.1, and C.2 to aid personnel in understanding TS 3.6.1.3. The additional information describes the criteria that a device must meet in order to be utilized for primary containment isolation in the event that an isolation device in the same containment penetration becomes inoperable. The changes do not modify any plant equipment or change any procedures used to operate the equipment.

B 3.5.1 and B 3.5.3, ECCS - Operating and RCIC System

This administrative change revises the description of the Note for SRs 3.5.1.8, 3.5.1.9, 3.5.3.3, and 3.5.3.4 to more accurately explain that SRs are being implemented rather than the note. The change does not alter the intent of Bases B 3.5.1 or B 3.5.3 and does not impact the physical condition or operation of the ECCS or the RCIC System.

B 3.10.8, SHUTDOWN MARGIN Test - Refueling

This change provides additional clarifying information indicating that while in Mode 5 the reactor is depressurized and insufficient pressure is available to scram the control rods; however, the charging water header pressure ensures that, if a scram were to be required, capability for rapid control rod insertion would exist. The change is administrative, and reflects actual plant configuration. The intent of B 3.10.8 is not altered, and the physical condition or operation of the plant is not changed.

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B 3.10.4, Single Control Rod Withdrawal - Cold Shutdown

This editorial change revises the Bases description of the Note for TS 3.10.4 to clarify that the Note refers to a single control rod withdrawal while in Mode 4 rather than while in Mode 3. The change is administrative and does not alter any accident or transient or accident precursors.

B 3.6.1.8, Suppression Chamber-to-Drywell Vacuum Breakers

These changes provide clarifying information relative to the suppression chamber-to-drywell vacuum breaker status and position indication. In some cases the description for the affected TSs is removed from the Bases and placed in the TRM. In other cases it is clarified in the appropriate Bases Action, Surveillance, etc.. The changes are administrative since they only provide additional direction for the determination of vacuum breaker status and vacuum breaker position indicator operability. They provide for a more accurate determination of plant equipment status. No unreviewed safety questions are involved.

B 3.1.1, B 3.5.2, B 3.9.7, and B 3.9.8, SHUTDOWN MARGIN, ECCS Shutdown, RHR - High Water Level, and RHR - Low Water Level

The changes rearrange the wording and add more detailed information to clarify the requirements for secondary containment isolation and acceptable compensatory actions when a primary isolation device in a penetration becomes inoperable. The changes do not alter any plant structures, systems, or components, or change the way plant equipment is maintained and operated. They do not affect any accident or transient precursors.

B 3.8.1, AC Sources - Operating

This administrative change adds clarifying information to the Background section to more clearly explain that while startup auxiliary transformer (SAT) 1C/2C is sized to accommodate the simultaneous starting of all required ESF loads on receipt of an accident, the ESF loads are, in practice, sequenced when being fed through SAT 1C/2C and will be fed through SAT 1C/2C only if 1D/2D fails. The description provided in B 3.8.1 agrees with the description presented in FSAR Chapter 8.

B 3.10.6 and B 3.10.7, Multiple Control Rod Withdrawal - Refueling and Control Rod Testing - Operating

Editorial changes ensure that the Required Actions designations for B 3.10.6 agree with TS 3.10.6, and the SRs designations for B 3.10.7 agree with TS 3.10.7. The changes do not alter the intent of TSs 3.10.6 and 3.10.7, and Bases B 3.10.6 and B 3.10.7. No unreviewed safety questions are involved.

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B 3.1.3, Control Rod OPERABILITY

Administrative changes clarify and make consistent the information contained in the Notes for SR 3.1.3.2 and SR 3.1.3.3 to indicate the two SRs are to be performed 7 and 31 days, respectively, after withdrawal of the control rod and after thermal power is greater than the LPSP. The change adds information to the Bases that is already in the TSs and does not alter the intent of TS 3.1.3 or Bases B 3.1.3.

B 3.7.4 and B 3.7.5, Main Control Room Environmental Control (MCREC) System and Control Room Air Conditioning (AC) System

These editorial/administrative changes more accurately describe the power supply arrangement and the MCREC support function of the air handling units. No accident initiating condition or assumption is affected. No physical changes to the plant will result.

B 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation

More detailed information is added to SR 3.3.2.2.1 to indicate that the feedwater and main turbine high water level trip instrumentation logic Channel Functional Test may be performed from the input of the alarm unit. The only event of concern is the excess coolant inventory transient. Narrowing the scope of the Channel Functional Test for the main and RFPT high instrumentation has no effect on the initiating condition (feedwater controller failure) of that event. No unreviewed safety questions are involved.

B 3.4.8, B 3.9.7 and B 3.9.8, RHR Shutdown Cooling System - Hot Shutdown, High Water Level, and Low Water Level

This administrative changes clarify the RHR shutdown cooling requirements specified in TSs 3.4 and 3.9. Expanded guidance is provided in the specific areas of the definition of the heat exchanger/pump relationships for RHR and RHRSW; differentiation of RHR shutdown cooling subsystems, versus RHR, versus RHRSW; definition of the required capability for an RHR shutdown cooling subsystem; and inclusion of the RHR crosstie valve restriction in Mode 3. This change does not involve any physical, analytical, or operational change.

B 3.3.3.1, Post Accident Monitoring Instrumentation

This administrative revision changes the available range description for the Drywell H₂ and O₂ analyzers in the LCO from 1-10% to 0-10%. The change does not physically alter the equipment or the operation of the equipment. The H₂ and O₂ analyzers are not associated with any accident initiators.

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B 3.3.5.1, ECCS Instrumentation

This editorial change revises LCO 3.5.2 to state that LCO 3.5.2 is not applicable to the HPCI system. The intent of the TSs or the Bases is unchanged.

B 3.3.6.1, Primary Containment Isolation Instrumentation

This change adds a clarifying statement to B.3.3.6.1, applicable Safety Analysis, LCO and Applicability to clarify that the spatial requirements associated with Turbine Building Temperature - High instrumentation is to assure that no unmonitored area should exceed 40 feet in length. Adding the clarifying statement ensures that the premise of the calculation associated with the instrumentation used for event detection is maintained. The change does not modify any plant equipment or change the operation of any equipment associated with the initiation of accidents analyzed in the FSAR. The Turbine Building Area - Temperature monitors will continue to function as designed.

B 3.3.6.3, LLS Instrumentation

The description for Action B, for one or more LLS valves inoperable, of LCO 3.3.6.3 is revised to indicate that Action B refers to an allowance for repair prior to entering Mode 2 or 3 from Mode 4. The editorial change achieves consistency with TS 3.3.6.3.

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B 3.3.6.1, Primary Containment Isolation Instrumentation

1. The primary containment isolation signals associated with Isolation Group 6 are swapped with those assigned to Isolation Group 11. Reassigning Isolation Group numbers for two Unit 1 Isolation Groups (swapping Groups 6 and 11) is administrative and does not alter the response of any containment barriers to an isolation signal. Initiation of automatic isolation of affected lines penetrating the primary containment whenever monitored variables exceed preselected operational limits is not affected.
2. Editorial and clarification information is added to the Background section, Applicable Safety Analyses, LCO, and Applicability sections to reflect actual plant configuration and clarify what isolation signals isolate which valves and the applicability of certain time delay relays. The changes reflect actual plant configuration and do not alter the response of any containment barrier to an isolation signal. No unreviewed safety questions are involved.
3. Two isolation valves that are not PCIVs, but receive group isolation signals, are added to the Bases for information purposes. Adding the valves to the bases does not alter the response of any containment barrier to an isolation signal or alter the physical condition or operation of any PCIV or containment penetration.

B 3.6, Containment Peak Accident Pressure

These changes are made in support of plant power uprate. The peak accident pressure (Pa) changes from 48.7 psig to 45.5 psig based on analyses for the power uprate project. The revision does not change any plant equipment or mode of operation. No unreviewed safety questions are involved.

B 3.3.5.2, RCIC System Instrumentation

This change removes references to the RCIC steam supply line bypass valve 2E51-F119 from the Bases. This valve was removed from the RCIC system by Unit 2 DCR 94-0034, Rev 0.

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In the conversion to TSs Amend 195, many specifications and details of specifications were relocated to the Unit 1 TRM. The following changes were made in the creation of Rev 0 of the TRM.

T 3.0, TRM Specifications

This change provides exceptions for three TRM Specifications to the new LCO 3.0.3 which is added by Amend 195. The exceptions maintain the requirements in existence prior to the changes reflected in the relocation-to-TRM effort and are, therefore, administrative and create no unreviewed safety questions.

T 3.3.1, Reactor Protection System Shorting Links

This change removes the requirement of enhanced (i.e., non-coincident) RPS Functions in MODE 5, unless a control rod is withdrawn from a core cell containing fuel assemblies and SDM for the core configuration has not been demonstrated. Control rod withdrawal from or insertion into core cells without fuel assemblies does not significantly affect core reactivity, and therefore, the RPS scram function serves no purpose. Once SDM is demonstrated, the required shutdown margin (LCO 3.1.1) and the required one-rod-out interlock (LCO 3.9.2) ensure no event requiring RPS will occur. The change does not impact the capability of the RPS system to perform its required function; therefore, no unreviewed safety questions exist.

T 3.3.1, Reactor Protection System Shorting Links

This change allows a controlled insertion of control rods vice initiation of a manual scram in the event control rods are withdrawn from a core that has not had SDM demonstrated and RPS shorting links are not removed. This action decreases challenges to safety related equipment while accomplishing the desired safety function (increasing the margin of safety to inadvertent criticality). This change is consistent with changes per TSs Amend 135 to specifications retained in the TSs. Therefore, no unreviewed safety question results.

T 3.3.3, Non-Type A, Non-Category 1 Post Accident Instrumentation

The CHANNEL CHECK and CHANNEL CALIBRATION frequencies are extended consistent with that approved by the NRC for the safety related PAM instruments retained in the TSs, as well as the Unit 2 requirements.

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T 3.3.8, Offgas Post-Treatment Instrumentation

The Applicability is clarified to be only when there is a potential for excessive fission products in the offgas system; i.e., MODES 1 and 2, only if the flow path from the reactor to the offgas system is not isolated. Accordingly, the requirement to go to cold shutdown is amended as the Applicability will be exited by MODE 3. These changes introduce no unreviewed safety questions.

T 3.4.1, Reactor Coolant System Chemistry

This change specifies time limits for reactor shutdowns required by this specification. This change is administrative and does not constitute an unreviewed safety question.

T 3.6.1, Suppression Chamber-to-Drywell Vacuum Breaker Position Indication

Details for performance of a leakage test are provided. No change in intent or technical requirements is made; therefore, no unreviewed safety question results.

T 3.7.3, Sealed Source Contamination

This change relocates inventory control to plant procedures.

T 3.7.3, Sealed Source Contamination

This change clarifies the statement "startup sources subject to core flux" to include fission detectors. This change is consistent with Unit 2 wording; no unreviewed safety questions result.

T 3.7.3, Sealed Source Contamination

Since no time is specified for leak testing startup sources prior to subjecting to core flux, a 31 day window is added. This is consistent with Unit 2 requirements and with the intent of the previous Unit 1 requirements. As such, the change is administrative and introduces no unreviewed safety questions.

Table T5.0-1, Acceptance Criteria

The acceptance criterion for RPS actuation on main TCV fast closure is changed from <0.03 seconds to 0.08 seconds.

This activity does not change the operation of the TCV fast closure or RPS logic, nor does it change the times in which they must respond to a low EHC oil pressure signal. The oil pressure sensor will continue to respond to a low oil pressure condition in 0.03

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seconds; the RPS will continue to respond to a pressure sensor signal in 0.05 seconds. The TRM requires an acceptance criterion of 0.08 seconds for the combined response times for both paths. No unreviewed safety questions result.

Table T6.0-1, Instrumentation and Controls Required for Remote Shutdown

The equipment required by GDC 19 to be available at the remote shutdown panel is inserted into Table T6.0-1. The change represents more restrictive requirements in that no remote shutdown equipment was previously included in Unit 1 TSs or the Unit 1 FSAR.

Table T6.0-1 specifically defines equipment required for remote shutdown of Unit 1 previously addressed in the FSAR only in a "big picture" manner. The change serves as a clarification to existing FSAR requirements and changes no procedures described in the FSAR. No unreviewed safety questions result.

Table T7.0-1, Primary Containment Isolation Devices

This administrative change deletes valves which are not PCIVs. No unreviewed safety questions result.

Table T7.0-1, Primary Containment Isolation Devices

Notes assigned to various PCIVs are changed to reflect current leakage testing practices. Some notes are modified, some are deleted, and some are added. Changes to the penetration notes do not impact the physical condition of the penetrations or the ability of the penetration to isolate since their function is to describe the leakage test practices of an informational nature. Actual leakage test requirements are addressed by other licensing and implementing documents.

Table T7.0-1, Primary Containment Isolation Devices

Several automatic PCIVs listed in the Unit 1 Table T7.0-1 were not previously assigned to the same Isolation Group as the corresponding Unit 2 automatic PCIVs. For consistency between the two units, the valve assignment practices for Unit 2 are applied to Unit 1. No change in the physical condition or operation of the PCIVs results from the change in Isolation Group designators. The changes are administrative, and no unreviewed safety questions are involved.

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Table T7.0-1, Primary Containment Isolation Devices

As part of the change to adopt the Unit 2 primary containment Isolation Group numbering scheme for Unit 1, the isolation signal descriptions for each Isolation Group are modified. These changes are administrative and do not alter the response of any containment barrier to an isolation signal.

Table T7.0-1, Primary Containment Isolation Devices

These changes are as follows: 1) Several valves are reclassified as nonautomatic isolation valves to reflect their actual containment isolation status; 2) Several valves which receive automatic isolation signals are added; 3) The table is revised to indicate that the drywell pneumatic isolation valves are normally closed and remain closed for isolation. These changes do not alter the physical condition or operation of the affected valves. They are administrative, and no unreviewed safety questions are involved.

The following 50.59 summaries relative to Rev 0 are identical for Unit 1 and Unit 2; therefore, these changes are considered common to both units.

Table T2.1-1, Operability Details for LCO 3.7.4, MCREC System, and LCO 3.7.5, Control Room AC System

This change clarifies the power supply arrangement by which single failure criteria can be met. It recognizes the separate functions of the MCREC System and main control room air conditioning systems, and applies TSs criteria, resulting in direction for Operability determinations. The activity in no way changes the method of system operation as described in the FSAR and introduces no unreviewed safety questions.

T 2.2, Operability Details for LPCI Inverter Room Essential Coolers

This change clarifies the power supply and cooling water supply arrangements by which single failure criteria is met. It embellishes the cooler arrangement as it relates to LPCI system Operability.

The activity in no way changed the method of system operation as described in the FSAR and introduces no unreviewed safety questions.

T 3.0, TRM Specifications

Amend 195 to the Unit 1 TSs and Amend 135 to the Unit 2 TSs approved the relocation of some requirements to plant-controlled documents. Many requirements became TRM Specifications. This change incorporates into the TRM, by reference, the revised TSs of

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Sections 1.0 and 3.0. The incorporation of those revised Specifications and inclusion of new details regarding TSs usage and application are specifically addressed in Amends 195 (Unit 1) and 135 (Unit 2).

T 3.0, TRM Specifications

This global administrative change to the TRM Specifications omits from the TRM those portions of the relocated Specifications that were retained in the TSs. Therefore, all requirements remain, and no unreviewed safety questions result.

T 3.0, TRM Specifications

This activity applies to all previous TSs requirements relocated to the TRM. Those requirements are reformatted into ITS-like format. The editorial reformatting retains the requirements but presents them in a more user friendly fashion. The changes are administrative, and no unreviewed safety questions are involved.

T 3.0, TRM Specifications

This activity applies to certain Notes and LCO statements that are administratively rewritten so that the TRM Applicability contains the plant conditions that require Operability. The changes are administrative, and no unreviewed safety questions are involved.

T 3.0, TRM Specifications

This activity deletes portions of specific requirements that are overly complex in providing excessive detail or state redundant requirements. The changes are editorial and do not change the intent or meaning of the requirements; therefore, no unreviewed safety questions result.

T 3.0, TRM Specifications

This change adds the Note, "Separate Condition Entry is allowed for each [Function/channel/...]," to the Actions of several Specifications to provide more explicit instructions for proper application of the Actions for TRM compliance. Consistent with TS 1.3, "Completion Times," approved in Amends 195/135, this ACTIONS Note provides direction consistent with the intent of the existing ACTIONS. This change is administrative and creates no unreviewed safety questions.

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T 3.0, TRM Specifications

This change adds more restrictive requirements to several TRM LCOs. The more stringent requirements do not result in any operation that will increase the probability of initiating an analyzed event. They continue to ensure process variables, and SSCs are maintained consistent with the safety analyses and licensing bases.

T 3.0, TRM Specifications

This activity removes allowances for several Specifications, thereby imposing more restrictive requirements. The more stringent requirements do not result in any operation that will increase the probability of initiating an analyzed event. They continue to ensure process variables SSC are maintained consistent with the safety analyses and licensing bases.

T 3.3.2, Control Rod Block Instrumentation

As approved in similar applications of TSs Amends 195 (Unit 1) and 135 (Unit 2), this change removes the "prior to a reactor startup" Frequency from the control rod block instrumentation. The normal Frequency (along with SR 3.0.4) ensures that the verification is performed within that Frequency prior to plant startup. The removal of redundant surveillance requirements does not result in an unreviewed safety question.

T 3.3.3, Non-Type A, Non-Category 1 Post Accident Instrumentation

This change revises the Required Actions for inoperable monitoring instruments that are not restored to service with the allowed out of service time. Due to the passive function of this instrumentation and the operator's ability to respond to events utilizing alternate instruments and methods for monitoring, some Required Actions are overly stringent. With the nominal times for restoration exceeded, this change requires a special report to the SRB, allowing proper review and attention by this operational oversight committee to assure necessary and prudent actions are taken.

T 3.3.3, Non-Type A, Non-Category 1 Post Accident Instrumentation

This change limits the required Applicability for the post accident monitoring instrumentation to those MODES during which events requiring the function of these instruments are significant; i.e., MODES 1 and 2. This is consistent with the MODE requirements of the Type A, Category 1 Post Accident Monitoring Instrumentation retained in the TSs.

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T 3.3.3, Non-Type A, Non-Category 1 Post Accident Instrumentation and T 3.3.9, Offgas Hydrogen

The wording "submitting a report to the NRC" is replaced with "reporting to the Hatch SRB." This ensures increased management attention when the TRM instrumentation is not restored to OPERABLE in the allowed repair time. Since this change has no effect on either accident initiation or mitigation, no unreviewed safety question results.

T 3.3.7, MCREC System Instrumentation

Rather than requiring a unit shutdown when MCREC System instrumentation channels are inoperable and untripped, the TRM is modified to require placing the system in the pressurization mode of operation. System initiation places the MCREC System in the status where the instrumentation has performed its function, and no further action is necessary.

T 3.3.9, Offgas Hydrogen

The specification is clarified to define 14 days as the allowed time for submission of a Special Report to the SRB.

T 3.3.9, Offgas Hydrogen

The Applicability is changed. Limiting hydrogen concentration and providing monitoring for hydrogen only serves a purpose when there is a potential for excessive hydrogen in the offgas system; i.e., MODES 1 and 2, and then only if the flow path from the reactor to the offgas system is not isolated.) Making the TRM Applicability consistent with the conditions when monitoring and limiting this parameter does not result in an unreviewed safety question.

T 3.3.9, Offgas Hydrogen

The hydrogen sampling requirements are rewritten for clarification and format consistency. The change is administrative and does not introduce an unreviewed safety question.

T 3.3.11, Main Steam Line Radiation Instrumentation

The allowed out of service time for the Isolation Instruments is extended to 24 hours. The evaluations applicable to and approved for the TSs changes are applicable to the MSL Radiation - High Function. Therefore, no unreviewed safety question is introduced.

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T 3.3.3, Non-Type A, Non-Category 1 Post Accident Instrumentation and T 3.3.9, Offgas Hydrogen

The wording "submitting a report to the NRC" is replaced with "reporting to the Hatch SRB." This ensures increased management attention when the TRM instrumentation is not restored to OPERABLE in the allowed repair time. Since this change has no effect on either accident initiation or mitigation, no unreviewed safety question results.

T 3.3.7, MCREC System Instrumentation

Rather than requiring a unit shutdown when MCREC System instrumentation channels are inoperable and untripped, the TRM is modified to require placing the system in the pressurization mode of operation. System initiation places the MCREC System in the status where the instrumentation has performed its function, and no further action is necessary.

T 3.3.9, Offgas Hydrogen

The specification is clarified to define 14 days as the allowed time for submission of a Special Report to the SRB.

T 3.3.9, Offgas Hydrogen

The Applicability is changed. Limiting hydrogen concentration and providing monitoring for hydrogen only serves a purpose when there is a potential for excessive hydrogen in the offgas system; i.e., MODES 1 and 2, and then only if the flow path from the reactor to the offgas system is not isolated.) Making the TRM Applicability consistent with the conditions when monitoring and limiting this parameter does not result in an unreviewed safety question.

T 3.3.9, Offgas Hydrogen

The hydrogen sampling requirements are rewritten for clarification and format consistency. The change is administrative and does not introduce an unreviewed safety question.

T 3.3.11, Main Steam Line Radiation Instrumentation

The allowed out of service time for the Isolation Instruments is extended to 24 hours. The evaluations applicable to and approved for the TSs changes are applicable to the MSL Radiation - High Function. Therefore, no unreviewed safety question is introduced.

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T 3.3.11, Main Steam Line Radiation Instrumentation

This is an administrative change which clarifies the requirements for setpoint adjustment based upon increases in hydrogen injection.

T 3.4.1, Reactor Coolant System Chemistry

This change is administrative in that it eliminates redundant wording for conductivity monitoring. There is no technical change, and no unreviewed safety question results.

T 3.4.2, Structural Integrity

The ACTIONS are rewritten to encompass the same intended ACTION regardless of the operating condition of the unit. Previously, ACTIONS were written assuming the unit is shutdown; no ACTION was provided for the unit in operation when a Class 1, 2, or 3 component is discovered to not have integrity. Since these changes are editorial, no unreviewed safety questions are introduced.

T 3.6.1, Suppression Chamber-to-Drywell Vacuum Breaker Position Indication

This change provides ACTIONS for inoperable position indication to assure vacuum breaker closure for continued operation. The TSs for each unit contain the required ACTIONS, including unit shutdown, for any vacuum breaker discovered to be open. The combination of shift verification and a periodic confirmatory test provides a level of assurance equal to that provided by the single OPERABLE indication and periodic confirmatory test previously allowed. As such, no unreviewed safety questions result.

T 3.6.1, Suppression Chamber-to-Drywell Vacuum Breaker Position Indication

Since the TRM deals only with indication, not with vacuum breaker OPERABILITY, an ACTION is added to direct the user to the TSs when a vacuum breaker is determined to be open.

Table T7.0-1, Primary Containment Isolation Devices

This administrative change relocates PCIVs previously in the TSs to Table T7.0-1, changing the format to include additional information. The relocation of the valves is approved by Amend 195/135.

The change does not alter the ability of any penetration to isolate. No unreviewed safety question results.

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Table T7.0-1, Primary Containment Isolation Devices

Changes made to correct various editorial errors and omissions. No unreviewed safety questions are involved.

Table T7.0-2, Primary Containment Isolation Devices

This change creates Table T7.0-2 to provide a concise reference for the operators to use in determining the applicable penetration number for a given PCIV MPL number. The table was developed from information contained in Table T7.0-1, Primary Containment Isolation Devices - Testable Penetrations. This administrative change develops Table T7.0-2 strictly as an operator aid. It does not alter the ability of any penetration to isolate.

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Table T3.3.5-1

This change does not involve any physical modification to the plant. The RCIC system allowable high and low flow rates are revised to ensure the RCIC minimum flow valve operates within a range that will protect the pump from overheating and result in timely injection. The minimum flow line valve is opened when flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect to pump. The setpoint is intended to be high enough to ensure the pump flow rate is sufficient to protect the pump, yet low enough to ensure the closure of the minimum flow valve is initiated to allow full flow into the core. This change increases reliability of the RCIC function and does not negatively impact any accident precursors. No unreviewed safety questions are involved.

The following 50.59 summaries relative to Post Rev. 0 are identical for Unit 1 and Unit 2; therefore, these changes are considered common to both units.

Table T5.0-1, SR 3.1.4.3, Table T3.3.2.1, T 3.7.1, T 3.3.3, T 3.3.7, and Table T7.0-1

Changes are made to correct obvious typographical or editorial mistakes. These changes do not involve any physical modification to the plant or affect any procedure used to operate plant equipment. No accident or transient precursors are affected; no unreviewed safety questions are involved.

UNIT 1/COMMON TRM CHANGES

Table T10.3-1, Qualified Post Accident Monitoring Instrumentation

This change does not involve any physical modification to the plant or any procedure used to operate plant equipment. 1/2P33-P601A&B are removed from the "Indicator" column for Function 7 (Drywell H₂ Concentration) and Function 8 (Drywell O₂ Concentration). Under the "Indicator" column for Function 7, 1P33-R0604A is changed to 1P33-R604A. 1/2P33-P601A&B are MCR H₂/O₂ analyzer control panels, rather than indicators. Removing the panel MPLs and making an editorial correction to an indicator MPL in no way affects any accident precursors. No unreviewed safety questions are involved.

Table of Contents, Table T10.1-1, and Table T10.2-1

These editorial and administrative changes do not involve any physical modification to the plant or affect any procedure used to operate plant equipment. No unreviewed safety questions are involved.

Table T8.2-1, Secondary Containment Devices

These administrative changes, which correct omissions from Rev 0 of Table T8.2-1 and delete information for the deleted Type D containment, do not involve any physical modification to plant equipment or affect any procedure used to operate plant equipment.

Tables T10.1-1, Master Equipment Cross Reference-Sorted by MPL, and T10.2-1, Master Equipment Cross Reference-Sorted by Specifications

These editorial changes do not involve any physical modification to the plant or affect any procedure used to operate plant equipment. The editorial revisions correct typographic errors and inadvertent omissions from the Rev 0 Tables.

T 1.2, Definitions

This change does not involve any physical modification to the plant or affect any procedure used to operate plant equipment. The proposed change clarifies the definition of "Operation with the Potential to Drain the Reactor Vessel" as applicable only when fuel is in the vessel. No unpreviously analyzed accidents or transients would be initiated nor is the mitigation of any accident or transient affected. Therefore, no unreviewed safety questions are created.

UNIT 1/COMMON TRM CHANGES

T 3.3.2, Control Rod Block Instrumentation

These changes do not involve any physical modification to the plant or affect any procedure used to operate plant equipment. Notes have been added to SRs 3.3.2.1 and 3.3.2.2 to specify that 12 hours are allowed following mode changes before the SRM and APRM channel functional tests are required to assure operability of the control rod block instrumentation. Since no DBAs or transients take credit for control rod block signals initiated by the SRMs or APRMs, no unreviewed safety questions are created.

T12.0, Safety Function Determination Program

These editorial changes do not involve any physical modification to the plant or affect any procedure used to operate plant equipment. Changes to sections 2.1, 2.3, and 4.2 are administrative changes to correct typographical errors. Revisions to sections 3.0 and 5.2 provide more accurate information. Accident or transient precursors are not affected; no unreviewed safety question are involved.

Specification T 3.3.10, Turbine Overspeed Protection

This change does not involve any physical modification to the plant or affect any procedure used to operate plant equipment. The change to the TSR 3.3.10.3 clarifies the requirement to "cycle each turbine control valve (TCV) through at least one complete cycle of full travel" to appropriately read "cycle each TCV through at least one cycle of travel from its open position to full closed." This change is consistent with current interpretation and satisfies the intent of GE-recommended testing in that each valve is demonstrated not to be bound and the fast acting solenoid of each valve is exercised. For FSAR Section 15.1.1, Generator Load Rejection - Turbine TCV Fast Closure, the scenario starts with the TCV at a partially open position corresponding to 105% of rated steam flow operating in the full arc steam admission mode. Therefore, cycling the TCVs from a partially open position does not involve an unreviewed safety question.

Specification T 3.3.4, Traversing Incore Probe (TIP) System

These changes do not involve any physical modification to the plant or affect any procedure used to operate plant equipment. The surveillance frequency for TIP normalizations is changed from once every 31 EFPDs (~ 744 EFPHs) to once per every 1000 EFPHs. The time for completing Required Action B is changed from 38.75 EFPHs to 1250 EFPHs. No DBA or transient described in the FSAR is initiated by the functioning or failure of the TIP system, including its detectors, drive mechanism, signal processing equipment, or the collection or interpretation of the data which are a measure of the 3-D power distribution in the core. No unreviewed safety question is involved.

UNIT 1/COMMON TRM CHANGES

Table T2.1-1, Operability Details for LCO 3.7.4, MCREC System, and LCO 3.7.5, Control Room AC System

This editorial change does not involve any physical modification to the plant or affect any procedure used to operate plant equipment. Reformatting Table T2.1-1 provides more delineation between sections and additional text to enhance readability. The technical content of the table is unchanged. No unreviewed safety question is involved.

T 2.2, Operability Details for LPCI Inverter Room Coolers

These changes do not involve any physical modification to the plant or affect any procedure used to operate plant equipment. Revisions to Section T 2.2, Operability Details for LPCI Inverter Room Essential Coolers, acknowledge the existing backup heat sink availability to the LPCI inverter room coolers and do not affect any accident precursors. No unreviewed safety questions are involved.

Table T3.3.2-1, Control Rod Block Instrumentation

These changes do not involve any physical modification to plant equipment. The changes delete Table T3.3.2-1 Function 3.a, APRM Flow Biased Simulated Thermal Power - Upscale Control Rod Block, and specify that plant procedures be revised to treat the APRM flow-biased rod block equation as a nominal trip setpoint rather than an allowable value. Since the flow-biased APRM rod block function only serves as an aid to plant operations, it is not assumed to function to either prevent or mitigate the consequences of any accident or transient. This function is redundant to other rod block functions required by the TSs and is, therefore, deleted from the TRM. However, because of its value in preventing scrams, it remains operable in the plant with setpoint determination and testing conducted under control of plant procedures. For the same reason, the flow-biased values of the APRM rod block may be treated as nominal trip setpoints in the plant procedures.

T 8.3, Secondary Containment Expansion

This change does not involve any physical modification to the plant or change any procedures used to operate plant equipment. The change adds Section T 8.3, Expanding Secondary Containment, to the TRM to define the method by which additional volume may be included in an existing Secondary Containment Type. The expansion of the Type of Secondary Containment to include additional volume is considered necessary in certain circumstances. One obvious example occurs during refueling outages when it is necessary to include the drywell airspace volume within the boundaries of the Secondary Containment Type. That is generally a transient condition to allow the drywell manways to be closed and sealed for vessel and cavity flooding. These hybrid containment configurations are not specifically addressed as Secondary Containment Types in the TRM. Hence it is desirable to define the criteria by which the operability of secondary

UNIT 1/COMMON TRM CHANGES

containment can be judged when expanded to include an additional airspace. Compliance with TSs will be verified for the new volume, and secondary containment Function will be maintained. No unreviewed safety questions are involved.

T 3.0, TRM Specifications

This change does not involve any physical modification to the plant or change any procedures used to operate plant equipment. Section T 3.0 of the TRM indicates that TSs Section 3.0, "Limiting Conditions for Operability Applicability and SRs Applicability," applies to the TRM with certain exceptions. One of the exception is "LCO 3.0.6, regarding support/supported system ACTIONS is not applicable to TRM Specifications". This change adds: "However, when an inoperable TS support SSC provides support to a TRM SSC, which in turn supports a supported SSC addressed in TS, LCO 3.0.6 remains applicable". It is not acceptable to delay entry into TSs Conditions and Required Actions for an SSC, crediting actions taken external to TSs. It is not the intent of the statement to preclude transitioning through the TRM in the support/supported relationship for the case where an inoperable TSs support SSC provides support to a TRM SSC, which in turn supports a supported SSC addressed in TSs. The change represents a more conservative approach by clarifying the applicability of LCO 3.0.6. When the initiating entry into 3.0.6 is an inoperable TSs SSC, it is intended for LCO 3.0.6 to apply to the cascading inoperabilities, including those which transition through the TRM. Since the change clarifies the allowed use of LCO 3.0.6 and is conservative, no unreviewed safety questions are involved.

T 12.0, Safety Function Determination Program

This change does not involve any physical modification to the plant or change any procedures used to operate plant equipment. Sheet 1 of Attachment 1 to T 12.0 states; "If LCO 3.0.6 is entered, perform Support SSC LCO Required Actions, initiate Supported SSCs completion time tracking, AND Enter SFDP". That statement contains information that is inconsistent with the philosophy of the Safety Function Determination Program (SFDP). To provide more accurate information, "initiate Supported SSCs completion time tracking" is deleted from the statement. There is no requirement in the body of the SFDP to initiate completion time tracking clocks on each supported SSC if no loss of function exists, nor is the program intended to require the tracking of the supported systems from the time of the support system inoperability. The change corrected the SFDP flowchart to make the direction consistent with LCO 3.0.6 and Bases, as well as the body of the SFDP. The change accurately reflects the philosophy of the SFDP and TSs LCO 3.0.6. No unreviewed safety questions are involved.

UNIT 1/COMMON TRM CHANGES

T 8.1, Secondary Containment Types

These changes do not involve any physical modification to the plant or procedures used to operate plant equipment. The allowable combinations of SGT subsystems to be operable for Type B1 Containment are changed from "any three of the 4 subsystems" to the following two combinations - "1A, 2A, 2B" and "1B, 2A, 2B". Secondary containment drawdown tests performed at Plant Hatch show that for Type B1 containment (Zones 1 and 3), testing utilizing only two Unit 1 SGT subsystems provides unsatisfactory results. For combinations for 2A, 1A, 1B and 2B, 1A, 1B and assuming that the Unit 2 subsystem fails as the selected single failure, the remaining subsystems would not be adequate to perform the required drawdown. Similarly Type B2 containment requirements are changed to require all four SGT subsystems to be OPERABLE. The changes provide more restrictive requirements for SGT system operability and do not effect any accident precursors. No unreviewed safety questions are involved.

UNIT 2 TRM CHANGES

In the conversion to TSs Amend 135, many specifications and details of specifications were relocated to the TRM. The following changes were made in the creation of Rev 0 of the Unit 2 TRM.

T 3.3.1, Reactor Protection System Shorting Links

This change removes the requirement of enhanced (i.e., non-coincident) RPS Functions in MODE 5 unless a control rod is withdrawn from a core cell containing fuel assemblies and SDM has not been demonstrated for the core configuration. Control rod withdrawal from or insertion into core cells without fuel assemblies does not significantly affect core reactivity, and therefore, the RPS scram function serves no purpose. Once SDM has been demonstrated, the required shutdown margin (LCO 3.1.1) and the required one-rod-out interlock (LCO 3.9.2) ensure no event requiring RPS will occur. The change does not impact the capability of the RPS system to perform its required function; therefore, no unreviewed safety questions exist.

T 3.3.1, Reactor Protection System Shorting Links

Per TSs Amend 135, this change allows a controlled insertion of control rods instead of initiating a manual scram in the event control rods are withdrawn from a core that has not had SDM demonstrated and RPS shorting links are not removed. This action decreases challenges to safety-related equipment while accomplishing the desired safety function (increasing the margin of safety to inadvertent criticality). No unreviewed safety question results.

T 3.3.2, Control Rod Block Instrumentation

The change eliminates the applicable conditions for the APRM rod block function during MODE 5 except during SDM demonstrations. APRMs are not necessary for safe operation of the plant while operating in MODE 5 with the mode switch in "Refuel". Additionally, the APRM rod blocks neither initiate any accidents or transients, nor are they credited for mitigation of any accident or transient in the omitted conditions. Therefore, this change results in no unreviewed safety questions.

T 3.3.2, Control Rod Block Instrumentation

Per the TSs, during spiral offload or reload in MODE 5, SRMs outside of the fueled region are no longer required to be OPERABLE, since they are not capable of monitoring neutron flux in the fueled region of the core.

UNIT 2 TRM CHANGES

T 3.3.4, Traversing Incore Probe System

This change applies the 25% surveillance interval extension of SR 3.0.2 to the TIP normalization interval. This is a change in presentation only, with no change in intent or requirement.

T 3.3.4, Traversing Incore Probe System

This change allows the use of substitute values from symmetric TIP locations, or from calculations performed by the on-line computer core monitoring system in the event of TIP system inoperabilities. This is identical to the specification approved by the NRC for Hatch 2, Cycle 12. (Cycle specific approval was a conservative measure with no technical basis for the exclusion of other cycles.) This change does not affect the consequences of plant transients because total core TIP reading (nodal power) uncertainty remains less than 8.7%.

T 3.3.6, Seismic Monitoring Instrumentation

This changes the requirement for submitting a report to the NRC to submitting a report to the SRB. This is consistent with direction provided by the NRC, while maintaining an appropriate level of management attention to inoperable equipment. This change does not constitute an unreviewed safety question.

T 3.3.6, Seismic Monitoring Instrumentation

These changes clarify the OPERABILITY requirements of the monitors and the Required Actions for inoperable or event-actuated monitors. The changes are administrative, with no change in intent and or technical content.

T 3.3.7, MCREC System Instrumentation

The option to isolate the main steam lines is captured in the TRM by the administrative rewrite of the Applicability, which allows the MSIVs to be closed with no further action since the Applicability is exited. This change is administrative, since there is no change in technical content or intent.

UNIT 2 TRM CHANGES

T 3.3.7, MCREC System Instrumentation

The Applicability is revised from MODE 5 to only during those operations which have potential to create a need for the system to operate; i.e. during movement of irradiated fuel assemblies, CORE ALTERATIONS, and during OPDRV. The omitted conditions are not initiators for events which require the system, and therefore, the change is not an unreviewed safety question.

T 3.3.7, MCREC System Instrumentation

The allowed out of service time for the isolation instruments is extended to 24 hours as approved in TSs Amend 135 for comparable ECCS and isolation instrumentation. No unreviewed safety question is introduced since this extension was reviewed for Amend 135.

T 3.3.10, Turbine Overspeed Protection

Changes the frequency of testing the main turbine control valves to once/92 days and changes the frequency of testing the main stop valves, reheat stop valves, and turbine intercept valves to once/31 days. GE missile probability calculations conclude that there is no increased risk associated with the changes. The requirements continue to ensure OPERABILITY of the Turbine Overspeed system; therefore, no unreviewed safety question exists.

T 3.3.10, Turbine Overspeed Protection

Changes the frequency for disassembly and inspection of the turbine valves from 60 months to 6 years based upon manufacturer's recommendations and GE TIL 820-4. Per GE the re-inspect cycle is based upon accumulated operation versus a calendar duration. No unreviewed safety questions are introduced.

T 3.3.11, Main Steam Line Radiation Instrumentation

This change combine the actions for inoperable channels, independent of whether one or both trip systems are affected. This allows the conservative action of tripping the inoperable channels and introduces no unreviewed safety questions.

UNIT 2 TRM CHANGES

T 3.4.1, Reactor Coolant System Chemistry

This change revises the requirement for submitting a report to the NRC to submitting a report to the SRB. This is consistent with direction provided by the NRC, while maintaining an appropriate level of management attention to inoperable equipment and out of specification parameters. This change does not constitute an unreviewed safety question.

T 3.9.2, Communications

Removes a redundant "prior to" Frequency. The remaining 12-hour Frequency ensures (along with SR 3.0.4) that the verification is performed within 12 hours prior to the start of CORE ALTERATIONS. Removal of these "prior to" Frequencies was approved in other similar applications in the Unit 2 TSs Amend 135.

Table T6.0-1, Instrumentation and Controls Required for Remote Shutdown

This change redefines the instrumentation, controls, and transfer switches required to be OPERABLE from the Remote Shutdown Panel as those required to meet the GDC 19 scenario defined in FSAR 7.5.1.4. As compared to previous Specifications, additional equipment is added, while some is omitted. This is a more restrictive change and creates no unresolved safety questions.

Table T7.0-1, Primary Containment Isolation Devices

This change correctly relocates several valves in Table T7.0-1 which receive automatic isolation signals but which are not listed in the automatic isolation valve section.

Table T7.0-1, Primary Containment Isolation Devices

This change swaps the isolation valves assigned to Isolation Group 6 with those assigned to group 11. Also the 18-inch containment purge and vent valves assigned to Isolation Groups 6 and 10 are reassigned to group 2.

The change is administrative and does not alter the response of any containment barriers to an isolation signal. The change is consistent with the isolation signals assigned to individual valves in Unit 2 FSAR Table 6.2-5 and does not alter the physical condition or operation of any PCIV or containment penetration.

UNIT 2 TRM CHANGES

Post-Rev 0 Changes

Section T 3.7.1, Snubbers, and Table T7.0-1, Primary Containment Isolation Devices - Testable Penetrations

These editorial changes do not involve any physical modification to plant equipment or change any procedures used to operate plant equipment. In Section T.3.7.1, the Condition A Note is made applicable to Required Action A.2 rather than Required Action A.1. In Table T7.0-1, Notes D.a and D.k are revised to clarify that MSL high radiation provides only annunciation signals, and isolation signals are provided by MSL high-high. No accident or transient precursors are affected, and no unreviewed safety questions are involved.

Table T7.0-1, Primary Containment Isolation Devices - Testable Penetrations

This change modifies 2E11-F015A&B by increasing the size of the valve motors, changing the gear sets, and replacing the spring pack. This change is made in compliance with NRC GL 89-10 which recommends that owners of light water reactors develop a program that provides the necessary assurance that safety-related MOVs will function, either opening or closing, when subjected to the maximum (worst-case) dP across the valve during normal operation and abnormal events within the design basis of the plant.

The modification increases the stroke times of the valves. The new stroke times do not alter the analysis assumptions and inputs regarding the ECCS (LPCI) response time. The stroke time used in the LOCA analysis is still valid, and the NRC approved accident analysis and resulting PCT do not change. The valves are not an input to the safety analyses associated with radiological consequences of an accident. Electrical loading on the LPCI inverter, and the stress and seismic load associated with the larger motor are acceptable.

Tables T10.1-1 and T10.2-1

These editorial changes do not involve any physical modification to the plant or change any procedures used to operate plant equipment. References to the "2B DG" are changed to "B DG" because the B DG has a Unit 1 MPL no. and is referred in other sections of these tables as the "B DG". Some instrumentation listed as applying to the "1A DG" and "1C DG" are changed to apply to "2A DG" and "2C DG". The changes do not affect any previously evaluated accident precursors or create any new accident precursor.

UNIT 2 TRM CHANGES

Table T5.0-1, Acceptance Criteria

This change does not involve any physical modification to the plant or change any procedures used to operate plant equipment. Three notes are added to Table T5.0-1 to eliminate certain specific instruments response time testing and rearrange the order of some of the notes. The rearrangement of existing notes is strictly administrative since the text of the notes is not changed. Adding the new notes is supported by the NEDO-32291, "System Analyses for Eliminating Selected Response Time Testing Requirements," which was approved by the NRC as indicated in the SER in support of Licensing Amend 137. No unreviewed safety questions are involved.

Section T 3.9.3, Refueling Crane and Hoist

This change does not involve any physical modification to the plant or change any procedures used to operate plant equipment. The setpoint for the Refueling Crane and Hoist slack cable cutoff for the main hoist is changed from 100 lb to $50 \text{ lb} \pm 25 \text{ lb}$. The slack cable interlock prevents excess cable being dispensed from the drum. The $50 \text{ lb} \pm 25 \text{ lb}$ are consistent with vendor manual SX-28057, Rev 5. The slack cable cutoff is not associated with dropping a fuel bundle as analyzed in the FSAR. It only provides indication the load is being carried by the mast and grapple.

Section 11.0, Loss of Function Diagrams, and Tables T10.1-1 and T10.2-1

These numerous editorial and administrative changes do not involve any physical modification to the plant or change any procedures used to operate plant equipment. No unreviewed safety questions are involved.

Table T7.0-1, Primary Containment Isolation Devices-Testable Penetrations

This TRM change is made in support of DCR 94-013 which adds a new penetration assembly in the spare nozzle designated as MPL No. 2T52-X1041. The penetration reroutes IRM and SRM cables for improved cable separation. The cables are routed through stainless steel conduit to provide separation within the assembly. Vertical metal barriers are installed in both the inboard and outboard junction boxes to enhance channel separation. LEMO connectors are installed at the detector end of all SRM and IRM cables.

The change to the SRM and IRM circuits in the RPS neither prevents the reactor from being scrambled on appropriate RPS inputs nor prevents a control rod withdrawal block. The new penetration assembly meets or exceeds all applicable codes and standards. The neutron monitor circuits are not adversely affected by the replacement LEMO connectors and the additional LEMO quick connect cable splices. The penetration is seismically

UNIT 2 TRM CHANGES

qualified and the shielding is equivalent to that of other existing penetrations. No unreviewed safety questions are involved.

Table T7.0-1, Primary Containment Isolation Devices-Testable Penetrations

This change is made in response to DCP-034-0-001 which modifies MOVs 2B31-F031A&B, 2E11-F028A&B, 2E21-F031B, 2E41-F001, 2E41-F006, 2E51-F007, 2E51-F045, 2E51-F046, 2E51-F119, and 2P41-F115A&B. Valve 2E51-F119 is deleted from the RCIC system, and associated piping and power supplies are removed. The functions performed by this valve are performed by the new 2E51-F045 Anchor/Darling valve. Valves 2P41-F115A&B are modified by drilling a 1/4 inch hole in the upstream side of the flex wedge disc to eliminate potential valve disc pressure locking. The valve stroke time is increased for the remainder of the valves, and the motors for some of the valves are replaced with larger motors.

This change is in compliance with NRC GL 89-10 which recommends that owners of light water reactors develop a program that provides the necessary assurance safety-related MOVs will function, either opening or closing, when subjected to the maximum (worst-case) dP across the valve during normal operation and abnormal events within the design basis of the plant.

Table T5.0-1, Acceptance Criteria

These changes do not involve any physical modification to the plant or change any procedures used to operate plant equipment. The change deletes specific response time testing requirements associated with the MSL drain and reactor sample valves, the drywell to torus dP valves, and the mechanical vacuum pump trip. The deletion of the response time testing is supported by NEDO-32291 which was approved by the NRC as indicated in the SER in support of Licensing Amend 137. No unreviewed safety questions are involved.

Tables T10.1-1 and T10.2-1

DCP 94-054-0-003 replaces TSV limit switches with new qualified switches to improve reliability, and assigns new MPL nos. to the new switches. The current switches are experiencing intermittent contact chatter due to vibration during power increases. The new limit switches satisfy operational requirements, and the function of the switches and valves is not altered. No unreviewed safety questions are involved.

UNIT 2 TRM CHANGES

Tables T5.0-1, T6.0-1, T7.0-1, T10.1-1, and T10.2-1

These changes are made in support of DCR 94-034 which provides for the modification of various MOVs to accommodate the operating parameters of GL 89-10 and for increased margin to operate during a design basis event. Valve stroke times for certain primary containment valves are increased, and valve logic is changed.

The valve modifications satisfy GL 89-10 requirements, and no unreviewed safety questions are involved.

UNIT 1/COMMON TESTS OR EXPERIMENTS

95-001 Confirm proper operation of the logic modifications made to the SGTS per
Rev 0 DCR 1H95-006, Rev 0.

The test ensures at least three SGTS trains (one of Unit 1 and two of Unit 2) are always operable, either in auto or running, and are available to perform their safety function. The procedure manipulates logic or controls of the operable Unit 1 SGTS train, when necessary, to either cause a response of the inoperable train or to ensure the operable train continues to run. Unit 2 SGTS trains are auto-initiated with all system responses unchanged. The normal ventilation systems shut down, isolate, and the applicable Secondary Containment Isolation Dampers close when the SGTS trains receive an automatic initiation signal. The possibility of temperature rises exists in the Units 1 and 2 RWCU rooms and in the Steam Chase resulting from the loss of normal ventilation. Consequently, temperatures are monitored when the Units 1 and 2 SGTS trains auto-initiated. The functional testing of that subsection is terminated if unacceptable temperature rises are observed and then resumed, as determined by the Unit 1 and/or Unit 2 Shift Supervisors, when area temperatures permit. Operation of the Units 1 and 2 trains is similar to existing approved surveillance procedures and does not inhibit the system's response. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR, are not increased by this activity.

One Unit 1 and two Unit 2 SGTS trains are capable of performing their intended safety function. Manipulation of the logic or controls of the applicable operable Unit 1 SGTS train is independently verified to ensure logic operability is not degraded. The Unit 2 SGTS trains are only manipulated to cause a normal auto-initiation system response. The procedure does not operate the Units 1 and 2 trains any differently than their existing approved system operating procedures or surveillances. This ensures the operable trains are capable of performing their intended safety function. Therefore, this activity does not create the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

The procedure adheres to TSs requirements by initiating Required Actions for the applicable inoperable Unit 1 SGTS train prior to and during the test. This ensures the applicable operable Unit 1 SGTS train is operable throughout the test. Independent verification of logic/controls manipulation is required to ensure logic operability is not degraded. The operable SGTS trains are capable of performing their intended safety function. Therefore, the margin of safety as defined in the TSs is not reduced.

UNIT 1/COMMON TESTS OR EXPERIMENTS

95-002 Evaluate the ability of specific combinations of SGTS trains to vent
Rev 0 Secondary Containment, given specific containment boundaries specified by
TSs and the TRM, with and without the Unit 1 refueling floor hatch
installed. Validate some of the assumptions used in developing
configurations in the TSs and the TRM.

The SGTS trains are aligned in specific combinations to evaluate the ability to draw vacuum in the Secondary Containment. Normal ventilation is secured during the testing. Temperatures in the Main Steam Chase and the RWCU room for Unit 1 and Unit 2 are monitored throughout the testing to ensure the testing does not significantly increase the risk of actuating ESFs, including a scram. Testing is stopped and the ventilation restored if temperatures in these rooms exceed predetermined limits. The Secondary Containment boundary is not changed from that described in the FSAR. The SGTS trains are operated (started and aligned) manually so that in the event of an accident, the automatic initiation signal will override the manual operation and align the system as described in the FSAR. Previous evaluations concluded that accident analysis are not invalidated. Varying the combinations of trains running does not affect the reliability of the affected components beyond that previously evaluated in the FSAR. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased by this activity.

The Secondary Containment boundary will not be changed from that described in the FSAR. The SGTS trains are operated (started and aligned) manually so that in the event of an accident, the automatic initiation signal will override the manual operation and align the system as described in the FSAR which depends on the Secondary Containment isolation dampers being closed. Therefore, this activity does not create the possibility of an accident nor the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

The procedure tests Secondary Containment configurations specified in the TSs. Secondary Containment and SGTS remain operable during the testing. TSs regarding the Secondary Containment and SGTS are complied with throughout performance of the procedure. Therefore, the margin of safety as defined in the TSs is not reduced.

UNIT 1/COMMON TESTS OR EXPERIMENTS

95-004
Rev 0

Provide information allowing GE to predict how NMA will affect reactor coolant chemistry requirements and radiation dose rates in the main steam system, when operating under NWC and HWC conditions. The data from these injection tests are fundamental to the overall design and ultimate implementation of the NMA process which lowers dose rates by providing an environment in the vessel which requires less hydrogen be injected to protect the reactor vessel internals from intergranular stress corrosion cracking. The test provides instructions for injecting ammonia under NWC and HWC conditions. Low concentrations of ammonia are injected into the feedwater system while varying the hydrogen injection rate.

The installation and operation of ammonia injection equipment does not alter any initial inputs, conditions, or assumptions for the probability of accidents. Water chemistry parameters, conductivity, chloride, and pH are maintained within acceptable levels to eliminate possible adverse effects on reactor vessel materials and fuel components. This test does not adversely affect any system needed to mitigate or prevent radioactive releases to the environment. Although an increase in activity is not expected for the MSLRM, this procedure provides controls to terminate ammonia injection if the main steam activity increases by a factor of two above normal background. The change in reactor water chemistry is controlled by maintaining the conductivity < 0.4 uS/cm. By maintaining the reactor water conductivity within an acceptable level, any effect on the reactor vessel internals and other fuel components is eliminated. All safety systems are able to function as designed. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

This activity does not require any safety system or component important to safety to operate outside its normal designed function. No new accidents are introduced, since all equipment is operated as designed and as specified in present plant procedures. The ammonia is converted to nitrate in NWC conditions and the NH_3 ion under HWC conditions. There are no adverse effects on the fuel and other vessel internals. Water chemistry parameters, conductivity, chloride, and pH are maintained within acceptable levels to eliminate possible adverse effects on reactor vessel materials and fuel components. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the FSAR.

UNIT 1/COMMON TESTS OR EXPERIMENTS

The ability to safely shut down the plant or the prevention/mitigation of an accident as described in TSs is not impacted during the installation or operation of the ammonia injection equipment. The procedure requires termination of ammonia injection if the reactor conductivity is $> 0.4 \text{ uS/cm}$. This conductivity limit ensures the chloride TRM specification limit is not exceeded due to physical chemistry relationship between conductivity, chloride, and pH. Therefore, the margin of safety as defined in TSs is not reduced.

UNIT 2 TESTS OR EXPERIMENTS

- 95-003 Identifies the source of unexplained leakage from the Unit 2 Primary Containment
Rev 0 utilizing helium leak detection technology.

Adding helium (an inert, non-toxic, non-reactive gas) in very small concentrations does not significantly increase containment pressure. Control room habitability is not impacted by the injection of helium into, or its leakage from, the primary containment. Containment venting requirements prior to personnel entry and normal reactor building ventilation capabilities ensures sufficient oxygen is present in the breathing air. Helium, which does not react with plant components contacted, has no effect on the likelihood of a malfunction of any equipment important to safety. Introducing helium in the concentrations and pressure as described has no impact on the failure mode of any equipment. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

Introducing helium into the primary containment does not adversely affect its integrity, operation, or reliability of any equipment therein. The operation or reliability of any SSC with which the helium comes into contact with is not adversely affected because of its inert, non-toxic, and non-reactive characteristics and its use in such low concentrations. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

No failure point is lowered, and no acceptance limit is increased. Therefore, the margin of safety as defined in the TSs is not reduced.

- 95-005 Collect hot spots temperature data for MSIVs and SRVs to provide a better basis
Rev 0 for establishing the qualified life of the MSIV pilot solenoid assemblies and the SRV solenoid valves. CITMs are placed in the Unit 2 Drywell and Steam Chase to collect temperature data of potential hot spots for the current operating cycle.

A totally passive device, the CITM, does not impact the function or reliability of any plant equipment and has no affect on any failure modes. It does not affect any of the precursors for any of the accidents previously evaluated in the FSAR. The CITM is rigid and has adequate structural integrity to withstand the loads resulting from the simultaneous application of 100g acceleration in three orthogonal directions. As a passive device able to withstand accident conditions, it has no affect on any system or component assumed to operate in response to an accident. This device requires no operation, maintenance, or surveillance of any kind. The consequences of a malfunction of any equipment important to safety will remain the same as presently analyzed. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment, important to safety previously evaluated in the FSAR is not increased by this activity.

UNIT 2 TESTS OR EXPERIMENTS

The CITM is a passive device which performs no safety-related function. Based on its configuration and seismic qualification by Westinghouse, the device is structurally adequate for any credible seismic event. It can be mounted to equipment/components weighing more than 1/2 lb on the MSIVs, SRVs, RTDs, or attached conduit with no adverse effect on these components. This passive device does not affect the function or reliability of any equipment and does not introduce any new failure modes to any equipment. The device requires no operation, maintenance, or surveillance of any kind. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

The CITM is designed to withstand all environmental and seismic conditions it is expected to encounter. It is a totally passive device which does not interface with any plant equipment or systems. This device requires no operation, maintenance, or surveillance of any kind. It has no affect on any acceptance limits or failure points for any system or component addressed in the TSs.

95-006
Rev 0

Determine the capability or capacity of one of the RFPs by increasing power level until the RFP no longer maintain water level. Determine whether the unit can withstand a RFPT trip at or near 100% rated power.

Although this test results in a RFP being operated at or near its maximum capability, several administrative controls are contained in the Special Purpose Procedure to offset any increase in the probability of the RFPT tripping and the loss-of-feedwater event occurring. Power increases are small due to PCIOMR limitations at the power levels at which this test is performed. Therefore, the point at which power (steaming rate) exceeds RFP capacity (feeding rate) is approached slowly and is exceeded only a small amount. The rate of water level decrease due to power exceeding RFP capacity is expected to be slow. Should reactor vessel water level begin to decrease, power is reduced quickly to a point known to be within the capability of the operating RFP. This procedure requires reactor water level to be increased approximately 4 in. above its normal operating value to provide more time before the scram setpoint is reached. Additional parameters, such as suction pressure and turbine speed, related to RFP performance are included as test termination conditions. If any limit is exceeded, the test is terminated, even if water level can be maintained.

No systems or components are modified or placed in an abnormal state in order to perform this test. The HPCI and RCIC systems are required to be operable prior to performing this test should one RFP not be sufficient to maintain water level. The other RFP is readily available for injection should it be needed to prevent a scram.

UNIT 2 TESTS OR EXPERIMENTS

The test affects only one RFP and its associated RFPT. These components are not safety related or important to safety. The RFP and RFPT are not assumed to mitigate the consequences of any postulated transient or accident. No event resulting from this test could be worse than the transient already analyzed. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

The test does no more to the RFPT than increase its speed to its maximum capability as defined by the test termination conditions. No systems or components are modified or placed in an abnormal state, either temporarily or permanently, in order to perform this test. Emergency makeup sources assumed in the Unit 2 FSAR transient analysis for a loss-of-feedwater event are available. No other activities which could combine with this test to make a loss-of-feedwater-flow event different than that analyzed are allowed. Because the test determines the maximum capability of an RFP, the possibility the RFPT may trip as it neared its point of maximum capability exists. The possibility exists the reactor vessel water level may decrease if power level exceeds RFP capacity. Therefore, this activity will not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

No SSCs covered by the TSs, or those which support structures, systems, or components covered by the TSs, are affected by this test. No safety limits, limiting safety system settings, setpoints, LCOs, required actions, or operability requirements are changed or are required to be changed to perform this test. TSs requirements for the RPS low reactor vessel water level scram, HPCI and RCIC initiation, and RRS pump trip on an ATWS signal are unaffected by the performance of this test. No required instrumentation or equipment is rendered inoperable by or for this test. RFP operation, except for the RFPT trip on high reactor vessel water level, is not covered by the TSs. The high water level setpoint is not changed or defeated by or for this test. Therefore, the margin of safety as defined in the TSs is not reduced.

95-007
Rev 0

Perform the Simplified Heat Rate Test to document steam/feedwater cycle operation and assess changes in unit efficiency as part of the power uprate effort.

This activity requires data acquisition/testing in four operating/test conditions: 1.) normal operation at various power levels, 2.) normal operation with steam/feedwater leaks exiting the cycle isolated, "isolated cycle operation, 3. operation with No. 4 TCV just closed and isolated cycle, and 4.) normal operation at rated thermal power with varying levels of condenser vacuum.

The objectives are to: 1.) confirm the heat rate and electrical generating capability of the unit after power uprate, 2.) obtain baseline turbine performance at the new

UNIT 2 TESTS OR EXPERIMENTS

100% power level for comparison with future performance tests, 3.) provide on-line uprate information on the unit in parallel with plant instrumentation, 4.) determine the change in unit performance as a result of the unit outage, 5.) determine the performance loss associated with suspected leaking valves within the turbine cycle, and 6.) quantify the effects of condenser pressure and cooling water temperatures on heat rate and unit electrical output.

The only applicable accident analysis is loss of feedwater heating. The only event in the FSAR which has applicability is overpressurization of the condenser. The procedure governing this activity relies on the alarm functions and places limitations on unit operation designed to ensure the alarm function is not encountered. The procedure ensures FSAR evaluations are maintained by imposing precautions and limitations. One precaution is the posting of appropriate personnel to monitor and respond to configurations which may decrease equipment response. Another limitation is imposed to prevent configurations which could impact the FSAR (i.e., isolating feedwater heater high level lines). The possible rupture of the condenser due to low vacuum levels is prevented by relying on alarms and trips, and limiting the vacuum to a level with sufficient operating margin from the alarm point. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased by this activity.

The procedure governing this activity takes precautionary measures of having I&C and Operations personnel stationed locally should a heater high level control function be isolated. Limitations are placed on the number of adjacent heaters which may have high level control functions isolated. Therefore, this activity will not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

Should a function, such as condenser vacuum low, become inoperable the portion of this activity dependent on that function will not occur. These actions and management/shift oversight ensure the margin of safety as described in the TSs is maintained.

95-008
Rev 0

Confirm the proper operation of the RHR Interlocks which could establish a drain path from the reactor vessel to the torus when they do not function as designed.

The procedure contains provisions to prevent the loss of vessel inventory and maintain the unit in the cold shutdown condition. While in the cold shutdown condition, there is no credible event involving radiological release from the reactor vessel, and primary or secondary containment. The procedure contains provisions to prevent loss of coolant inventory and maintain the ability to rapidly flood up the vessel. The RHR systems are inoperable for the LPCI function. Both CS systems

UNIT 2 TESTS OR EXPERIMENTS

are available for automatic injection should they be required. The CS systems, arguably important to safety in this condition, are unaffected. No other equipment important to safety are affected by this procedure. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased by this activity.

The procedure maintains the plant in an analyzed condition, namely cold shutdown. Provisions are included to ensure draining of reactor vessel inventory through the RHR system piping does not occur. The isolation of the shutdown cooling suction and discharge paths are maintained during the performance of this procedure to ensure the RHR system does not experience a malfunction of a different type which might result in the draining of inventory from the reactor vessel. The LPCI function of RHR is unavailable. However, this is allowed by the TSs since both loops of CS are operable. No systems other than the RHR system are affected by this procedure. Therefore, this activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR.

TSs allow for shutdown cooling to be out of service for a period of time for testing or other purposes. If this time is exceeded, certain actions must be complied with. These requirements are discussed as part of a pretest brief. The required TSs actions are complied with during the procedure. The procedure ensures required actions are met by ensuring reactor vessel inventory will support natural circulation. The procedure ensures the RWCU is in service for temperature indication and decay heat removal. At least one recirculation system is available for forced circulation. CS is operable. Therefore, the margin of safety as defined in the TSs is not reduced.

DATA TABULATIONS AND UNIQUE REPORTING REQUIREMENTS

OCCUPATIONAL PERSONNEL RADIATION EXPOSURE FOR 1995

This section satisfies the requirements of Edwin I. Hatch Nuclear Plant Units 1 and 2 reporting requirements and assures compliance with the Code of Federal Regulations. The methods in RG 1.16 were used to report all exposure of personnel at this facility. The effectiveness of the plant radiation program is exhibited by the large number of individuals with no measurable exposure or with minimal dose.

The time period covered by this tabulation is from January 1, 1995, through December 31, 1995. Individual exposures indicated by Electronic Direct Reading Dosimeters (EDRDs) were recorded daily with use of an ALARA Computer System. These exposures and the remaining dose margin were tabulated, printed, and posted on a daily basis and upon individual request. The EDRD results, recorded in dosimetry computer files, were supplanted by thermoluminescent dosimeter measurements made over a period of approximately one calendar quarter as the data became available.

Each person listed in the dosimetry computer files was assigned a usual job category based on daily activities. The five job categories are identified in the following table. Running totals of doses acquired in each of these categories were placed in each person's dosimetry file. Each dosimeter reading was added for individual exposure records and to the total representing the cumulative dose in the appropriate job category.

The implicit assumption involved in this method of accounting for exposure in different tasks is that all exposure acquired in job categories other than those mentioned above is documented by a Radiation Work Permit.

Further delineation regarding the number of persons and the amount of exposure to individuals in different job categories and by various personnel categories is indicated by the standard reporting format of RG 1.16. Each personnel dosimetry file contains the personnel category information required to accomplish the record keeping. The individual running dose totals for each job are used by the ALARA Computer to compute the number of man-rem indicated in each group. Backup disc files and hard copy records are maintained.

1995 ANNUAL OPERATING REPORT EDWIN I. HATCH NUCLEAR PLANT

Regulatory Guide 1.16 Report Information for 1995
Georgia Power Company - Nuclear Generation
Plant E.I. Hatch - Licensee: DPR-57 , NPF-5

Task and Job Function	No. of Personnel > 100 mRem			Total Man-Rem		
	Station	Utility	Contract	Station	Utility	Contract
REACTOR OPERATIONS AND SURVEILLANCE						
Maintenance + Construction	32	0	0	12.622	.042	3.088
Operations	67	0	0	34.843	.000	.000
Health Physics + Laboratory	62	4	21	20.765	1.173	6.668
Supervisory + Office Staff	10	0	0	4.044	.008	.536
Engineering Staff	2	0	0	1.169	.016	.232
Maintenance + Construction	176	13	325	112.209	3.371	143.386
Operations	10	0	0	3.719	.000	.000
Health Physics + Laboratory	14	1	3	4.221	.139	1.886
Supervisory + Office Staff	25	0	9	12.481	.056	3.050
Engineering Staff	11	0	8	3.902	.078	2.396
INSERVICE INSPECTION						
Maintenance + Construction	4	0	20	.777	.112	15.391
Operations	17	0	0	4.326	.000	.000
Health Physics + Laboratory	7	3	15	3.558	1.663	6.733
Supervisory + Office Staff	0	0	0	.030	.038	.010
Engineering Staff	0	0	1	.143	.050	.429
SPECIAL PLANT MAINTENANCE						
Maintenance + Construction	29	0	53	14.597	.006	31.465
Operations	0	0	0	.248	.000	.000
Health Physics + Laboratory	1	0	0	.618	.028	.465
Supervisory + Office Staff	3	0	0	2.097	.000	.019
Engineering Staff	0	0	3	.046	.000	1.649
WASTE PROCESSING						
Maintenance + Construction	4	0	0	1.552	.000	.230
Operations	1	0	0	.397	.000	.000
Health Physics + Laboratory	2	2	3	1.378	.325	1.646
Supervisory + Office Staff	1	0	0	.330	.000	.032
Engineering Staff	0	0	0	.009	.000	.051
REFUFLING OPERATIONS						
Maintenance + Construction	0	0	59	.025	.037	16.537
Operations	0	0	0	.555	.000	.000
Health Physics + Laboratory	0	0	9	.096	.013	2.302
Supervisory + Office Staff	1	0	0	.252	.000	.071
Engineering Staff	0	0	4	.100	.000	1.443
SITE TOTALS						
Maintenance + Construction	245	13	457	141.782	3.568	210.097
Operations	95	0	0	44.088	.000	.000
Health Physics + Laboratory	86	10	51	30.636	3.341	19.700
Supervisory + Office Staff	40	0	9	19.234	.102	3.718
Engineering Staff	13	0	16	5.369	.144	6.200
Grand Totals:	479	23	533	241.109	7.155	239.715
Site Man-Rem Total: 487.979						

REACTOR COOLANT CHEMISTRY

REACTOR COOLANT CHEMISTRY

Monthly tabulations of SJAE isotopic values and reactor coolant parameters are found in the following tables. Unit 2 values are also shown, although it is not required they be reported. Isotopic values listed as "0" are less than the lower limit of detection for the counting system.

1995 ANNUAL OPERATING REPORT EDWIN I. HATCH NUCLEAR PLANT

UNIT 1
1995
SJAЕ ISOTOPICS
uCi/SEC

DATE 1995	MWT	Xe-133	Xe-135	Xe-138	Kr-85m	Kr-87	Kr-88	Σ 6
Jan 5	2436	0.5	5.9	146	1.4	8.8	5.0	168
Feb 2	2436	0.5	5.9	146	1.3	8.6	5.8	168
Mar 2	2436	0.4	6.1	153	1.3	7.8	4.8	174
Apr 1	2436	17	61	241	16	52	46	434
May 1	2436	17	72	596	21	87	70	863
June 2	2436	24	77	592	21	86	68	868
July 3	2436	19	76	436	19	64	58	671
Aug 2	2436	19	71	252	15	37	45	439
Sep 2	2436	57	178	945	39	116	119	1362
Oct 2	2436	56	199	1352	44	155	136	1942
Nov 2	2436	65	223	2583	52	210	159	3291
Dec 1	2435	64	271	4023	60	300	212	4930

REACTOR CHEMISTRY

IODINES uCi/ml							
DATE 1995	MWT	I-131	I-132	I-133	I-134	I-135	DEI-131
Jan 5	2436	2.28E-6	1.29E-4	4.26E-5	5.16E-4	1.06E-4	3.61E-5
Feb 3	2436	0	1.24E-4	3.70E-5	5.25E-4	9.99E-5	3.17E-5
Mar 2	2436	0	1.41E-4	4.53E-5	5.66E-4	1.21E-4	3.71E-5
Apr 6	2436	0	1.30E-4	4.35E-5	5.30E-4	1.78E-4	3.52E-5
May 4	2436	2.10E-6	1.35E-4	4.84E-5	5.36E-4	1.39E-4	4.07E-5
June 8	2436	0	1.51E-4	4.46E-5	6.39E-4	1.13E-4	3.78E-5
July 5	2434	2.06E-6	1.40E-4	4.47E-5	5.45E-4	1.20E-4	3.84E-5
Aug 2	2436	4.47E-6	1.85E-4	5.53E-5	5.82E-4	1.29E-4	4.67E-5
Sep 6	2434	6.32E-5	7.13E-4	2.09E-4	1.63E-4	3.82E-4	2.05E-4
Oct 4	2436	7.43E-5	1.18E-3	3.34E-4	3.00E-3	6.54E-4	3.12E-4
Nov 1	2434	1.24E-4	1.93E-3	6.42E-4	5.74E-3	1.34E-3	5.76E-4
Dec 6	2436	1.24E-4	3.70E-3	1.16E-2	1.30E-2	2.94E-3	1.04E-3

UNIT 2
1995
SIAE ISOTOPICS
uCi/SEC

DATE 1995	MWT	Xe-133	Xe-135	Xe-138	Kr-85m	Kr-87	Kr-88	Σ 6
Jan 2	2436	20	274	6470	46	278	171	7260
Feb 6	2436	19	263	6209	48	295	176	7010
Mar 3	2436	17	240	5928	41	257	135	6619
Apr 3	2436	20	258	6063	45	276	168	6830
May 10	2436	6.8	206	5986	45	278	159	6681
June 2	2435	16	239	5554	40	257	128	6233
July 3	2433	14	239	6143	42	266	137	6840
Aug 2	2377	13	203	4895	31	223	129	5495
Sep 1	2177	13	205	5001	34	220	129	5603
Oct 2	0	0	0	0	0	0	0	0
Nov 2	0	0	0	0	0	0	0	0
Dec 2	2506	9	167	3788	27	179	99	4269

REACTOR CHEMISTRY

IODINES uCi/ml							
DATE 1995	MWT	I-131	I-132	I-133	I-134	I-135	DEI-131
Jan 6	2436	6.94E-5	2.97E-3	1.23E-3	1.22E-2	3.30E-3	9.92E-4
Feb 3	2436	4.70E-5	1.96E-3	8.97E-4	8.48E-3	2.19E-3	6.86E-4
Mar 3	2436	5.88E-5	3.03E-3	1.25E-3	1.32E-2	3.17E-3	9.95E-4
Apr 7	2436	4.77E-5	2.93E-3	1.07E-3	1.32E-2	2.84E-3	9.04E-4
May 12	2433	5.72E-5	2.15E-3	1.12E-3	1.40E-2	2.84E-3	9.13E-4
June 3	2436	7.68E-5	3.10E-3	1.06E-3	1.54E-2	2.71E-3	9.61E-4
July 5	2433	1.01E-4	3.24E-3	1.10E-3	1.50E-2	2.72E-3	9.96E-4
Aug 2	2377	6.23E-5	3.15E-3	1.14E-3	1.41E-2	3.20E-3	9.90E-4
Sep 6	2157	8.92E-5	2.90E-3	1.79E-3	1.79E-2	4.61E-3	1.36E-3
Oct 4	0	2.66E-6	2.84E-5	5.94E-7	0	0	3.84E-6
Nov 1	0	1.82E-6	0	4.45E-6	0	0	3.02E-6
Dec 6	2556	5.89E-5	1.96E-3	8.97E-4	8.46E-3	2.34E-3	7.10E-4