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Southern Nuclear Operating Company
the southern electric system

April 29, 1996

10 CFR 50

Docket Numbers: 50-348
50-364

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

Joseph M. Farley Nuclear Plant (
Revision 13 To The Updated FS
And 10 CFR 50.59 Report

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.71(e), Southern Nuclear Operating Company (SNC) hereby submits Revision 13 to the FNP, Units 1 and 2, Updated FSAR. This submittal contains all the changes necessary to reflect information and analyses submitted to the Commission by SNC or prepared by SNC pursuant to Commission requirement from April 25, 1994, to November 4, 1995 (i.e., from one day after the end of the 12th Unit 1 scheduled refueling outage to the end of the 13th Unit 1 scheduled refueling outage). As required, this submittal is provided on a replacement-page basis, and is accompanied by an effective page list that identifies the current pages of the FSAR following page replacement. Pursuant to 10 CFR 50.4(b)(6), the original and 10 copies are being submitted to the Commission's Document Control Desk, and a copy is also being supplied to the NRC Region II office and to the NRC Resident Inspector at FNP.

Also, in accordance with the requirements of 10 CFR 50.59(b)(2), SNC hereby submits a report summarizing changes, tests and experiments made or conducted at the FNP, Units 1 and 2, as reflected in Revision 13 to the Updated FSAR. Pursuant to 10 CFR 50.4(b)(1), the original is being submitted to the Commission's Document Control Desk, and a copy is also being supplied to the NRC Region II office and to the NRC Resident Inspector at FNP.

If there are any questions, or if additional information is needed, please advise.

Respectfully submitted,

D Morey

Dave Morey

DRC/r13-fsar.doc

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cc list and enclosure list: See next page.

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cc for FNP FSAR Revision 13 (Enclosure 1):

U. S. Nuclear Regulatory Commission, Washington, DC
Mr. B. L. Siegel, NRR Senior Project Manager (cover letter only)
Document Control Desk (10 complete packages)

U. S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Region II Administrator (1 complete package)
Mr. T. M. Ross, FNP Senior Resident Inspector (1 complete package)

cc for the 10 CFR 50.59 Report (Enclosure 2):

U. S. Nuclear Regulatory Commission, Washington, DC
Mr. B. L. Siegel, NRR Senior Project Manager

U. S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Region II Administrator
Mr. T. M. Ross, FNP Senior Resident Inspector

Enclosures
1. FNP FSAR Revision 13
    - Replacement Pages (8.5x11 pages and 11x17 pages)
    - Insertion Instructions (with 8.5x11 pages)
    - Effective Page List (with 8.5x11 pages)
2. 10 CFR 50.59 Report

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

UNIT NUMBER 1 -- DOCKET NUMBER 50-348
UNIT NUMBER 2 -- DOCKET NUMBER 50-364

REVISION NUMBER 13
TO
THE UPDATED FINAL SAFETY ANALYSIS REPORT
FOR
JOSEPH M. FARLEY NUCLEAR PLANT
UNIT NUMBERS 1 AND 2
UNDER THE ATOMIC ENERGY ACT OF 1954

Southern Nuclear Operating company hereby files Revision 13 to its updated Final Safety Analysis Report for Farley Units 1 and 2.

This revision consists of updated pages for the Final Safety Analysis Report.

SOUTHERN NUCLEAR OPERATING COMPANY

By D Morey

Dave Morey
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 29th DAY OF April, 1996

Martha Gayle Dow

Notary Public

My commission Expires: November 1, 1997

ENCLOSURE 1

FNP FSAR REVISION 13

REPLACEMENT PAGES

INSERTION INSTRUCTIONS

EFFECTIVE PAGE LIST

ENCLOSURE 2

10 CFR 50.59 REPORT

JOSEPH M. FARLEY NUCLEAR PLANT (FNP)
10 CFR 50.59 REPORT
REPORT FORMAT AND CONTENT

As required by 10 CFR 50.59(b)(2), this report includes a brief description of changes, tests, and experiments, including a summary of the safety evaluation of each, as reflected in Revision 13 of the Farley Nuclear Plant (FNP) Final Safety Analysis Report (FSAR).

Each change, test, or experiment is listed with an "identifier." This identifier is typically the design change number used by Southern Nuclear to implement the change. The identifier may also include the specific revision of the document, for example, "R0" in the list would indicate "Revision 0." Some design change numbers may also have a "B" for Bechtel Power Corporation as the responsible design organization, "S" for Southern Company Services as the responsible design organization, or "P" for an FNP site developed change. For procedure changes the identifier is typically the title of the procedure, for example, FNP-0-AP-3 for an administrative procedure. Other miscellaneous identifiers are also used.

The "Description" section for each entry contains a brief description of the change, test, or experiment being reported.

The "Safety Evaluation Summary" contains a summary of the safety evaluation developed to support the determination that no unreviewed safety question was involved. This summary was developed solely for use in this report -- no activities important to safety have been based on these summaries.

This report is sorted alpha-numerically by identifier. As FNP has a "shared" FSAR, this report contains all changes, tests, and experiments affecting either or both FNP units.

For additional information concerning any change, test, or experiment included in the report, contact Southern Nuclear.

Identifier	Description	Safety Evaluation Summary
ABN 93-0-0048, R1	Increased post-accident auxiliary building room temperatures	The post-accident room temperatures in the auxiliary building rooms cooled by the service water system will be revised. This will not alter the inputs or assumptions made in the accident analysis. There are no physical changes made to the service water system or related room coolers. Assumptions previously made in evaluating the radiological consequences of an accident are not altered.
ABN 93-0-0059, R0	Revise gasket description to "asbestos (or equivalent)" for ECCS MOVs	A description of ECCS MOV gaskets in the FSAR will be changed from "spiral wound asbestos gasket" to "spiral wound asbestos (or equivalent) gasket" to indicate that the spiral wound gaskets can be composed of materials other than asbestos. The MOVs will continue to function and operate as before, with no reduction in reliability. The design function and performance of the ECCS will not be adversely affected.
ABN 93-0-0060, R0	Low probability of control room evacuation	This change is a clarification of a statement in the FSAR concerning evacuating the control room due to a postulated control room fire. This clarification does not make any changes in assumptions or data in FSAR section 9B. This clarification does not change the fire heat loading of the control room or other fire protection assets. This clarification will not affect or modify any plant equipment.
ABN 93-0-0125, R1	CCW system engineering study	Documentation is being revised to resolve discrepancies to reflect the as-built configuration of the plant. The proposed activity does not introduce any physical modifications to the CCW system. The proposed activity does not cause the CCW system to be operated outside the design limits nor is system reliability degraded. The CCW system is not an accident initiator and the functions of the system are not changed for any accident evaluation described in the FSAR.
ABN 93-0-0171, R0	Discrepancies in fire detection instrumentation	This change updates the number of fire detectors and the location of two sprinkler systems to the as-built configuration for three FSAR tables. This documentation change of the as-built configuration does not adversely affect the detection systems or their components' ability to perform safety related functions. The effectiveness of the fire protection system is not reduced.
ABN 93-0-0187, R0	Combustible amounts of cable insulation in cable trays	The addition of combustibles in the form of cable insulation will not adversely affect the fire protection system since the areas affected have fire detection and suppression available to accommodate an increase in fire rating. The addition of combustibles in the form of cable insulation will not change, degrade, or prevent actions described or assumed in the FSAR for a fire.
ABN 93-1-0012, R0	CCW from RCP thermal barrier vent valves	The addition of vent valves to two drawings to reflect the as-built condition does not affect any system, structure or component which has a safety related function. There are no direct or indirect impacts to the design basis of the component cooling water system. None of the affected equipment is relied upon to mitigate the consequences of a malfunction of equipment important to safety.
ABN 93-1-0120, R0	As-built for waste holdup tank local sample line	Documentation is being revised to accurately reflect the as-built configuration of the plant. This does not change the design or operation of the waste holdup tank.

Identifier	Description	Safety Evaluation Summary
ABN 94-0-0289, R1	Revised RCP motor curves-motor protection	Revised curves and motor data for the RCPs supplied by the vendor are being incorporated into the affected documents. The RCP starting current must be changed in the FSAR. The new RCP motor data does not adversely affect the existing overcurrent protection relay settings for the RCP motors. The RCP motors will continue to perform their safety function.
ABN 94-0-0290, R0	Assessment Observation CI-CM-02 and CI-MECH-01 developed during the Containment Isolation System Safety Assessment	This ABN updates the FSAR as a result of the Containment Isolation System Safety Assessment. These miscellaneous changes incorporate information supported by plant design and design documents, and provides consistency between FSAR sections. These FSAR revisions do not change the operation of the containment systems.
ABN 94-0-0305, R0	RHR system FSAR changes	This ABN revises the FSAR to reflect the resolutions of open items associated with the RHR system Self Initiated Safety System Assessment. These revisions to the FSAR correct discrepancies between the FSAR and design basis documents. These FSAR revisions do not adversely affect any structure, system, or component associated with the RHR system since design conditions are not affected by the changes. The revisions to the FSAR do not introduce any physical modifications to the RHR system or to the interfacing systems. The revisions do not cause the RHR system to be operated outside of the design limits. The revisions do not degrade RHR system reliability.
ABN 94-0-0311, R0	Reactor Protection System - revision to FSAR due to FSD	Documentation is being revised to accurately reflect the as-built configuration of the plant. Several discrepancies were discovered between the FSAR and design documentation during the writing of the RPS FSD. This does not change the design or operation of the reactor protection system.
ABN 94-0-0321, R0	HVAC/CO2 system discrepancies	Documentation is being revised to accurately reflect the as-built configuration of the plant. This does not change the design or operation of the HVAC and CO2 systems.
ABN 94-0-0340, R0	Control room ventilation FSD	This ABN documents editorial changes and current design (as-built configuration) on referenced drawings as a result of discrepancies found during preparation of the Functional System Description for the Control Room Ventialation System. The control room ventilation system design and operation are not changed. The non-editorial changes have been evaluated and do not change the control function of the control room ESF Habitability System. The editorial type changes do not have any safety significance.
ABN 94-0-0348, R0	Discrepancies found in CVCS FSD	Discrepancies found in the CVCS Functional System Description have been evaluated and will be corrected by revising the FSAR. This will better document the as-built configuration of the plant. No components or systems involved in the initiation of transients will be modified. These documentation changes do not change system function or design parameters beyond the qualification of any of the components which could cause it to fail.

Identifier	Description	Safety Evaluation Summary
ABN 94-0-0414, R1	Revision to the aluminum inventory described in the FSAR	The presence and quantities of hydrogen producing materials inside containment is not an accident initiator. This does not change, degrade, or prevent actions described or assumed in an accident. Assumptions previously made are not altered since the post accident environment pH is not adversely affected and equilibrium sump solution will remain above the pH required to assure that iodine is retained in the sump solution.
ABN 94-0-0452, R0	Update FSAR to accurately reflect as-built conditions concerning the RWST level alarms	Documentation is being revised to reflect the actual condition of the plant and are editorial changes. This does not change the design or operation of the Residual Heat Removal System.
ABN 94-0-0513, R0	Deletion of door 451	Door 451 has been removed. This door did not represent a fire barrier. No fire resistance design credit was taken for door 451. Removing door 451 does not change the fire analysis or habitability systems of the control room. Removing door 451 will not affect the positive pressure design of the control room or the isolation damper closure functions of the control room HVAC system. Removal of door 451 will not affect the safety of the plant.
ABN 94-0-0520, R0	Clarify operating position of CCW to RHR heat exchanger MOVs 3185A & B	This change is to maintain CCW to RHR heat exchanger MOV 3185A or B open in the off-service CCW train. This modification does not affect the design basis of the CCW system nor the interfacing RHR system during its ECCS injection and recirculation functions. It does not affect structures, systems or components used in mitigating the effects of an accident since the CCW system will perform its functions of post-accident heat removal with MOV 3185A or B open.
ABN 94-1-0459, R0	Location of test connection N1B12V011 in RCS	Documentation is being revised to accurately reflect the as-built configuration of the plant by showing a test connection in the RCS for local leak rate testing. This documentation change does not change the design or operation of the RCS.
ABN 95-1-0667, R0	As-built configuration of the argon supply to the post-accident sample panel	This FSAR figure change will show the as-built configuration of the tubing supplying argon to the post accident sample panel. No modifications are performed. This documentation change does not adversely affect the NSSS or its components' ability to perform safety related functions.
ABN 95-1-0736, R1	Diesel generator heat exchanger tube bundle replacement	The diesel generator heat exchanger tube bundle replacement has no adverse effect on any of the accidents that may have radiological consequences, nor will it change the radiation limits for the plant. There are no adverse effects on any component or structure in the diesel generator system. This implementation will have no impact upon the operation or reliability of the diesel generator system.
ABN 95-1-0753, R0	RHR (D/G Loading)	The increase in RHR motor power requirements does not affect the availability or capability of the RHR pump to perform its intended function. The increase is within the capacity of the electrical auxiliary system and diesel generators. The safety function of the RHR system has not been altered. This increase does not create any new failure modes for the RHR or electrical auxiliary systems. This change will not increase radiological consequences for any accident and will not affect any plant equipment important to safety.

Identifier	Description	Safety Evaluation Summary
ABN 95-2-0621, R0	Drawing change to reflect removal of temporary isolation plates in the cooling tower "B" bypass nozzles	The removal of the temporary isolation plates does not adversely impact the circulating water system's reliability or performance. The circulating water system was returned to it's original design by removing the temporary isolation plates, and this ABN changes documentation to reflect the current condition of the 2B tower.
ABN 96-0-0882, R0	ASME Code Class N-411 piping analysis change	The addition of ASME Code Case N-411 allows alternate seismic damping values for piping system analysis. The piping code stress allowables for the system will still be met. Pipe break locations will not be affected.
DCP 90-1-6760-0-002	Snubber reduction in the chemical and volume control system (inside containment)	Piping system re-analysis for the CVCS reduces the quantity of snubbers while maintaining stresses within code limits. This will result in significant savings in inspections and maintenance costs in addition to reducing personnel radiation exposure. This change does not affect system functions or operations.
DCP 94-1-8744-0-001, 002, & -1-003	EH fluid system pump replacement	The constant displacement pumps and unloader valves have been replaced with variable displacement constant pressure pumps to increase reliability and reduce maintenance costs of the EH fluid supply system. System pressure will be maintained after replacement. Implementation of these pumps will have no adverse impact upon the operation and reliability of the EH system.
DCP 94-1-8751-1-001	Replacement of William Powell service water valves with stainless steel valves	This modification, including deleting a check valve and adding two manual gate valves, meets applicable standards and has no adverse affect on the system, structure, or components. Stainless steel valves provide adequate strength to meet service water design conditions, and they are corrosion resistant. This change will not degrade the operational reliability or availability of the service water system. Seismic requirements will continue to be met.
DCP 95-0-8816-2-001	Control room air conditioning system - replacement of four 600 amp feeder breakers	This modification replaces four 600 amp feeder breakers with power distribution blocks. This modification will not decrease the level of protection for MCCs 1F and 1G. The proposed design complies with all the protection and control guidelines listed in the appropriate section of the Electrical Distribution System Functional System Description.
DCP 95-1-8806-0-001 & 002	Removal of steam dump warming lines	A stress analysis determined that the steam dump warming lines are not required for thermal stress protection of the steam dump valves and piping. The main steam dump warming lines perform no function in mitigating the consequences of an accident. The affected piping is non-seismic and non-safety related.
DCP 95-1-8823-0-001 & 002	Temporary A train service water return from control room A/C	This piping modification permits short-term utilization of the service water train A return line due to an inaccessible leak located in the vertical pipe chase. The loss of SW from the SW return system is inconsequential and will not adversely impact any cooling function of the SW system.
DCP 95-1-8941-0-001	CCW heat exchanger service water return line modification	This modification replaces service water isolation valves without operator extensions and removes tube bundles. This modification will have no adverse affect on any component or structure in the CCW or service water system. Seismic qualification and Code requirements will continue to be met.

Identifier	Description	Safety Evaluation Summary
DCP 95-2-8789-0-001	Snubber reduction - steam generator blowdown system	Piping system re-analysis for the steam generator blowdown system reduces the quantity of snubbers while maintaining stresses within code limits. This will result in significant savings in inspections and maintenance costs in addition to reducing personnel radiation exposure. This change does not affect system functions or operations.
DCP 95-2-8790-0-001	Snubber reduction - RHR relief valve to PRT	Piping system re-analysis for the residual heat removal (RHR) system pressure relief line reduces the quantity of snubbers while maintaining stresses within code limits. This will result in significant savings in inspections and maintenance costs in addition to reducing personnel radiation exposure. This change does not affect system functions or operation.
DCP 95-2-8807-0-001 & 002	Isolation of steam dump warming lines	A stress analysis determined that the steam dump warming lines are not required for thermal stress protections of the steam dump valves and piping. The main steam dump warming lines perform no functions in mitigating the consequences of an accident. The affected piping is non-seismic and non-safety related.
DCP 95-2-8844-0-001	Defeat of FD5 annunciator while in auto rod control	This design change maintains the main control board black board annunciator concept during the auto and manual rod control mode. This modification allows operation of Unit 2 at 100% reactor power in either auto or manual rod control mode with the C11 annunciator window FD5 in a cleared condition. These changes are non-safety and non-Class 1E. This design change also relocates the C9 condenser available BPLB light to a spare window adjacent to the C7 steam dump demand. This groups the C9 and C7 windows together for human factor consideration. There is no change to the C9 or C11 setpoints.
DCP 95-2-8847-0-001	Fuse modification for RWST low-low level rollover switches	These FSAR text changes provide clarification of the RWST low-low level actuation with respect to the energize-to-actuate design of this circuit. These design changes improve the reliability of the SSPS and will not adversely affect the operation of the SSPS. This wiring modification will not impact the required level of protection of the power supplies, since the power supplies will continue to be fused and breaker protected. The installation of the additional fuse does not create any new failure modes, but ensures the ability of the ESFAS to function as designed by mitigating the long-term consequences of a LOCA in the event of turbine building electrical faults.
DCP B-88-1-4773-1-001 & 003	Automatic sequencing for additional emergency loads	This design change will allow an air compressor on the A train DG during LOSP, LOSP/SI, and SBO events, and provide for automatic tripping of other loads. This modification will not adversely affect the design function or the performance of the compressed air system or the emergency on-site power system. The changes do not affect the reliability of the electrical distribution system. There will be no adverse impact on the capabilities of the DC battery. There will be no adverse impact on the safe shutdown capabilities of the plant during a fire.

Identifier	Description	Safety Evaluation Summary
DCP B-90-1-6633-0-001	Regenerative heat exchanger personnel barrier	This modification adds a permanent personnel wire mesh barrier around the regenerative heat exchanger, a new ladder, and a small landing inside containment. This modification does not change, degrade or prevent actions described or assumed in an accident described in the FSAR. No assumptions previously made in evaluating the radiological consequences of an accident have been altered. This activity does not adversely affect any structure, system or component used in mitigating the radiological consequences of an accident described in the FSAR. No mitigating product barriers are being adversely affected. This modification does not change the zone designations; the containment will remain zone designation V.
DCP B-90-1-6880-0-004	Reactor coolant vacuum refill	The interface connections installed for the reactor coolant vacuum refill system will have no adverse effect on the RCS system. The integrity of the systems' pressure boundary will be maintained. This modification does not degrade structure, system, or component reliability. The addition of the interface connections will not impair the safety functions of the RCS components and piping.
DCP B-90-1-6986-0-001	Unit 1 auxiliary building room cooler replacement	Venting and draining operation for the affected room coolers can be accomplished by other means. These modifications do not adversely affect the operation of the room coolers or the single failure criterion in FSAR Table 9.4-7. The affected room coolers are not discussed in the accident scenarios described in the FSAR. Deletion of the vent and drain connections does not adversely affect any structures, systems, or components used in mitigating the radiological consequences of any accident. No assumptions previously made in evaluating radiological consequences are altered.
DCP B-90-2-6987-0-001 & 002	Unit 2 auxiliary building room cooler replacement	Replacing room cooler coils with those of a more corrosion resistant material will improve the service and reliability of the coolers. This modification will not adversely affect the intended operation of the room coolers, and will not impact the safety of the plant. Assumptions previously made in evaluating radiological consequences of an accident are not altered.
DCP B-90-2-7129-0-001	CTMT emergency lighting	AC lighting fixtures powered by an uninterruptible power supply (UPS) will replace DC emergency lighting battery packs for containment emergency lighting. The UPS units will be installed in accordance with seismic category II/I requirements. This modification will not adversely affect the operation of the emergency lighting system, alter the design characteristics of any equipment important to safety, or impact the safety of the plant.
DCP B-93-1-8536-0-002	Steam generator narrow range level tap relocation (reference W SECL 93-305, R2)	This change revises documentation to depict the new locations of the steam generator narrow range level taps in the conical regions of the steam generator. The technical evaluation for the modification is addressed in SECL-93-305. The scope covered by this SECL will not adversely affect the operation of the steam generator narrow range level range transmitters, and will not impact the safety of the plant.

Identifier	Description	Safety Evaluation Summary
DCP B-93-2-8565-0-001, 002, & 003	Modification of CRDM cooling duct to delete the "doughnut"	These modifications remove the doughnut chamber surrounding the CRDM shroud and connect the cooling fans directly to two shroud outlet nozzles. The CRDM cooling system is not required to mitigate the radiological consequences of an accident. The reduction in containment heat sink does not adversely affect the containment pressure/temperature profile. No other system will be adversely affected.
DCP B-93-2-8626-1-001	S/G narrow range level - median signal selector	This modification adds the median signal selection (MSS) to the circuits for the steam generator level inputs to the feedwater control system. The installation of the MSS and associated design have no adverse affect on the function of equipment which performs, or is associated with, a radiological mitigating function. No assumptions previously made in evaluating the radiological consequences of an accident are altered. This proposed modification does not change, degrade nor prevent actions described or assumed during an accident.
DCP B-93-2-8627-0-004	Steam generator narrow range level tap relocation	This change revises documentation to depict the new locations of the steam generator narrow range level taps in the conical regions of the steam generator. The technical evaluation for the modification is addressed in SECL-93-305. The scope covered by this SECL will not adversely affect the operation of the steam generator narrow range level range transmitters, and will not impact the safety of the plant.
DCP B-94-2-8724-0-001	Relocation of main steam hanger 2MS-R84	The deletion of pipe support 2MS-R84 and the addition of SCS-2H-220 affect the stress levels in Unit 2 main steam line "C." However, the stress levels remain within code allowable limits. The postulated break locations are not changed. The design change has no adverse affect on the function or operation of the main steam piping.
DCP P-93-1-8644-0-001	Relocation of PAX phone from RCA exit to HP counting room	Plant telephone extension 2532 has been relocated to allow the HP foreman to effectively communicate during outages with technicians who obtain air sample results from various plant systems. The operation and reliability of the communication system will not be adversely affected by these changes. The communication system changes are an improvement to the plant phone system. Emergency communications are not jeopardized.
DCP P-94-1-8729, R1	Auxiliary building groundwater intrusion control in room 194	Fire Area 1-6 is revised to add the waterstop material to the list of room combustibles, but does not change the fire severity of the area. The waterstop material will be added to the isolation joint which is leaking water into room 194 of the Unit 1 auxiliary building. This change will not degrade the fire protection program nor the Appendix R requirements for safe shutdown. This modification will have no adverse impact on any plant system or structure nor will it affect the seismic design of the building.

Identifier	Description	Safety Evaluation Summary
DCP P-95-1-8808-0-001	Removal of transmitters N1C60PT505 and N1C60PT507	This modification deletes instruments installed to measure secondary plant conditions. This deletion will not adversely affect any equipment which is part of the accident analysis. No system or equipment required to perform dose mitigating functions are adversely affected. This modification does not change, degrade or prevent actions described or assumed in an accident. Assumptions previously made in evaluating the radiological consequences of an accident are not altered.
DCP P-95-1-8881-0-001	Addition of airline isolation valve for MSIV test cylinder	This change installs a 3/8" tubing isolation valve and piping between the existing airline tee and the test cylinder solenoid valve due to the leak-by of solenoid valves HV3369A and 3370A. Due to the small mass of the valve and the support available at the solenoid valve and the airline tee positions, no seismic concerns are created by this change. The operability of the MSIVs to perform their design function in the case of any automatic or manual actuation has not been compromised. The MSIVs closure time and availability will not be affected.
DCP P-95-2-8838, R1	Auxiliary building groundwater intrusion control in room 2194	Fire area 2-6 is revised to add the water stop material to the list of room combustibles, but does not change the fire severity of the area. The water stop material will be added to the isolation joint which is leaking water into room 2194 of the Unit 2 auxiliary building. This change will not degrade the fire protection program, nor the Appendix R requirements for safe shutdown. This modification will have no adverse impact on any plant system or structure nor will it affect the seismic design of the building.
DCP P-95-2-8845, R0	Addition of airline isolation valve for MSIV test cylinder	The leak-by of the solenoid valves on test cylinders for HV 3369A and 3370A required installation of an isolation valve between the airline tee and the test cylinder. Due to the small mass of the valve and the support available at the solenoid valve and the airline tee positions, no seismic concerns are created by this change. The operability of the MSIVs to perform their design function in the case of any automatic or manual actuation has not been compromised. The MSIVs closure time and availability will not be affected.
DCP S-84-1-2934-2-001 & 002	Auxiliary building DC ground detection system	This change has no adverse effects on the auxiliary building DC system or any other system, structure or component. The ground detection system meets Seismic II/I criteria to maintain the seismic integrity of the DC switchgear. This change does not adversely affect the capability of the auxiliary building DC system to perform the safety function during both normal operation and during design basis events.
DCP S-84-2-2933-2-001	Auxiliary building DC ground detection system	The design change has no adverse effects on the auxiliary building DC system or any other system, structure, or component. The ground detection system meets Seismic II/I criteria to maintain the seismic integrity of the DC switchgear. This design change does not adversely affect the capability of the auxiliary building DC system to perform its safety functions during normal operation and design basis events.

Identifier	Description	Safety Evaluation Summary
DCP S-86-2-3503-0-001	Circulating water pump canal level interlock removal	The removal of the circulating water canal level switch and pump start interlock circuitry will not degrade plant safety or affect the safe operation of the pumps. The circulating water pumps are not safety related and are not required to achieve and/or maintain safe shutdown of the plant.
DCP S-86-2-3528-0-001	Secondary chemistry lab expansion	This change expands the secondary chemistry lab and relocates storage lockers. The impact of the load additions on the electrical distribution system has been evaluated, and is acceptable. The changes proposed will have no adverse affect on any systems or structures required for safe shutdown, nor the Appendix R requirements for safe shutdown of the plant.
DCP S-91-0-7265-1-001	Utility building gasoline storage tank	The removal from service of the gasoline storage tank does not affect operation of any plant system or component. No safety related plant systems, components, or equipment will be affected by this change. The tank will be closed per the Alabama Department of Environmental Management Administrative code Rule 335-6-15 for Underground Stroage Tank Regulation.
DCP S-91-1-7578-0-001	Undervoltage relay modification for 4KV buses 1F, 1H, 1G, and 1J	The circuits modified are used to initiate LOSP signals. The modification does not alter the function of the 2 out of 3 under-voltage scheme now in use. This modification simplifies the circuit design, thereby making the systems more reliable. All of the original design specifications are met, including seismic.
DCP S-91-2-7579-0-001	UV relays for 4KV buses 2F, 2H, 2G, and 2J	The circuits modified are used to initiate LOSP signals. The modification does not alter the function of the 2 out of 3 under-voltage scheme now in use. This modification simplifies the circuit design, thereby making the systems more reliable. All of the original design specifications are met, including seismic.
DCP S-91-2-7662-0-002	RHR pump/motor coupling modification	These changes modify the RHR pumps to incorporate a removable spacer coupling between the pump and motor. This modification does not adversely affect the RHR system or CCW system design, operability or performance. This design change has no adverse effect on any of the accidents or transients that may have radiological consequences.
DCP S-91-2-7662-0-008	RHR pump/motor coupling modification	The RHR pump performance curves will be revised as a result of post-modification testing. The revised pump performance curves reflect the mininmum acceptable design curve. No modifications to the plant are required by this change in acceptance criteria. The purpose of the reduced acceptance curve is to ensure that the RHR pumps perform in a manner to bound the accident analysis for FNP.
DCP S-92-0-8229-1-001	SW chemical feed system freeze protection	This revision of the SW chemical feed system freeze protection design change deletes all descriptions and references to gaseous chlorine treatment of the SW system from the FSAR. This system has been replaced by a SW Chemical Feed System. This removal will not decrease the effectiveness of the SWS. There is no adverse affect on the function, operation, or reliability of the SWS. The SWS will continue to be treated with biocides, with no increase in biofouling of piping.

Identifier	Description	Safety Evaluation Summary
DCP S-92-2-7928-0-002 & 004	Restrictions of service water flow to the Unit 2 turbine building and the Unit 2 CCW heat exchanger	This revision provides design for the automatic stroking to a throttled position of the Unit 2 turbine building service water isolation valves upon receipt of a LOSP signal. It also provides design for limiting the Unit 2 CCW heat exchanger flow control valves' travel in the open direction to nominally 35 degrees open. These modifications will improve the ability of the service water system to provide cooling water to the emergency diesel generators without degrading its capability to support other plant systems under all conditions.
DCP S-93-0-8562-0-002	Seismic instrumentation replacement	This modification reflects the replacement of the existing analog system with a digital computer based system. The new system, utilizing modern technology, will provide more reliable on-site analysis of seismic events in a shorter length of time than the existing system. The change will enhance the operation of the seismic monitoring system and will not have an adverse effect on the safety of the plant. The new seismic monitoring system will provide the same monitoring function as the obsolete system.
DCP S-93-1-8587-0-001	Zinc addition and monitoring system (ZAMS)	The ZAMS skid components installed via this change are non-safety related. The connections to these systems are in accordance with applicable codes and standards, and are of a quality consistent with the associated systems. These connections have been designed such that the capability of these systems to perform their safety functions is not adversely impacted. The seismic aspects of this modification have been reviewed and found to be acceptable. The combustible loads added do not exceed the current fire severities.
DCP S-93-1-8637-0-001	Radwaste charcoal filter removal	Several waste gas area charcoal filter units, fans, and associated instrumentation and alarms will be removed. The radwaste area exhaust system will continue to minimize the spread of airborne contamination from the waste gas equipment rooms. The effectiveness of radiation monitor R-30 in monitoring the representative areas it serves will not be reduced. Ductwork modifications are analyzed to meet seismic II/I guidelines.
DCP S-93-1-8684-0-003	RCP undervoltage/underfrequency circuitry modification	The UV and UF logic change from "energize to trip" to "de-energize to trip," and the addition of a third DC source to one 4160 volt bus enhances reliability and does not increase the probability of an accident. The UV and UF relay logic changes will result in a fail-safe (trip) position. The hardware change will have no effect on the time required to actuate their trip signals. The change will meet the intent of IEEE 279 (1971) and will not compromise the integrity of the RPS. No other equipment important to safety is affected.
DCP S-93-2-8546-0-001	Rod control power from MG set	This modification reduces the probability of challenges to the RPS by reducing the probability of reactor trips caused by lightning strikes. This change will improve the reliability of the operation of the CRDM system. The amount of combustible material added will have negligible impact on the combustible loading in the CRDM room. This modification has been evaluated for seismic II/I requirements, and is acceptable. This modification will improve plant reliability by reducing the probability of plant trips caused by lightning, and will have no adverse affect on the rod control system, or any other plant system.

Identifier	Description	Safety Evaluation Summary
DCP S-93-2-8685-0-001 & 003	RCP undervoltage/underfrequency circuitry modifications	The change in primary reactor trip protection from UV & UF to low coolant flow does not increase the probability of occurrence of an accident. The reversal of the UV & UF logic enhances reliability and does not increase the probability of an accident. This modification improves independence and reliability of the tripping scheme. Channel separation is not affected. These changes were previously forwarded to the NRC on 6/6/95 in Revision 12A to the Updated FSAR.
DCP S-94-2-8743-0-002	600 and 480 VAC power supply for the turbine deck	This design change has no adverse affects on the AC electrical system or any other system, structure or component. 600V load center 2P and 480V lighting switchgear 2A are not required to mitigate or prevent a release of radioactive material. This change does not adversely affect the capability of any system or component to perform its safety function.
DCP S-94-2-8745-0-001 & 002	EH fluid system pump replacement	The constant displacement pumps have been replaced with variable displacement constant pressure pumps to increase reliability and reduce maintenance costs of the EH fluid supply system. System pressure will be maintained after replacement. Implementation of these pumps will have no adverse impact upon the operation and reliability of the EH system.
ES 90-1740, R0 (S)	SSPS slave relay testing	There is no functional change to the SSPS or its ability to perform its safety-related function. Testing of the AR slave relays has been identified as a contributor to the frequency of inadvertent equipment starts and reactor trips. Therefore, performing the testing of the AR slave relays during refueling and/or reducing the test frequency decreases the possibility of inadvertent equipment starts and reactor trips. The reliability or availability assumed in the accident analyses for reactor trip or ESFAS functions is not degraded.
FNP-0-AP-3, R8 (S)	Plant organization and responsibility	This is a reorganization of existing plant staff. Staffing assumptions for accidents evaluated in the FSAR are not affected. There is no change to any plant equipment.
FNP-0-AP-37.0, R11	Fire brigade organization	This revision deletes the need for a chemistry technician and adds one additional system operator responsible for emergency breathing equipment to the Fire Brigade. This change will not prevent the crew members from meeting all operational requirements required for safe operation of the plant. The minimum shift complement specified in the FSAR is maintained.
FNP-1-CCP-301, TCN 17B	Chemical control of the circulating water canal	The addition of chemicals to the circulating water system to control foam will have no detrimental effect on the circulating water system. Adding this chemical will improve the cooling tower efficiency. No plant structure, system, or component will be changed.
FNP-2-CCP-301, TCN 16B	Chemical control of the circulating water canal	The addition of chemicals to the circulating water system to control foam will have no detrimental effect on the circulating water system. Adding this chemical will improve the cooling tower efficiency. No plant structure, system, or component will be changed.

Identifier	Description	Safety Evaluation Summary
FNP-2-ETP-2098, R0	Flow test of the TDAFW pump to determine low flow noise conditions	Proper execution of the procedure will not permit the pump to be operated outside its design flowrates or send flow through piping beyond the design pressure temperatures and flowrates of the piping. The TDAFW pump will be declared inoperable and the appropriate actions required by Technical Specifications will be followed. Upon system realignment and removal of test equipment as directed in the procedure, all functional aspects of the TDAFW pump subsystem will be in the pre-test condition.
FNP-2-ETP-3032, R0	Zinc addition to the Unit 2 reactor coolant system	The zinc addition program represents a condition that equates to an enhancement to the original design basis of the plant and the probability of failure of the RCS is in no way increased. The addition of zinc to the RCS will have no significant impact on post-accident hydrogen generation and build-up in containment.
FNP-2-SOP-50.2, R19	Liquid waste processing system sump pump system operation	This modification allows draining of non-radioactive service water from the containment components to the turbine building sump, and ultimately to the river, during outages. This modification is consistent with the FSAR method for transferring the contents of a non-radioactive auxiliary building sump to the turbine building. The major issues evaluated and found acceptable due to this modification include containment integrity, removal of containment components from service, control of potentially radioactive process streams within the plant, and release of effluents to the environment.
FP 94-0302, R0 (1)	Isolating service water flow to one of the ID containment cooler's cooling coils	This temporary reduction in the cooling capacity of the containment cooler by isolating a cooling coil will have no effect on the accident containment temperatures and pressures because the accident analysis was performed assuming the presence of only a single containment cooler with a highly degraded cooling flow of 600 gpm. This temporary modification will not adversely affect the piping system or the SW flow to the containment coolers.
FP 95-0245, R0 (1)	Removal from service of one cooling coil in the 1B containment cooler	This temporary reduction in the cooling capacity of the containment cooler by isolating a cooling coil will have no effect on the accident containment temperatures and pressures because the accident analysis was performed assuming the presence of only a single containment cooler with a highly degraded cooling flow of 600 gpm. This temporary modification will not adversely affect the piping system or the SW flow to the containment coolers.
FP 95-0256, R0 (1)	Removal from service of one cooling coil in the 1A containment	This temporary reduction in the cooling capacity of the containment cooler by isolating a cooling coil will have no effect on the accident containment temperatures and pressures because the accident analysis was performed assuming the presence of only a single containment cooler with a highly degraded cooling flow of 600 gpm. This temporary modification will not adversely affect the piping system or the SW flow to the containment coolers.

Identifier	Description	Safety Evaluation Summary
MD 94-2426, R0 (S)	Filtered water storage tank repair	The piping and hoses used for this temporary modification while the FWST internal corrosion is being removed exceed the pressure and temperature requirements needed for this system. This temporary modification does not affect any system important to safety. A complete malfunction of the components installed by this temporary modification will have no impact on plant safety or any component affecting plant safety.
MD 94-2427 (S) & REA 94-0443 (S)	Diesel generator expansion joint spacer ring	The expansion joint changes will not degrade the operational reliability or availability of the SW system or the diesel generator. The addition of the ring spacer will continue to meet the applicable seismic qualification and code requirements. The expansion ring addition has no adverse effect on any radiological accident.
MD 95-2440, R0 (1)	Addition of a plug to the MSIV test cylinder solenoid vent port	Instrument line plugs have been placed in the vent port of MSIV test solenoid valves to reduce the likelihood of a loss of air through the solenoids, which could degrade the air supply to the MSIV main actuators. The design MSIV closure function is unaffected.
MD 95-2441, R0 (1)	Blocking of airline to MSIV HV 3369A, HV 3370A test cylinder solenoid valves	The airline tee that supplies air to both the MSIV main actuator and the test cylinder have been removed and replaced with a pipe nipple. This will reduce the likelihood of a loss of air through the solenoid, which could degrade the air supply to the MSIV main actuators. The design MSIV closure function is unaffected.
PCN B-85-1-3431, R66 & 67/68	RCA access control area modification	This change clarifies automatic water suppression coverage for the revised and new room configuration for the radiological access control area. This resulted in changes to the existing and new combustible loading for these rooms. The increase in the combustible load does not affect the design requirements for the fire detection systems. No special additional fire protection measures are required.
PCN B-88-2-5407, R10	Pressurizer loop seal insulation installation	The temporary temperature monitoring instrumentation installed on the loop seal piping, the associated safety valve, and the whip restraint base plate is disconnected. The disconnection will not impact the RCS reliability or performance. The instrumentation does not have any dose mitigating functions nor could it affect any equipment which has dose mitigating functions.
PCN B-90-1-6443, R1	New connection for R10, R11, and R21	This modification deletes the channel test alarm for R10, R11, and R21, and installs a test panel above the ratemeter drawer assemblies of R10, R11, and R21. No safety related equipment is affected by the change. The radiation monitors affected do not perform a post-accident dose mitigating function. These changes will not adversely affect the reliability or operability of any plant equipment.

Identifier	Description	Safety Evaluation Summary
PCN B-90-1-6443, R6	Replacement of R10, R11, and R21 paper drive and control	This change involves the deletion of radiation monitor recorders from the radiation monitoring system. The radiation monitoring system recorders are not an accident initiator and are not taken credit for in any accident evaluations described in the FSAR. The radiation monitoring system recorders do not have a dose mitigating function. Removing the recorders from the system does not affect the monitoring functions of the recorders. The monitors will still actuate an alarm on high radiation signals. This change does not adversely affect the structures, systems, or components used in mitigating the effects of an accident since the radiation monitoring system performs the same function with the recorders deleted from the system.
PCN B-91-1-7315, R0	Emergency power to the counting room and the primary chemistry lab	The removal of power for the BIT heaters and heat tracing, and addition of loads on the MCC for the sampling analysis equipment located in the Primary Chemistry Lab and the Counting Room will result in a net reduction of load on the associated diesel generators, and diesel generator reliability will not be decreased.
PCN B-91-1-7431, R1	Deletion of CCW pump trip on LO-LO surge tank level	These changes will not adversely affect the system, components, or structures associated with the CCW system since the design conditions are currently met and these changes delete a portion of the breaker control circuit which no longer provides any function. Adequate NPSH will be still be provided for the CCW pumps.
PCN B-91-2-7316, R0	Emergency power to the counting room and the primary chemistry lab	The removal of power for the BIT heaters and heat tracing and addition of loads on the MCC for the sampling analysis equipment located in the Primary Chemistry Lab and the Counting Room will result in a net reduction of load on the associated diesel generators, and diesel generator reliability will not be decreased.
PCN B-91-2-7432, R1	Deletion of CCW pump trip on LO-LO surge tank level	These changes will not adversely affect the system, components, or structures associated with the CCW system since the design conditions are currently met and these changes delete a portion of the breaker control circuit which no longer provides any function. Adequate NPSH will be still be provided for the CCW pumps.
PCN B-92-0-8228, R1	Replacement of differential pressure indicators for ITT Barton Model 227	The replacement of flow indicators does not adversely affect the system operation, since the basic design of the replacement flow indicator is the same as the original equipment. The increase in aluminum inventory in containment does not exceed the current design basis for hydrogen limits. The slight increase in hydrogen which may be generated post-LOCA does not adversely affect the redundancy of existing hydrogen control equipment, and will continue to be maintained below its flammability limit during post-LOCA.
PCN B-92-2-8267, R0-4	B2F and B2G sequencer times	The replacement relays will perform the same functions and have the same operating mode and voltage as the original relays. No accident previously evaluated can be caused by replacing the relays. The B2F and B2G sequencer panels are not accident initiators. The changes are performed in accordance with all safety related criteria to ensure that the affected equipment is maintained as Class 1E and redundancy of the affected systems is not compromised.

Identifier	Description	Safety Evaluation Summary
PCN P-93-0-8591, R0	Control room renovation	This revision increases the amount of combustibles in Fire Area 44. This change does not increase the fire severity in this room. Implementation of the change will not be in conflict with the FSAR and will not decrease the effectiveness of the fire protection program.
PCN P-93-0-8651, R0	Potable water supply to staging building	The potable and sanitary water system does not initiate any of the accidents analyzed in the FSAR. These changes will not adversely affect any equipment which is part of the accident analysis. The potable and sanitary water system is not required to mitigate the effects of any radiological accident. These changes will not adversely affect any equipment important to safety. The FSAR states that this system has no safety function and is safety class NNS.
PCN P-93-2-8477, R0	Disposable demineralizer system - addition of filter skid and pressurized vessels for microfiltration	This modification installs pressurized demineralized vessels and a microfiltration skid to the discharge of the DDS. This change to the radwaste processing system will not affect safety equipment required for accidents. The probability of release of un-monitored water to the environment is not increased. No equipment important to safety is affected.
PCN P-93-2-8676, R0	New connection for R10, R11, and R21	This modification deletes the channel test alarm for R10, R11 and R21 and installs a test panel above the ratemeter drawer assemblies of R10, R11, and R21. No safety related equipment is affected by the change. The radiation monitors affected do not perform a post-accident dose mitigating function. These changes will not adversely affect the reliability or operability of any plant equipment.
PCN S-84-1-2914, R38, 39, & 40	Service water "2" and under pipe replacement with stainless steel - addition of vent and drain valves	This change meets applicable standards and has no adverse effect on the system, structure, or components. The addition of the vent and drain valves will not degrade the operational reliability or availability of the service water system. The function of the service water system will not be adversely affected by this modification. Seismic qualification and Code requirements will continue to be met. The single failure analysis remains valid. This will have no adverse effect on any radiological accidents.
PCN S-84-2-2915, R32 & 33	Service water 2" and under piping replacement	The 2B charging pump and 2E 600V load center room cooler coil is replaced and vent and drain valves added to their piping. The addition of vent and drain lines have no effect on system structures or component. Stainless steel piping provides adequate strength to meet service water system piping while minimizing the corrosion problem. These changes will not degrade the operational reliability or availability of the service water system.
PCN S-87-1-4052, R3	Service water - Unit 1 supply to control room A/C unit	The addition of spool pieces and drain valves to permit cleanup of the piping do not adversely impact the service water system's reliability or performance. The modifications do not change the operations of the service water system and system flowrates specified in the FSAR will be maintained. The change does not change, degrade, or prevent actions described or assumed in an accident. Assumptions previously made in evaluating the radiological consequences of an accident are not altered.

Identifier	Description	Safety Evaluation Summary
PCN S-89-1-5950, R5	Replacement of station service air compressor	These changes for replacement of air compressor 1C are non-safety related. The function of the air compressors are not essential for safe shutdown of the plant. These changes will have no affect on equipment in the system important to safety.
PCN S-90-0-7111, R0	Sewage treatment plant interfaces and ties	The replacement of the sewage treatment facility has no effect on, or interface with, any plant safety related system, structure, or components. The sewage treatment facility is not relied upon to mitigate the consequences of any accident evaluated in the FSAR. There will be no modification to the safety-related or power generation systems. The new sewage plant has been designed to operate well within the environmental limits set forth in the ADEM permit.
PCN S-90-1-6576, R0	Stairwell #2 ceiling fireproofing	This change adds a fire protective coating to the structural steel supports of the ceiling in the auxiliary building. This change is an enhancement. This change has no adverse affects on any plant systems or structures and will not degrade the FNP Fire Protection Program.
PCN S-91-0-7207, R0	Relocate service water wet pit level switches QSP25LS0510 and 511	The relocation of the SW pond level switches will not impact the function or operation of the RW or SW systems. The switches are to be relocated upward to facilitate safety and ease of maintenance. The RW and SW systems will not be degraded by this modification since seismic requirements for mounting will be maintained. The required additional cable length will not impair the operation or function of the switches.
PCN S-91-1-7661, R5 & 6	RHR pump coupling modification - adding high point vents to CCW and replacing the RHR pump impellers with modified impellers	The proposed modifications do not adversely affect RHR system or CCW system design, operability, or performance. This design meets the required design, material and construction standards and has no adverse affect on the system, structure or component. The impeller modification does not impact RHR pump hydraulic performance. The safety function of the RHR and CCW systems has not been altered. This change has no adverse affect on any of the accidents or transients that may have radiological consequences.
PCN S-91-1-7661, R6-7	Verification of RHR flow rates following impeller filing	The change to the RHR impeller does not adversely affect RHR system pump operability and integrity. The safety function of the RHR system, which is related to accident mitigation, has not been altered. This does not affect the integrity of the RHR system such that its function in the control of radiological consequences is affected. The RHR impeller modification does not change, degrade or prevent the response of the RHR system to accident scenarios.
PCN S-92-0-8429, R0-1	Emergency light safety classification review	Changing the purchasing requirements from safety-related to nonsafety-related for the 8 hour emergency lighting installed for Appendix R will comply with the requirements of Appendix R and is in agreement with FSAR section 9B where the Appendix R emergency lighting installed at FNP is described. Eight hour emergency lights purchased as nonsafety-related equipment will meet all Appendix R requirements associated with fire protection equipment as presently described in the FSAR. The Appendix R evaluation evaluated the consequences of a fire on the safe shutdown of FNP, and this evaluation will not be affected by changing the purchasing requirements for the 8 hour emergency lighting.

Identifier	Description	Safety Evaluation Summary
PCN S-93-1-8489, R0	Condensate and heater drain pumps HVAC modification	This modification allows additional cooling to the condensate and heater drain tank pump motors by increasing the supply air ductwork and adding exhaust fans above the motors. The turbine building (T.B.) HVAC systems are not a precursor to any accident evaluated. The T.B. HVAC systems do not provide cooling to any equipment important to safety. The T.B. HVAC systems do not have any dose mitigating functions.
PDE 93-1-0219, R0	Replacement of seismic monitors SMA-2	The recommended replacement has been evaluated and is approved. The replacement meets the design requirements of ANSI 2.2-1988 and Regulatory Guide 1.12, R1. The replacement will continue to provide the same seismic monitoring function. The new instruments enhance the seismic monitoring because the data can be retrieved more rapidly and reliably.
PDE QC 95-0-0233, R0	Delete vendor specific names in FSAR 6.4.1.2.3F concerning HEPA filter type	Documentation is being revised to remove specific vendor names from HEPA filters as long as the filters conform to specific regulations. This change does not affect any system, structure, or component which has a safety related function. Assumptions previously made in evaluating the radiological consequences of an accident are not altered.
REA 93-0121, R0 (S)	Safety evaluation for containment P/T analysis	This analysis addresses the containment pressure/temperature response to design basis accidents with reduced service water supply to the containment coolers. The results of the analysis show that the containment peak pressure remains below 48 psig for the design basis LOCA and MSLB cases, so the original analysis described in FSAR section 6.2.1.3 remains the limiting case.
REA 94-0-0464, R0	Diesel generator building total flooding CO2 system	The diesel generator building low pressure CO2 fire suppression system is designed for automatic flooding of the diesel compartment and the fuel oil tank with a 34 percent CO2 concentration, not a 50 percent CO2 concentration as presently stated in the FSAR. Accurately stating the CO2 concentration in the FSAR does not represent a change to the fire protection program. The 34 percent concentration conforms to applicable regulatory and code requirements. The diesel building CO2 systems were designed to the 34 percent code requirement.
REA 94-0-0512, R0-1	Downgrade of forty-six safety-related sump pumps in auxiliary building to NNS	Forty-six (46) sump pumps along with their associated motors in the Unit 1 and 2 auxiliary buildings have been downgraded to commercial grade. The following potential effects were evaluated and found to be satisfactory: electrical coordination of 1E power sources based on non-1E sump pump motors, control room operating status indication for leak detection in each ESF pump room, impact of flooding in areas that contain more than one safety-related train in order to provide train separation, seismic II/I and missile-generation from nonsafety-related pumps, and impact on Appendix R SSD capabilities.
REA 94-0-0525, R3	Proposed HV switchyard equipment addition	The HV switchyard equipment addition which adds a second 230/500 KV auto transformer and a 230 KV shunt reactor significantly improves voltage control, reliability and availability of the off-site sources. These changes do not reduce the ability of the off-site or on-site power system to mitigate an accident. All equipment will be provided power as assumed in accident evaluations.

Identifier	Description	Safety Evaluation Summary
REA 94-0603, R0 (S)	Review of proposed piping changes to improve heat rate (isolating the steam dump warming lines)	Eliminating the warming steam flow for the steam dump piping will not adversely affect the design function or performance of the steam dump system. This modification has no adverse effects on any of the accidents that may have radiological consequences, nor will it change the radiation limits for the plant as currently licensed. There will be no adverse effect on any systems, structures or components in the system.
REA 94-0711, R0 (1)	Removal from service of a second cooling coil in the 1D containment cooler	This temporary reduction in the cooling capacity of the containment cooler by isolating a second cooling coil will have no effect on the accident containment temperatures and pressures because the accident analysis was performed assuming the presence of only a single containment cooler with a highly degraded cooling flow of 600 gpm. This temporary modification will not adversely affect the piping system or the SW flow to the containment coolers.
REA 95-0-0778 (S)	Turbine building IEEE separation of SSPS inputs	These changes provide clarification in the application of specific design criteria, as the design criteria relate to the existing as-built design of certain turbine building safety related field inputs to the SSPS and the 7300 system. The turbine trip inputs are not credited in any previous FSAR accident analysis for initiation of an automatic reactor trip. The changes to the FSAR do not impact the present function or design diversity of the SSPS.
REA 95-0798, R0 (S)	Spent fuel pool cooling flow reduction	The decrease in SFP cooling flow to eliminate periodic vortexing will not disrupt the flow regime within the SFP. The increase in temperature of only 1 degree will have no impact on the shielding ability of the SFP. There is no change in the acceptable radiation limits for the plant as currently licensed. There are no direct or indirect impacts to the design basis of the systems, structure or components.
REA 95-0873, R0 (S)	CCW room coolers	The temporary removal of a CCW pump room cooler from service does not impact the ability of the CCW pump to perform its safety function as described in the FSAR, nor does it change any of the limiting conditions described in the FSAR. Safety equipment in the CCW room can perform their accident mitigation functions with the cooler removed from service provided the mitigating manual actions are taken when a SW train fails.
REA 95-0971, R0 (1)	Temporary cooling of Unit 1 containment	Administrative controls will ensure proper configuration when fuel is being moved in containment or when heavy loads are being lifted over the reactor. The temporary cooling system does not affect the isolation, radiation monitoring, or filter functions of the purge supply and exhaust systems. The integrity of the safety-related isolation function of the purge system will not be challenged by the addition of the temporary cooling system. The temporary cooling equipment being installed will not jeopardize any safety-related function during a postulated seismic event. The quantity of wood used is acceptable as a transient fire loading if constructed of fire resistive wood. Rotating components will not be capable of generating a missile with a greater potential for damage than those already postulated.
SNC FS SECL (S)	Southern Nuclear organizational change - addition of position of Senior Vice President and Corporate Counsel	This is a reorganization of existing Southern Nuclear staff. Staffing assumptions for accidents evaluated in the FSAR are not affected. There is no change to any plant equipment.

Identifier	Description	Safety Evaluation Summary
SNC FS SECL, R0 (1)	FSAR change for SBLOCA modeling error for Unit 1 non-limiting break sizes	The current categorization of the SBLOCA as a Condition III event is not changed as a result of the reanalysis of this accident. Only the non-limiting break size SBLOCA peak clad temperatures (PCTs) are impacted. The limiting (bounding) 3-inch break PCT value remains unchanged (1805 degrees F). Overall emergency core cooling system performance is not adversely impacted by the SBLOCA reanalysis since system equipment will not be required to operate in a different manner outside its design basis limits to mitigate the accident.
SNC FS SECL, R0 (1)	Cycle 14 reload safety evaluation	This Cycle 14 redesign meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. Overall reactor system performance is not adversely affected by the reload redesign. The core overload is not considered an initiator for any FSAR transient. The core redesign does not have a direct role in mitigating the radiological consequences of any accident, and does not affect any of the current bases for the current analyses as described in the updated FSAR. No additional mass release or fuel failures should result from this reload.
SNC FS SECL, R0 (2)	FNP Unit 2 Cycle 11 Reload safety evaluation	The Cycle 11 reload core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. The fuel design changes, the revised large-break LOCA analysis, and the modified emergency boration system design basis have nc adverse impact on fuel rod performance or dimensional stability, nor will the core be operated in excess of pertinent design basis operating limits. Accordingly, overall reactor system performance is not adversely affected by the reload design.
SNC FS SECL, R0 (S)	Condensate storage tank missile protection	It has been determined that the increase in the probability of an accident due to the lack of missile protection of the subject connections to the CSTs is negligible. The loss of CST inventory would likely be discovered using control room level indication, level alarms, or operator rounds, and be corrected prior to loss of the CST function. SW is available as a back-up supply to the AFW pumps. Since the design of the FNP CSTs continues to reflect an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products, it is concluded that the lack of missile protection on these connections to the CSTs does not place the plant outside its design basis.
SNC FS SECL, R0 (S)	FSAR change for operator action times assigned in Chapter 15 accident analyses	This change to the FSAR to clarify the intent of assumed operator action times involves no physical alteration to the plant or changes to instrument setpoints or operating parameters. The physical operation, maintenance and testing of the plant will not be changed. There is no impact on the performance of any safety related equipment considered in the accident analysis. There is no dose increase above the 10 CFR 100 limits as analyzed. This change does not alter the design function of any system, structure or component important to safety.
SNC FS SECL, R0 (S)	FSAR update for core offloads	There are no physical changes to the SFP as a result of this revision. This is a change to the wording in the FSAR for consistency between FSAR sections, and between the FSAR and a 1992 revised SFP thermal analysis.

Identifier	Description	Safety Evaluation Summary
SNC FS SECL, R1 (2)	FNP Unit 2 Cycle 11 Reload safety evaluation	The Cycle 11 reload core design meets all applicable design criteria and ensures that all pertinent licensing basis acceptance criteria are met. The fuel design changes, the revised large-break LOCA analysis, and the modified emergency boration system design basis have no adverse impact on fuel rod performance or dimensional stability, nor will the core be operated in excess of pertinent design basis operating limits. Accordingly, overall reactor system performance is not adversely affected by the reload design.
SNC FS SECL, R2 (S)	FSAR project drawing removal	Removing project drawings from the FSAR and replacing them with a reference to their plant identification numbers in lieu of the FSAR figure number does not affect the design, function or operations of any structure, system, or component. These changes are administrative in nature and involve no physical alteration of the plant or changes to setpoints or operating parameters.
SNC SECL 95-072, R0 (2)	Zero power configuration with increased steam generator level for measurement of possible increase in sodium	The postulated occurrence of any non-LOCA transient initiated during the steam generator level test will not result in more limiting consequences than those already evaluated in the FSAR. Other events such as SGTR and LOCA are not affected by this configuration. Steam generator parameters will be continually monitored by the operators.
SNC TS LS SECL (S)	TORAP turbine valve surveillance change from monthly to quarterly frequency	A study was performed which included probability analysis and a review of overall valve reliability based on Farley's comprehensive TORAP program. This study provided the assurance that overall plant safety is maintained when extending the turbine valve surveillance from monthly to quarterly. Due to less frequent valve testing, the potential of an accident due to a plant transient or trip caused by the surveillance is reduced.
SNC TS LS SECL (S)	Southern Nuclear organization change - addition of position of Vice President and Corporate Counsel	This is a reorganization of existing Southern Nuclear staff. Staffing assumptions for accidents evaluated in the FSAR are not affected. There is no change to any plant equipment.
SNC TS LS SECL, R0 (S)	Peak clad temperature accounting errors associated with the use of NOTRUMP and SBLOCTA resulting in 10 CFR 50.46	This change address discrepancies identified with respect to use of the NOTRUMP and SBLOCTA codes in the 1985 SBLOCA model. The discrepancies impact the current PCT valves for limiting SBLOCA 3" size line breaks. Emergency Core Cooling System (ECCS) performance is not adversely impacted since system equipment will not be required to operate in a different manner outside its design basis limits to mitigate a SBLOCA. Fuel and core design is not affected as a result of the errors identified.
W SECL 91-159, R4 (S)	Reactor coolant vacuum refill	The addition of a reactor coolant vacuum refill system does not change the physical configuration of the reactor internals/CRDM nor the operating conditions of the system, thus, there is no safety impact on the reactor internals nor the CRDM system. The installation of the reactor coolant vacuum refill system (RCVRS) will not adversely affect the ultrasonic level measurement system performance, as the RCVRS does not increase the potential for signal deflection in the RCS hot leg. The safety function of the RCP seals will be preserved. The system has insignificant effects on the RCS piping. The integrity of the reactor coolant system pressure boundary will be unaffected.

Identifier	Description	Safety Evaluation Summary
W SECL 91-159, R6 (S)	Reactor coolant vacuum refill system	The addition of a reactor coolant vacuum refill system does not change the physical configuration of the reactor internals/CRDM nor the operating conditions of the system, thus, there is no safety impact on the reactor internals nor the CRDM system. The installation of the reactor coolant vacuum refill system (RCVRS) will not adversely affect the ultrasonic level measurement system performance, as the RCVRS does not increase the potential for signal effect on in the RCS hot leg. The safety function of the RCP seals will be preserved. The system has insignificant effects on RCS piping. The integrity of the reactor coolant system pressure boundary will be unaffected.
W SECL 93-167 (S)	Functional evaluation of elimination of the low feedwater flow reactor trip via implementation of the median signal selector (MSS)	The elimination of the low feedwater flow reactor trip does not affect the analysis for loss of normal feedwater, loss of all AC power to the station auxiliaries, major rupture of a main feedwater pipe and streamline break events. No changes will be required to any protection grade circuitry relied upon to bring the plant to a safe shutdown condition or to mitigate the release of radioactive material to the atmosphere.
W SECL 93-177 B/E (S)	Functional evaluation of the modification of the anticipated transients without SCRAM mitigation system actuation circuitry (AMSAC)	This modification satisfies the requirements of 10 CFR 50.62(C)(1). The following items were evaluated and found to be satisfactory: (a) the diversity between the reactor protection system and the AMSAC are not changed, (b) electrical isolation is provided between the AMSAC and the reactor protection system, (c) functional performance of the AMSAC is not adversely affected, and (d) the safety-related functions provided by the reactor protection system and the engineered safety features actuation system are not degraded.
W SECL 93-196, R3 (S)	CVCS letdown PCV-145 setpoint change	To reduce the number of chronic Chemical and Volume Control System (CVCS) letdown high flow alarms, the setpoint of valve PCV-145 will be increased from 257 psig to 450 psig. This setpoint change will not adversely affect the operability and integrity of the CVCS. The control valve setpoint change does not alter any assumptions previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident described in the FSAR.
W SECL 93-246, R0 (S)	FSAR change for reanalysis of inadvertent ECCS at power	The reanalysis of the inadvertent ECCS at power was performed to demonstrate that a small break LOCA would not occur. No dose consequences are affected by this reanalysis. The inadvertent ECCS at power event is not a limiting radiological release accident. No input assumption to any dose consequence analysis is affected. There is no increase in the consequences of a previously performed accident.

Identifier	Description	Safety Evaluation Summary
W SECL 93-305, R2 (S)	Steam generator lower narrow range liquid level tap relocation	The new level taps and associated closure hardware are designed and evaluated using ASME Code, Subsection NB requirements and criteria. In all cases, stresses in the new level tap assembly will be confirmed to be bounded by the ASME Code allowables. Therefore, the modification represents a condition equivalent to the original design basis of the plant. The location of the associated shell penetrations does not exceed the ASME Code minimum value regarding proximity of pressure vessel penetrations. Structural evaluations of the modification will confirm that steam generator integrity will be maintained during all plant conditions. All hypothetical failure of the tap, piping, or closure would be bounded by existing feed line/steam line break analysis.
W SECL 94-0115, R0 (1)	Increase in peaking factor (P-bar sub HA) from 1.48 to 1.514 for Unit 1 cycle 13	No design or regulatory limit will be exceeded. The increase in peaking factor is acceptable from the standpoint of the FSAR LOCA-related accident analyses, and will continue to satisfy the requirements of 10 CFR 50.46. There is no impact on any other analysis besides LOCA (e.g., non-LOCA, SGTR, et cetera).
W SECL 94-098, R0 (S)	Reanalysis of complete loss of flow transients using low flow reactor trip as primary trip function	The reversal of the UV & UF logic enhances reliability. This modification improves independence and reliability of the tripping scheme. The installation meets the performance mandates for seismic requirements. Channel separation is not affected. No fission barriers are affected by this change. No change in any assumptions used in radiological consequences evaluated that were previously performed is necessary. (These changes were previously forwarded to the NRC on 6/6/95 in revision 12A to the updated FSAR.)
W SECL 95-123, R0 & 1 (1)	Verification of RHR flow rates following impeller filing	The change to the RHR impeller does not adversely affect RHR system pump operability and integrity. The safety function of the RHR system, which is related to accident mitigation, has not been altered. This does not affect the integrity of the RHR system such that its function in the control of radiological consequences is affected. The RHR impeller modification does not change, degrade or prevent the response of the RHR system to accident scenarios.