Entergy

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NL-03-015
January 16, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Code of Federal Regulations 10 CFR 50.59,
Annual Report of Changes, Tests and Experiments

Dear Sir:

This letter transmits the 2001 Annual Report of changes, tests and experiments conducted at the Indian Point 3 Nuclear Power Plant in accordance with 10 CFR 50.59 for the period of January 23, 2001 to January 22, 2002. The report, required by 10 CFR 50.59(b)(2), is contained as Attachment 1 and 2 to this letter.

The summaries included in the attachment represent the changes at the facility, changes in procedures, and tests and experiments, including any that were previously implemented, used in support of changes to the Updated Final Safety Analysis Report (UFSAR).

Should you or any of your staff have questions concerning this matter, please contact Mr. Joseph DeRoy, General Manager - Engineering at (914) 736-8006.

Entergy is making no new commitments in this submittal.

Very truly yours,

Fred Dacimo
Site Vice President
Indian Point Energy Center

Attachments

IE47

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U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 2 Nuclear Power Plant

ATTACHMENT 1 TO NL-03-015

50.59 REPORT LISTING

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286
DPR-64

50.59 Evaluation Number	**Rev. No.**	**50.59 EVALUATION TITLE**
93-3-242	3	Holdup Tank Recirculation
95-3-044	4	Operation with a Steam Bubble in the Pressurizer and the Reactor Coolant System at Cold Shutdown
96-3-191	0	TCV-1107, Manual Valves & Piping Replacement
96-3-463	2	Installation of an Outage Equipment Hatch in Containment
97-3-320	0	PCV-1139 Valve and Controller Replacement
97-3-439	1	Installation of Signal Isolators in the Pressurizer Pressure and Pressurizer Level Signal Loops to Provide Boundary Separation Between the Reg Guide 1.97 Indicators and the Process Controllers
98-3-072	0	CT-85-1&2, Check Valve Internals Removal
98-3-080	1	Contractor Water Treatment System
98-3-127	1	Data Acquisition System for Performance Monitoring
98-3-142	0	Miscellaneous Installation of Backflow Preventers – PAB, FSB, RAMS, CB and DGB (Group 1)
00-3-003	3	Organizational Changes – Nuclear Generation Realignment
00-3-005	0	Install Auto-Closure Feature for Main Feedwater Motor Operated Valves BFD-5's and BFD-90's
00-3-008	1	Installation of Tornado Missile Barrier in Concrete Service Water Pipe Chase, Access Points and Internal Pipe Mechanical Seals in SWS Line 408
00-3-010	1	RCS Vacuum Refill
00-3-028	1	Core Reload for Cycle 12

50.59 Evaluation Number	Rev. No.	50.59 EVALUATION TITLE
00-3-051	0	Instrument Bus 31 and 32 SOLA Transformer Replacement
00-3-059	0	Replacement of Conitel Telemetering System and Retirement of Iniven Cabinet
00-3-064	0	Replacement of Westinghouse Type BF Relays and RE-Powering of CCR Racks
00-3-070	1	Technical Requirements Manual
00-3-076	1	Elimination of the Pressurizer Missile Shield
00-3-078	0	Removal of Gland Seal Leak-off Dropout Tank Oil Skimmer
00-3-092	1	Mansell Level Monitoring System MLMS
00-3-093	0	Replacement of 6.9 kV Undervoltage Relays 27-1A through 27-4A
00-3-095	0	Opening SWN-35-1 and SWN-35-2 Beyond Maximum Open Position Limits When RCS is Less Than 350' F
00-3-096	0	Live Transfer of 480V buses 312 and 313
01-3-004	0	Operation of Fuel Storage Building Overhead Crane Over the Spent Fuel Pit
01-3-008	0	Temporary Modification to Install Power Panel 31 & 32 Pigtails
01-3-009	0	Retirement of Iso-phase Bus Heat Exchanger Outlet Cooling Water Flow Switches
01-3-011	0	Evaluation of Bus 6A Outage Temporary Power
01-3-012	0	Use of Hydrogen Form Cation Resin in CVCS Mixed Beds
01-3-017	0	Handling of High Reactivity Fuel in New Fuel Racks
01-3-018	0	Evaluate Required Actions to Insure VC Sump Flow Monitor is Adequate to Support RCS Leakage Detection
01-3-019	0	Alignment of the Circulating Water Pumps to the Essential Service Water Header
01-3-024	0	Evaluation of Bus 6A outage Temporary Power for Fuel Storage building Exhaust Fan
01-3-027	0	Utilization of Excess Margin in OTDT and OPDT Analyses to Support Initial Cycle Startup
01-3-028	0	Basis For Core Offload Time Requirements

ATTACHMENT 2 TO NL-03-015

50.59 REPORT
SUMMARY LISTING

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286
DPR-64

50.59 Summary of Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

93-3-242	3	Holdup Tank Recirculation

This removed the interim means of filling the Reactor Coolant System from the Holdup Tanks and replaced it with a permanent flow path. Piping system components previously installed were utilized in conjunction with original plant CVCS piping and the replacement pump to provide a fill path to the RCS that complies with appropriate pressure boundary and seismic criteria.

95-3-044	4	Operation with a Steam Bubble in the Pressurizer and the Reactor Coolant System at Cold Shutdown

This evaluated the operating conditions resulting from creation of a steam bubble in the pressurizer when reactor coolant bulk temperature is below 200 °F and containment integrity is not established. This evaluation demonstrated that operation under these conditions is consistent with existing analyses already documented in the FSAR and that, for purpose of accident evaluation, it is bounded by the "Small Break LOCA During Purge" scenario in FSAR Section 14.3.5 and operation with a steam bubble in the Pressurizer, the rest of the RCS at cold shutdown, and containment integrity relaxed has been shown to be consistent with the safety analysis as described in the FSAR, providing that subcriticality time delays are observed. Any postulated accident involving loss of RCS inventory under these conditions, either via the pressurizer or anywhere else in the RCS, is bounded by an analyzed scenario.

96-3-191	0	TCV-1107, Manual Valves & Piping Replacement

This modification replaced the existing service water supply and return piping and valves for the Main Turbine Generator Seal Oil Cooler Hydrogen side. The existing cement lined carbon steel was replaced with unlined 6 – 7% molybdenum stainless steel piping. These materials are much less susceptible to stress corrosion cracking and Microbiologically Influenced Corrosion. Likewise in sediment conditions these materials have a proven resistance to crevice corrosion cracking.

96-3-463	2	Installation of an Outage Equipment Hatch in Containment

The original issue of this evaluation was to review the Indian Point 3 design basis for the purpose of identifying temperature, pressure and radiation parameters applicable to the installation of the Outage Equipment Hatch (OEH) installed in the Vapor Containment (VC) during outages. Revision 2 of this evaluation was issued to insure that the OEH met the requirements specified in the latest approval Improved Technical Specifications (ITS). An evaluation of the four relevant scenarios associated with the parameters at cold shut down showed that the OEH will perform satisfactorily for limiting design conditions of 6 psig, 240°F and 1.0 x 10^6 RAD, provided that it is not installed sooner than 24 hours after reactor shutdown subcritical).

50.59 Summary of
Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

97-3-126		Replacement of Component Cooling Water Valve AC-803

Modification 97-3-126 replaced the existing 8 inch manually operated butterfly valve on the Component Cooling Water (CCW) by an 8 inch manually operated ball valve. This valve is throttled to maintain spent fuel pool temperatures within acceptable temperature ranges. This valve is not discussed in the text of the FSAR and this change was only depicted on an FSAR figure. The design, operation and function of the Component Cooling Water and Spent Fuel Cooling Water systems were not altered by this change to the plant.

97-3-320	0	PCV-1139 Valve and Controller Replacement

Modification 97-3-320 replaced the Turbine Driven Auxiliary Feedwater Pump Pressure Control Valve, PCV-1139, and installed a new control system to ensure reliable operation of the Turbine Driven Auxiliary Feedwater Pump. This modification also installed a proportional plus integral controller to improve the operation of the Turbine Driven Auxiliary Feedwater Pump by providing better process control. This modification was necessary since the existing PCV-1139 pressure control valve was obsolete and spare parts were unavailable.

97-3-439	1	Installation of Signal Isolators in the Pressurizer Pressure and Pressurizer Level Signal Loops to Provide Boundary Separation Between the Reg Guide 1.97 Indicators and the Process Controllers

The purpose of Revision 1 to this evaluation was to evaluate the removal of Steam Generator Narrow Range Level Indicators LI-417B, LI-427B, LI-437B, and LI-447B from the list of Reg Guide 1.97 required instrumentation. The level indicators were re-classed from QA Category I to QA Category M since they are Appendix R safe shutdown components. The result was the steam generator level controllers were re-classed as QA Not-Category I and will not need isolators to separate them from the level indicators. All isolators already installed in the steam generator level control loops, for Reg Guide 1.97 protection, were removed or retired in place.

98-3-072	0	CT-85-1&2, Check Valve Internals Removal

This modification involved removing the internals of check valves CT-85-2, AFWP 331 Balance Check Valve, and CT-85-1, AFWP-33 Balance Check Valve. The check valves are in the motor driven AFW pump balancing flow paths and allow flow and pressure relief from the outboard end of the pump to the pump suction pipe. The check valves are required to open to prevent severe damage to their associated pumps. The evaluation concluded that there was no need for a check valve in the pump balance drum leak off paths because there is no need to prevent reverse flow. The modification enhanced the reliability of the 31 and 33 Auxiliary Feedwater Pumps because the pumps will no longer have a failure mode associated with failure balancing check valves. This change revised FSAR flow diagram to reflect the new status of the check valve.

50.59 Summary of
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50.59/Design Change Evaluation Number	Rev. No	TITLE
98-3-080	1	Contractor Water Treatment System

Design Change Package 98-3-080 WTS installed the pipes, valves, structural supports and electrical components necessary for the Contractor Water Treatment System (CWTS) to supply treated water to both Unit 2 & 3 and a wastewater effluent line to Unit 3. Additional pipes and valves were installed in the House Service Boiler (HSB) and the Condensate Polishing Facility (CPF) to allow treated water to fill the Regeneration Water Storage Tank, Condensate Storage Tank, or both. A transformer and disconnect switch was mounted near the CWTS to supply 480 Volts to the CWTS. A foundation was built to adequately support and anchor the CWTS, transformer and disconnect switch. The In-House Water factory (IHWF), the Ultra filtration System (UF) and the Condensate Polisher Water Factory (CPWF) will be retired in place at Unit 3 since the CWTS replaced their functions.

98-3-127	1	Data Acquisition System for Performance Monitoring

This evaluation demonstrated that no unreviewed safety questions were involved with the acceptance of the temporary Data Acquisition System as a permanent plant system. The Data Acquisition System was originally installed as a temporary plant modification but was upgrade to permanent status. The Performance Group uses the system to monitor performance on the Main Steam System. The equipment added has minimal or no interface with other plant systems and does not interface with any plant system that has a safety function. This change is classified as QA Non-Category I.

98-3-142	0	Miscellaneous Installation of Backflow Preventers – PAB, FSB, RAMS, CB and DGB (Group 1)

Design Change 98-3-142 installed backflow preventers and anti-siphon vacuum breakers in the Unit 3 City Water System for protection against cross-contamination of hazardous pollutants. City water is piped into virtually all building on the sire and is used for emergency eye and body shower stations, back-up plant/system/component cooling (i.e. the Auxiliary Feedwater System is credited as a redundant water source), fire protection system water make-up, flushing of chemical feed lines, drinking fountains, lavatory/sink stations, personnel showers and multi-purpose hose stations. The operating condition of the City Water System was not in compliance with the New York State Health Code regarding backflow prevention. This modification restored the City Water System to compliance with the NYS Health Code.

50.59 Summary of
Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

00-3-003	3	Organizational Changes – Nuclear Generation Realignment

This evaluation involved the realignment of the Nuclear Generation organizational structure, as described in the UFSAR, and the resulting changes to reporting relationships. These changes do not eliminate any functional requirements, are administrative in nature, and do not involve plant equipment or operating conditions. They do not reduce the effectiveness of the management of activities or of the oversight of plant operations.

Positions Eliminated:

- Vice President – Appraisal & Compliance
- Executive Vice President – Project Operations
- Vice President – Nuclear Business Operations

Position Transferred:

- Nuclear Safety Speak out Program from Director of Security to the Director – Regulatory Affairs and Special Projects

Position Established:

- Manager – Programs and Components Engineering

00-3-005	0	Install Auto-Closure Feature for Main Feedwater Motor Operated Valves BFD-5's and BFD-90's

In July 1999 Indian Point was informed by Westinghouse of the existence of a non-conservative assumption in the IP3 analysis of a postulated Main Steam Line Break (MSLB) with coincident failure of the associated feed regulating valves (FRV). The implementation of Design Change Package (DCP) 00-3-005 affected automatic closure of the existing motor-operated block valves for the main feed regulating valves and the low-flow (bypass) regulating valves. The DCP provided automatic isolation signals to the FRV and bypass valve MOV block valves for the FRVs and the low-flow bypass valves. The isolation signal is derived from existing redundant feedwater isolation–safety injection relays located in the CCR. The modification also replaced the existing CCR control switches for the eight MOVs. The control circuits for the 480V breaker from safeguards bus 5A to MCC-311 was modified to ensure the MCC will remain energized under both SI-non-blackout and SI-blackout conditions. The eight affected MOVs were added to the 89-10 MOV program and modified as needed to ensure their performance is consistent with 89-10 standards and also ensure that the valves will perform their auto-close function within the time limit assumed in the revised MSLB analysis and under design basis pressure differentials.

50.59 Summary of
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50.59/Design Change Evaluation Number	Rev. No	TITLE
00-3-008	1	Installation of Tornado Missile Barrier in Concrete Service Water Pipe Chase, Access Points and Internal Pipe Mechanical Seals in SWS Line 408

Revision 0 of this evaluation consisted is pre-outage and outage work for the installation of a movable steel tornado missile barrier on the roof of the concrete pipe chase over the discharge canal and cutting of a rectangular opening in the concrete roof slab and cutting and removal of pipe sections from Line 408 at two locations, internal pipe inspection, cement and weld repairs, installation and testing of pipe mechanical seals, installation of 2 permanent access points, closure of Line 408 and post modification testing. The purpose of this evaluation reflects a change only to the mechanical seal qualifications from Rev. 0 of this 50.59 evaluation.

00-3-010	1	RCS Vacuum Refill

The original evaluation was prepared for the performance of a Vacuum Refill evolution to the Reactor Coolant System (RCS) and addition of a flange connection on the power operated relief valve (PORV) vent line. The Reactor Coolant Vacuum Refill System provides a reliable, alternate method for filling the Reactor Coolant System. Boundaries which will come in contact with the vacuum conditions were evaluated and determined that no safety issues existed. This method reduces the personnel exposure, Radwaste developed, amount of chemical treatment required and reduces the critical path time of an outage.

This revision to the evaluation was to clarify how RCS level indication, during mid-loop operation using ultrasonic devices was qualified. Reference to ASME Section XI was deleted because this code does not apply to measuring fluid level in pipe and specific requirements for qualifying the process were included.

00-3-028	1	Core Reload for Cycle 12

The purpose of this evaluation was to establish controls on reload fuel selection, Cycle 12 design, core design and procurement, to ensure compliance with Entergy procedures and energy requirements. The fuel assembly design and core loading pattern have been prepared as described in Design Change Package 00-3-028. Evaluations were performed to show that the design satisfies Cycle 12 energy requirements while keeping within the bounds established by the FSAR, Improved Technical Specifications and the COLR. A Reload Safety Evaluation (RSE) was provided by Westinghouse that summarized all aspects of the core design, operation and safety analysis review. This document was based on anticipated End-of-Life (EOL) conditions for Cycle 11 and reflects the agreed upon Cycle 12 core design. Actual EOL conditions are consistent with those assumed in the preparation of the RSE, and therefore the RSE remains valid for Cycle 12.

50.59 Summary of
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50.59/Design Change Evaluation Number	Rev. No	TITLE

This evaluation incorporates the Cycle 12 Reload Safety Evaluation provided by Westinghouse, evaluates revisions to the Source Range high flux trip setpoint, incorporates the finalized core loading pattern, revises fuel failure analysis and response, based on R11 testing results, and revises and evaluates FSAR sections addressing core performance parameters, reactor vessel specimen retrieval and both VCT and WGT isolation. This evaluation supports plant operation up through startup and throughout the entirety of Cycle 12, including a postulated coastdown of 20 Effective Full-Power Days at End-of-Life, and provides adequate energy to allow full-power plant operation until the Spring of 2003.

00-3-051	0	Instrument Bus 31 and 32 SOLA Transformer Replacement

The new Instrument Bus 31 & 32 Alternate Supply Transformers replaced the existing units that were aging and beginning to show signs of performance degradation. These replacements required revisions to two FSAR figures. The replacement of both sets of transformers was performed while the respective Instrument Buses were supplied from their normal Static Inverter sources and installed while the plant was operating at normal power. Temporary transformers were installed during the replacement of the alternate supply transformers that would function in place of the alternate supply transformers if they had been required.

00-3-059	0	Replacement of Conitel Telemetering System and Retirement of Iniven Cabinet

The activity consisted of removing the existing Conitel equipment located in the 33ft. elevation of the control building. The Conitel system was replaced with a D20 Energy Control System manufactured by GE Harris. Along with the back up signal transmission the D20 provides and ensures reliable transmission of the existing signals sent to the Emergency Control Center. The modification also included the "Retirement In Place" of the non-functioning Iniven system. Iniven had been out of service for several years and no longer had any functional value. These systems are used for the sole purpose of transmitting IP3 electrical output data to various offsite locations. They do not provide nor interact with any equipment which is nuclear safety related. This evaluation was required solely for removal of information on two FSAR figures.

00-3-064	0	Replacement of Westinghouse Type BF Relays and RE-Powering of CCR Racks

Westinghouse is no longer the product manufacturer for the BF and NBF type relays. The purpose of this evaluation was to evaluate the impact of replacing the obsolete Westinghouse type BF auxiliary relays with Cutler-Hammer Type NBF relays and the additional load on the 120V AC electrical distribution system. These relays are installed in control circuits associated with the Reactor Protection System (RPS) and the Engineered Safeguards System (ESS). FSAR Figure 8.2-9 required revision as a result of this change.

50.59 Summary of
Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE
00-3-070	1	Technical Requirements Manual

The purpose of this activity was to evaluate the development and implementation of the Technical Requirements Manual (TRM), as part of the requirements for the conversion from the Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS). This activity also evaluated the incorporation of the Operational Specifications (OS) into the TRM in order for the TRM to supercede the OS. Upon the effective date of the ITS, the TRM superceded the OS.

00-3-076	1	Elimination of the Pressurizer Missile Shield

There is a reinforced concrete missile shield wall around the pressurizer above the operating floor. The original design was to protect the containment steel liner from postulated valve piece or instrument missiles connected to the pressurizer. Currently these missiles have been shown not to be credible. This evaluation was to demonstrate that the elimination of the pressurizer 3" steel plate missile shield does not adversely affect safe plant operation or shutdown. This activity included the elimination of the pressurizer 3" steel plate missile shield and adding new bearing plates at the 2" tension rods in order to facilitate the safe transfer of the post tension forces of the rods to the enclosure concrete sections and the relocation and support of the seismic monitor located on top of pressurizer shield.

00-3-078	0	Removal of Gland Seal Leak-off Dropout Tank Oil Skimmer

The Gland Seal Leak-off Dropout Tank Oil Skimmer was originally installed for the purpose of removing surface oil from the dropout tank, which collected both gland steam leak-off and turbine bearing pocket drain oil leakage. The loop seal on the dropout tank was supposed to act as a barrier to oil,.permitting the drainage of seal leak-off to the Discharge Canal, while oil floated on the surface of the tank to be picked up by the skimmer. However, due to past oil spills via the loop seal, the oil drain lines were cut and rerouted to a new separate bearing oil drain tank, leaving the dropout tank solely for collection and discharge of seal leak-off. The past several years of plant operation removed all oil residue from the tank, making the skimmer no longer necessary. The existing Gland Seal Leak-off Dropout Tank Oil Skimmer and associated skimmer discharge piping to the Main Turbine Generator Bearing Oil Drain Tank was removed and the tank nozzle plugged. Additionally, a flexible dike was installed around the nearby floor drain.

50.59 Summary of
Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

00-3-092	1	Mansell Level Monitoring System MLMS

Generic Letter 88-17 requires a licensee to "provide at least two independent, continuous RCS water level indications whenever the reactor coolant system (RCS) is in a reduced inventory condition. Water level indications should be periodically checked and recorded by an operator or automatically and continuously monitored and alarmed. IP3 satisfied the requirements by using two level instruments from the Intermediate Leg Level Indicating System, which provides an alarm feature when the RCS level is in the Hot Leg during Mid-Loop operations. However, the system is not qualified for a vacuum environment. This evaluation was prepared for the implementation of the Mansell Level Monitoring System (MLMS) that is a highly accurate, reliable, and redundant method proven dependable form of shutdown RCS level indication capable of functioning under a vacuum. This system utilizes two independent means for measuring both wide ranges and meets all the requirements for level measurement of the RCS in accordance with GL 88-17. The electrical and mechanical interfaces required to implement MLMS were designed in accordance with original design codes and requirements.

00-3-093	0	Replacement of 6.9 kV Undervoltage Relays 27-1A through 27-4A

The purpose of the Design Change Package was to replace the 6.9kV undervoltage relays on 6.9kV buses 1 through 4. The existing device, a Westinghouse relay, was being used in an application that was not recommended by the vendor. An FSAR figure was also revised.

00-3-095	0	Opening SWN-35-1 and SWN-35-2 Beyond Maximum Open Position Limits When RCS is Less Than 350' F

The purpose of this evaluation was to determine whether revising the operating procedures to allow opening of valves SWN-35-1 & -2 beyond their prescribed (single pump runout) limits when the RCS is less than 350°F results in an unreviewed safety question and to remove this restriction on the maximum open positions of these valves when the RCS temperature is below 350°F, excluding times when SW headers are cross-tied or are being swapped. The Service Water System Checkoff List and Component Cooling Water System operating procedure specify that service water valves SWN-35-1 & -2 outlets be limited to maximum open positions of 27.5 & 27 degrees, respectively. The basis for this restrictions on valve position is to prevent service water pump runout when re-establishing non-essential service water during the post-accident transfer to recirculation or loss of offsite power assuming that a second non-essential service water pump fails to start.

It was concluded that the SWN-35 valves may be throttled open beyond their single pump runout limits, provided that 2 non-essential service water pumps are operating, the maximum pump flow is not exceeded, and SWN-35-1 & -2 are restored to their 27.5 / 27 degree maximum position limits upon a single failure that renders one pump operation and prior to any manual re-start of non-essential service water pumps.

50.59 Summary of
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50.59/Design Change Evaluation Number	Rev. No	TITLE

00-3-096	0	Live Transfer of 480V buses 312 and 313

The purpose of this evaluation was to allow a live transfer of 480 Volt Buses 312 and 313. 480 Volt Buses 312, 313 are normally supplied from 6900 Volt buses 1 and 3 respectively. Tiebreaker 312T313 between Buses 312 and 313 permits one bus to serve as a backup for the other. A key interlock was provided as part of the original design between the two incoming breakers and the bus tiebreaker such that only two of the three breakers can be closed at one time. With the key interlock, a bus transfer can only be made by first de-energizing the bus to be transferred then closing the tiebreaker and re-energizing the associated loads. Likewise, the transfer back to a normal alignment also requires load shedding. To reduce the number of operations and the time necessary for transfers of buses 312 and 313, while still maintaining the original design consideration of not tying two different 6.9kV power sources through the 480V system, a change to the FSAR was done to allowing a live transfer that used the spare Kirk Interlock key. Thus with administrative limits, the original design considerations are being maintained and the potential for cross tying two different 6.9kV sources through the 480V system during a live transfer of 480V buses 312 and 313 is eliminated.

01-3-004	0	Operation of Fuel Storage Building Overhead Crane Over the Spent Fuel Pit

This evaluation revises the Technical Specification Basis and FSAR to be consistent with the Technical Specifications and establishes administrative guidelines for operation of the overhead crane to allow movement of the crane over all parts of the Spent Fuel Pool (SFP), provided that:

- The emergency ventilation system is operable,
- The SFP boron concentration is greater than 1000 ppm, and
- No loads in excess of 2000 pounds re moved over the SFP.

This is consistent with the design basis of the fuel handling and ventilation systems and the remainder of the Technical Specifications.

01-3-008	0	Temporary Modification to Install Power Panel 31 & 32 Pigtails

This was to evaluate the impact of Temporary Modification TM 94-01648-09 for Power Panel 31 & 32 cross tie installation. In addition, the spared breakers were utilized to install temporary "pigtail" cables for a temporary battery tie-in to the DC buses. This feature allows the panels to be cross-tied for battery maintenance and/or battery replacement activities when the plant is in cold shutdown.

50.59 Summary of
Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

01-3-009	0	Retirement of Iso-phase Bus Heat Exchanger Outlet Cooling Water Flow Switches

This evaluates the effect on the safe operation of the plant of the implementation of Design Change Package 01-3-009 SWS, for permanently retiring the Iso-phase Bus Cooling Water Flow switches. Retirement of these switches eliminated the alarm function for low cooling water flow, and is acceptable based on the alternate indications below:
In the absence of water flow indication, temperature of the bus can be monitored by the existing instrumentation. The existing bus temperature monitoring instrumentation provides an alarm of at predetermined bus temperature and current Alarm Response Procedures direct any remedial actions. In addition, routine Operator rounds will verify that bus temperatures are within the allowable bands. The former common alarm for low water/air flow will still be available for low airflow.

01-3-011	0	Evaluation of Bus 6A Outage Temporary Power

This evaluated Maintenance procedure SYS-017-GEN, "Installation, Control, and Removal of Support Electrical Equipment Required for Scheduled Plant 480 Volt Bus Outage"" that was developed to address reconfiguring power to equipment from various sources in order to keep that equipment in service during the Bus 6A outage. The purpose of procedure SYS-017-GEN was to provide Temporary 480 Volt electrical power to various plant equipment that is either required by Technical Specifications, the FSAR or Refueling Outage work operational needs. Loads were swapped from MCCs 37 & 36B to MCC 36A, MCC 39 and HVAC Panel 53 respectively.

01-3-012	0	Use of Hydrogen Form Cation Resin in CVCS Mixed Beds

This permits the substitution of hydrogen form cation resin for the lithium form cation resin that was used in the CVCS Mixed Bed Demineralizers. The configuration in the Mixed Bed Demineralizers was a mixture of cation resin in the lithium form and anion resin in the hydroxide form. When in service, the lithiated cation resin does not remove lithium from the Reactor Coolant System (RCS). When delithiation is needed, it is carried out by the CVCS Cation Demineralizer, which contains cation resin in the hydrogen form. Whether lithiated or not, the cation resin will perform comparable ion exchange functions and capabilities for ionic impurities and maintain the reactor coolant and corrosion product activities well within design levels described in the Technical Specifications and FSAR. With the ability to use either the hydrogen form or lithium form cation resin in the CVCS Mixed Bed Demineralizers, their operation is now more versatile.

50.59 Summary of
Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

01-3-017	0	Handling of High Reactivity Fuel in New Fuel Racks

The purpose of this evaluation was to support a change to the FSAR relative to requirements associated with subcriticality in the New Fuel Racks (NFR) in that the description includes allowance for integral burnable absorbers found in the Improved Technical Specifications (ITS). This change also evaluated the use of Cycle 12 reload fuel assemblies, which exceed a k_{eff} of 0.95 in infinite array and establishes redundant controls over receipt inspection to ensure that the k_{eff} limits in the new fuel racks are met, even when human error or equipment failure are considered. This change provides guidance for inspection and disposition of high reactivity fuel assemblies in the NFR and a description of the NFR in the FSAR was updated to the current ITS standards.

01-3-018	0	Evaluate Required Actions to Insure VC Sump Flow Monitor is Adequate to Support RCS Leakage Detection

The activity evaluated the acceptability of the Vapor Containment (VC) sump flow monitor and actions required to insure it is adequate to support Reactor Coolant System (RCS) leakage detection, under the Improved Technical Specifications (ITS). The changes required a revision to the ITS and FSAR regarding VC sump flow monitor as a viable RCS leakage detection system to insure the VC sump flow monitor could be used to support RCS leakage detection with the necessary sensitivity. Revision 1 of this evaluation allowed the new alarm to be credited for RCS leak detection and also addressed the required FSAR revision due to the previous removal of the waste disposal/boron recycle panel (WDP) category alarm in the Control Room.

01-3-019	0	Alignment of the Circulating Water Pumps to the Essential Service Water Header

The purpose of this change aligns the Circulating Water Pumps (CWP) to the Essential Service Water header to improve the reliability of the Circulating Water Pumps. This was done to permanently re-align the service water supply which resulted in increased reliability of cooling water to the CWPS and reduce the probability of loss of CWPs due to interrupted non-essential service water supply. The hydraulic impact of this change was evaluated utilizing PROTO-FLO computer program instead of the SWS Hydraulic analysis, PIPEFLOW, as described in the FSAR. All safety related SW users continue to receive their required normal and post-accident flows and therefore, the new alignment has no impact on the ability of the SWS to perform its design basis function.

50.59 Summary of
Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

01-3-024	0	Evaluation of Bus 6A outage Temporary Power for Fuel Storage building Exhaust Fan

During Refueling Outage 11, 480 Volt Bus 6A was taken out of service for planned Preventive Maintenance activities. Bus 6A provides 480 Volt power to Motor Control Centers (MCC) 36B and 37. Various pieces of equipment feed from MCCs 36B and 37 which were needed to support the outage and plant operation needs. Maintenance Procedure SYS-017-GEN, Rev. 0 was developed to address reconfiguring power to all the equipment in order to keep that equipment in service during the Bus 6A outage. This evaluation addresses the changes in SYS-017-GEN, Rev. 1 for the addition of temporary power to the Fuel Storage Building Exhaust Fan during the Bus 6A outage. The changes were temporary in nature and where during RO11 only.

01-3-027	0	Utilization of Excess Margin in OTDT and OPDT Analyses to Support Initial Cycle Startup

For plant startup in a new operating cycle, the design basis and license documentation provide no guidance for operability of the Overtemperature ΔT and Overpower ΔT instrumentation prior to measurement of loop ΔT and T-average at full power. This evaluation provided clarification on the methodology for initial cycle calibration of Overtemperature ΔT (OTDT) and Overpower ΔT (OPDT) instruments and supports recalibration of the OPDT/OTDT instrumentation at 100% power and supports implementation of a methodology for OTDT and OPDT instrument calibration which establishes a preliminary estimate of these parameters and verifies that safety margin is retained until the instruments are calibrated using measured full-power valves. It also allows for an adjustment of the runback function up to the trip setpoint on each loop. This evaluation defines the conditions that control whether a calibration at 90% RTP is required or not, based on available excess margin in the safety setpoints.

01-3-028	0	Basis For Core Offload Time Requirements

This evaluation clarified the basis behind the 145-hour restriction prior to withdrawing irradiated fuel assemblies from the reactor core, as applied to refueling procedures and describes the origin of the 145-hour time limit for handling of irradiated fuel prior to core offload. The 145-hour rule was established to ensure that the heat load in the Spent Fuel Pit (SFP) never exceeded its heat removal capacity. This is based on the SFP thermal-hydraulic analysis that modeled core offload for all postulated incore shuffles and full-core offloads. This made no formal change to any scheduled plant activities however, it did correct an inconsistency in the Improved Technical Specifications bases and added a clarification to the FSAR on inspection of the upper core plate during internals removal.