Summer

Initiating Events

Significance: G Jun 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to establish an adequate annunciator response procedure resulted in exceeding licensed thermal power
Technical Specification 6.8.1.a and Regulatory Guide 1.33, Revision 2 (in part) requires procedures be established for alarm conditions. Contrary to these requirements, the licensee failed to establish an adequate alarm response procedure for a feedwater transient alarm. Specifically, Annunciator Response Procedure ARP-001-XCP-627, Revision 11D was inadequate for the Feedwater Heater 1, 2, 4 Isolate / Level Hi-Hi alarm, in that, it failed to direct a power reduction for an isolated feedwater heater. This inadequate procedure contributed to exceeding the licensed 2900 megawatts thermal power limit on May 21, 2001. This item was entered in the licensee's corrective action program as PIP 0-C-01-0616.
Inspection Report# : 2001002*(pdf)*

Significance: G Mar 31, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to include the turbine runback circuitry within the scope of the Maintenance Rule monitoring program
The inspectors identified a non-cited violation for failure to include the turbine runback circuitry within the scope of the Maintenance Rule monitoring program as required by 10 CFR 50.65. The turbine runback circuitry is a non-safety related system that mitigates an over-power delta temperature or over-temperature delta temperature transient which would otherwise result in a reactor trip. The turbine runback circuitry was discovered to be failed and would have been unable to performed its function if called upon. The finding was of very low safety significance because the safety-related reactor protection system also mitigates an over-power delta temperature or over-temperature delta temperature transient.
Inspection Report# : 2000007*(pdf)*

Significance: G Mar 31, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to install steam generator vent line support
10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that design basis requirements be correctly translated into specifications, drawings, procedures and instructions. Contrary to those requirements the seismic design basis of the plant was not translated into specifications, drawings, procedures and instructions, in that, a support was never designed to prevent failure of the B steam generator vent valve line during a seismic event. This item is documented in the licensee's corrective action program as PIPs 0-C-00-1019 and 0-C-00-1359. This item was identified in inspection report 50-395/00-06 as an apparent violation, AV 50-395/00006-03. This licensee identified non-cited violation was characterized as an issue of very low significance.
Inspection Report# : 2000007*(pdf)*

Significance: G Mar 31, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure to include five check valves in the in-service test (IST) program
Technical Specification (TS) 4.0.5 requires, in part, surveillance requirements for inservice testing (IST) of ASME Code Class 1 components. The applicable Code required that these components with pressure retaining bolted connections be visually inspected for leakage with the insulation removed. Contrary to these requirements, on October 18, 2000, the licensee discovered that five such Code Class 1 check valves were not being visually inspected for leakage with the insulation removed. This item is documented in the licensee's corrective action program as PIP 0-C-00-1479 and is the subject of Licensee Event Report 50-395/2000010-00.
Inspection Report# : 2000007*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: root cause analysis

The licensee's root cause analysis was thorough and well-organized and was performed utilizing personnel with appropriate expertise. The root causes and contributory factors for the leak and the extent of condition were adequately determined. Actions have been established to address the root causes, contributing factors, and the extent of condition. The team concluded that the licensee's assessment provided reasonable assurance that structural integrity of the RC [reactor coolant] system, from an impending gross failure standpoint, was maintained during past operation.
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: orginal weld quality
The team's review of original weld fabrication records verified compliance with American Society of Mechanical Engineers (ASME) Code requirements for the original weld. The records were detailed and provided a good weld history. No Code compliance issues were identified. However, the records revealed extensive repairs to the ID [inside diameter] of the "A" hot leg nozzle-to-pipe weld, which were determined by the root cause analysis to be a contributor to the crack by producing high residual tensile stresses at the ID of the weld. The radiographic (RT) film did not reveal any fabrication flaws that could have contributed to the through-wall leak.
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: Code inspection of new welds
The inspection team concluded that a comprehensive and effective inspection, that met or exceeded the requirements of the ASME [American Society of Mechanical Engineers] Code, was conducted on the replacement nozzle-to-pipe dissimilar metal weld (DMW) and the stainless steel pipe-to-pipe weld.
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: boric acid corrosion inspections
The boric acid corrosion inspections performed in the last two refueling outages (RO) (April 1999 and October 1997) were adequately performed, and the pre-entry radiological survey for RO-12 identified the large accumulation of boric acid crystals on the reactor building floor that led to the discovery of "A" hot leg RC [reactor coolant] leak. Since some details had been deleted from the program, the team concluded that the boric acid corrosion inspection program should be enhanced to improve guidelines for early detection of reactor coolant leakage and to expand the scope of inspection to include the welds that are susceptible to PWSCC [primary water stress corrosion cracking]. By letter dated December 29, 2000, the licensee stated that their boric acid inspection procedures will be enhanced to provide additional detail for the inspection and evaluation of RC system leakage, and that specific components and locations to be inspected will be listed and guidance provided on methodologies for evaluation.
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: enforcement discretion
In addition to the unidentified leakage limit, Technical Specifications (TS) do not allow any pressure boundary leakage and require shutdown within six hours. This leakage was pressure boundary leakage and existed for several months prior to its discovery and therefore constitutes a violation of the Technical Specifications. However, based on the team's conclusion that the violation was not avoidable by reasonable licensee quality assurance measures and management controls, the NRC is refraining from issuing enforcement action in accordance with section VII.B.6 of the NRC Enforcement Policy. [EA-01-071]
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: metallurgical evaluation
The licencee's metallurgical evaluation of the cracked hot leg loop "A" nozzle weld, including the size measurements (length and depth) of the reported UT [ultrasonic test] and ET [eddy current test] indications by destructive examinations, was thorough. The licensee adequately characterized the failure mode to be primary water stress corrosion cracking (PWSCC).
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: new weld examinations and repairs
All welding and NDE [nondestructive examination] activities for the new welds met Code requirements. The gas tungsten arc welding (GTAW) process with Alloy 52 material resulted in rejectable weld defects in the new nozzle-to-pipe weld. A number of repair attempts were required before

successful repair using the shielded metal arc welding (SMAW) process with Alloy 152 welding material. The team agreed with the licensee's evaluation that the new weld, with different, more resistant material and less ID [inside diameter] tensile stress, should be much more resistant to PWSCC [primary water stress corrosion cracking] than the old weld. However, based on the fact that PWSCC is not totally understood, the team concluded that further evaluations and inspections will be needed before it can be concluded that the new weld is totally immune to PWSCC.
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: preservice and inservice weld inspections
The nozzle-to-pipe inspections conducted in 1980 for the preservice inspection (PSI) and in the 1987 and 1993 inservice inspections (ISI) met the applicable ASME [American Society of Mechanical Engineers] Code requirements. These inspections used the state-of-the-art NDE [nondestructive examination] technology that was available at that time. No flaws were detected that were unacceptable to the 1977 Edition of the ASME Code including the Summer of 1978 Addenda.
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: reactor coolant leak practices
The licensee's leak detection practices generally would not have been expected to identify the small leak on the "A" hot leg leak during plant operation. Although 0.3 gallons per minute (gpm) of unidentified reactor coolant leakage was present during the operating cycle, as determined by a periodic water inventory balance, this leak rate was not considered unusual and was well below the Technical Specification limit of 1 gpm. The licensee plans a number of enhancements to their RC [reactor coolant] system leakage detection practices.
Inspection Report# : 2000008*(pdf)*

Significance: N/A Feb 15, 2001
Identified By: NRC
Item Type: FIN Finding
"A" hot leg weld crack: refueling outage hot and cold leg weld inspections
Overall, the team concluded that the examinations performed on the nozzle-to-pipe welds during the current outage were of high quality. The examination methods were successfully demonstrated on a mockup with qualified personnel and state-of-the-art NDE [nondestructive examination] equipment and procedures. The use of redundant and complementary NDE techniques (visual, ultrasonic and eddy current) and the successful demonstrations of these techniques provided confidence that a sensitive inspection was conducted. The examinations exceeded the minimum requirements of the ASME [American Society of Mechanical Engineers] Section XI Code and NRC Regulatory Guide 1.150.
Inspection Report# : 2000008*(pdf)*

Significance: G Dec 23, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE SURVEILLLANCE TEST AND SYSTEM OPERATING PROCEDURES TO CONTROL PRESSURIZER TEMPERATURE LIMITS
The inspectors identified a non-cited violation for failure to establish adequate procedures, as required by Technical Specificiation (TS) 6.8.1, to ensure that the pressurizer temperature heatup and cooldown limits were maintained within the requirements of TS 3.4.9.2. As a result during the shutdown for refueling outage 12, the licensee failed to recognize that the TS pressurizer temperature heatup and cooldown limits were exceeded for short period of times, i.e., less than the allowed TS action statement time. The finding was of very low safety significance because a licensee's engineering evaluation, which included fracture toughness considerations, determined that the pressurizer remained acceptable for continued operation.
Inspection Report# : 2000006*(pdf)*

Significance: G Sep 29, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to perform evaluation required by 10 CFR 50.59, improper screening
The inspectors identified a non-cited violation that involved the licensee's failure to perform a 10 CFR 50.59 evaluation for a procedure change that provided an alternate method to supply reactor makeup water to the reactor coolant system. The issue was determined to be of very low safety significance (Green) because although the procedure change was approved for use from August 26 until September 24, the licensee never used the new procedure section and the licensee subsequently completed a 50.59 evaluation which determined that a license amendment was not required.
Inspection Report# : 2001003*(pdf)*

Significance: G Jun 15, 2000
Identified By: NRC
Item Type: FIN Finding
CERTAIN GRID CONDITIONS CAN INCREASE THE LOSS OF OFFSITE POWER INITIATING FREQUENCY
The licensee's Transient Stability Study of the Offsite Power System identified that under certain grid conditions (the transmission system lightly loaded, the Fairfield Pumped Storage Plant operating in the pumping mode at ½ or more of its rated capacity, and a fault on the 230 kilovolt (KV) offsite power supply bus) a loss of offsite power (LOSP) could occur. The licensee's probabilistic risk assessment (PRA) screening analysis of the grid conditions described above showed that there would be a slight increase in the LOSP initiation frequency resulting in a change in the core damage frequency (CDF) of less than 1.0 x 10-6. A Region II senior reactor analyst reviewed the PRA screening analysis and concluded that, based on the change in the LOSP initiation frequency and the change in CDF, this issue was of very low safety significance.
Inspection Report# : 2000003*(pdf)*

Mitigating Systems

Significance: G Jun 30, 2001
Identified By: NRC
Item Type: FIN Finding
Reactor building cooling unit test frequency
The inspectors identified that the test interval for measuring reactor building cooling unit performance was relatively long when compared to that for heat exchangers included in a Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," program. The finding was of very low safety significance because the reactor building cooling units were normally cooled by industrial cooling water which was chemically treated to reduce fouling.
Inspection Report# : 2001002*(pdf)*

Significance: G Jun 30, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
Failed to follow procedure to enter the B component cooling water heat exchanger in Action Level II and place limitations on service water temperature when testing indicated degradation
The inspectors identified a non-cited violation for failure to follow procedures to restrict service water temperatures when testing indicated that the component cooling water heat exchanger performance was degraded. The finding was of very low safety significance because service water temperatures never exceeded the more restrictive temperature limits.
Inspection Report# : 2001002*(pdf)*

Significance: SL-III Jun 23, 2001
Identified By: NRC
Item Type: VIO Violation
Failure to perform a safety evaluation required by 10 CFR 50.59
The inspectors identified an apparent violation for not performing a detailed safety evaluation, as required by 10 CFR 50.59, for a change to the facility as described in the Final Safety Analysis Report which involved a unreviewed safety question. The licensee changed the facility by removing the mullion (center divider) in the steam propagation barrier (SPB) door at the entrance to the 1DB 7.2 kV AC switchgear room. Disabling the 1DB SPB resulted in the potential that a single high energy line break could render emergency AC power to both trains of safety-related equipment inoperable. The finding was of low to moderate risk significance based upon the initiating event frequency of a high energy line break accident and the cumulative time the 1DB SPB was disabled during 1998 (Section 4OA3). [The apparent violation was dispositioned as a Severity Level III violation (Supplement I) by letter entitled "Notice of Violation (Virgil C. Summer Nuclear Station - NRC Special Inspection Report No. 50-395/01-08)," dated August 31, 2001. Following is the Notice of Violation text from enclosure 1 to this letter: 10 CFR 50.59(a)(1) states, in part, that the holder of a license authorizing operation of a utilization facility may make changes in the facility as described in the safety analysis report without prior Commission approval, unless the proposed change involves an unreviewed safety question. 10 CFR 50.59(a)(2) states, in part, a proposed change shall be deemed to involve an unreviewed safety question if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased. 10 CFR 50.59(b)(1) requires, in part, that the licensee shall maintain records of changes in the facility to the extent that these changes constitute changes in the facility as described in the safety analysis report. These records must include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question. Final Safety Analysis Report Section 3.11.1.1 defines harsh and mild environments and references drawing SS-021-018. The drawing depicts the 7.2 Kilovolt 1DA and 1DB emergency switchgear rooms as mild environments and the hallway adjacent to the 1DB room as a harsh environment. Fire Protection Procedure FPP-025, "Fire Containment," Revision 1C, allowed steam propagation barriers to be disabled, one at a time, for a maximum of 12 hours. Contrary to the above, on March 25, 1997, the licensee failed to perform an adequate written safety evaluation which provided the bases for the determination that a change in the facility did not involve an

unreviewed safety question. Specifically, a written safety evaluation for revision 1C to procedure FPP-025 failed to adequately evaluate that the licensee's disabling of intermediate building door DRIB/315 would change the 1DA and 1DB switchgear rooms from a mild environment to a harsh environment. This change in the facility increased the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report, in that, a single high energy line break could potentially result in the loss of both trains of the 7.2 Kilovolt emergency power to safety-related equipment. Consequently, the change to the facility involved an unreviewed safety question and was made without prior NRC approval. The disabling of the door existed on seven occasions during 1998 for a total of approximately 30 hours. The August 31, 2001, letter stated that disabling door DRIB/315 involved an unreviewed safety question and was done without prior NRC approval. Disabling the door had no actual safety consequences; however, the violation was of concern to the NRC because of the potential for impacting our ability to perform certain regulatory functions. Based on NRC review of your corrective actions, as documented in the letter, no civil penalty was proposed. The Severity Level III violation was closed in inspection report number 05000395/2001003, dated October 18, 2001. The August 31, 2001, letter stated that "information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and to prevent recurrence, and the date when full compliance was achieved is already adequately addressed on the docket."]
Inspection Report# : 2001003*(pdf)*
Inspection Report# : 2001008*(pdf)*

Significance: N/A May 24, 2001
Identified By: NRC
Item Type: FIN Finding
Supplemental inspection results for a White Finding and White Performance Indicator
This supplemental inspection was performed to assess South Carolina Electric & Gas Company's activities associated with identification, root cause analysis, and corrective actions for the inoperability of the turbine driven emergency feedwater pump (TDEFWP) due to a closed discharge isolation valve. The White Finding was previously characterized in NRC Inspection Report 50-395/00-05 and in the NRC's Final Significance Determination for a White Finding and Notice of Violation (dated December 28, 2000). Using Inspection Procedure (IP) 95001, "Inspection for One or Two White Inputs In a Strategic Performance Area," the inspector concluded that the licensee's problem identification and root cause analysis was acceptable. The licensee determined the root cause was due to human error, a failure to open the valve coupled with inadequate independent verification. Additionally, the licensee identified four causal factors associated with this event. The completed and proposed corrective actions, including actions to prevent recurrence, adequately addressed the results of the root cause evaluation. Additionally, IP 95001 was used to assess the licensee's evaluation and corrective actions associated with a White Performance Indicator (PI) for safety system unavailability, heat removal system (Auxiliary Feedwater). The major contributor for the PI crossing from Green to White (unavailability threshold is greater than 2%) was due to the time the TDEFWP was inoperable due to the White Finding and the extended time to complete a refueling outage. The licensee reported the White PI to the NRC during the routine first quarter 2001 PI submittal. The corrective actions identified to correct the mis-positioning of the TDEFWP discharge isolation valve were considered sufficient to address the White PI.
Inspection Report# : 2001007*(pdf)*

Significance: G Dec 23, 2000
Identified By: Licensee
Item Type: NCV NonCited Violation
INADEQUATE EMERGENCY OPERATING PROCEDURE FOR TRANSFER TO COLD-LEG RECIRCULATION
Technical Specification 6.8.1.a, requires that written procedures shall be established, implemented and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Loss of coolant accidents are an activity covered in Appendix A, under Section 6, "Procedures for Combating Emergencies and Other Significant Events." This requires appropriate procedures to respond to and combat emergencies involving loss of coolant accidents and the associated response involving transfer to cold leg recirculation. The licensee failed to establish, implement and maintain an adequate Emergency Operating Procedure EOP-2.2, "Transfer to Cold-Leg Recirculation," Revisions 0 thru 11, in that, they did not provide the necessary instructions to operators for timely actions. This issue is captured in the licensee's corrective action program as PIPs 0-C-99-1026 and 0-C-00-1101.
Inspection Report# : 2000006*(pdf)*

Significance: W Sep 21, 2000
Identified By: Licensee
Item Type: VIO Violation
Failure to follow procedures results in the [turbine-driven emergency feedwater] pump being inoperable for approximately 48 days during power operation due to its manual discharge valve being closed
The licensee's failure to properly position and independently verify the turbine driven emergency feedwater (TDEFW) pump discharge isolation valve in accordance with procedures required by Technical Specification (TS) 6.8.1 resulted in the failure to comply with TS 3.7.1.2 for TDEFW pump operability. The failure to adhere to these regulatory requirements was cited as one violation in a December 28, 2000, letter to the licensee. The two apparent violations, AV 50-395/000005-01 and 50-395/000005-02 are considered closed. In the December 28, 2000, letter the inspection finding was characterized as White (i.e., an issue with low to moderate increased importance to safety). The NRC determined that the Human Error Probability methodology, using the Technique for Human Error Rate Prediction approach, appropriately estimated the increase in risk associated with the accident sequences containing the TDEFW recovery term. The change in core damage frequency was approximately 4x10-6/year. The violation, characterized as White, was reviewed and closed in NRC Supplemental Inspection Report No. 50-395/01-07, dated July 10, 2001. The supplememtal report Summary of Findings state: "Using Inspection Procedure (IP) 95001, "Inspection for One or Two White Inputs In a Strategic Performance Area," the inspector concluded that the licensee's problem identification and root cause analysis was acceptable. The licensee

determined the root cause was due to human error, a failure to open the valve coupled with inadequate independent verification. Additionally, the licensee identified four causal factors associated with this event. The completed and proposed corrective actions, including actions to prevent recurrence, adequately addressed the results of the root cause evaluation."
Inspection Report# : 2000007*(pdf)*
Inspection Report# : 2001007*(pdf)*

Significance: G Dec 29, 2001
Identified By: NRC
Item Type: FIN Finding
Procedures Did Not Demonstrate Ability Of Back Up Air Supply To Open And Maintain Open Alternate Cooling Water Valves To The Emergency Diesel Generators
A finding was identified for procedures not demonstrating the ability of the backup air supply to maintain service water valves open to provide cooling to the emergency diesel generators during certain events. This feature is utilized in emergency operating procedures involving loss of AC power and in mitigating Appendix R fire scenarios. This finding was determined to be of very low safety significance because the potential for a loss of normal instrument air was reduced due to an installed diesel driven air compressor which backs up the normal electrically driven instrument air compressors.
Inspection Report# : 2001004*(pdf)*

Significance: G Dec 29, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure To Follow Procedure For Chemistry Sampling Of Reactor Coolant System
Technical Specifications 6.8.1.a and Regulatory Guide 1.33, Appendix A, Section 10 requires procedures be implemented covering the control of sampling of radioactive liquids. On December 12, 2001, the licensee identified that a sample valve was not closed, as required by chemistry procedure CP-903, after reactor coolant system (RCS) sampling was completed. Approximately 32 gallons was drained from the RCS during the 2 hours and 10 minutes the valve was open. Automatic RCS makeup was in service during this time. This issue has been documented in the licensee's corrective action program under Problem Identification Program report 0-C-01-2324.
Inspection Report# : 2001004*(pdf)*

Significance: G Dec 29, 2001
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure To Follow Procedure During Surveillance Test Results In Inadvertent Start Of B Motor Driven Emergency Feedwater Pump
Technical Specifications 6.8.1.c requires procedures be implemented covering surveillance and test activities of safety-related equipment. On November 20, 2001, the licensee failed to properly implement surveillance test procedure STP-120.004, in that, the B Motor Driven Emergency Feedwater Pump was started when the procedure required the pump's control switch to be placed in pull-to-lock. This issue has been documented in the licensee's corrective action program asProblem Identification Program report 0-C-01-2127.
Inspection Report# : 2001004*(pdf)*

Significance: TBD Nov 28, 2001
Identified By: NRC
Item Type: URI Unresolved item
Decision of When to Enter Fire Emergency Procedure FEP-4.0 and Evacuate the Main Control Room Due to a Fire
A finding was identified, in that, the lack of operator training combined with licensee management's expectations regarding when to enter fire emergency procedure (FEP)-4.0, Control Room Evacuation Due to Fire, could result in the operators taking actions during a fire in the main control room (MCR) that would not be consistent with the licensee's safe shutdown analysis, fire hazards analysis, or procedure FEP-4.0. The operator training program neither addressed nor had job performance measures (JPM)/simulator scenarios for MCR operator actions and evacuation due to a fire in accordance with procedure FEP-4.0. This finding was determined to have a credible impact on safety because it affected the ability of the operators to perform actions (within the times required by the licensee's safe shutdown analysis and fire hazards analysis) necessary to achieve and maintain post-fire safe shutdown conditions. Licensee management's philosophy and expectations contributed to the operators' performance and slow response in deciding whether to enter procedure FEP-4.0 and evacuate the MCR during two simulator scenarios observed by the team.
Inspection Report# : 2001009*(pdf)*

Significance: G Nov 28, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
Emergency Lighting Installation Deficiencies for Performing Alternative Shutdown Actions

A non-cited violation of Virgil C. Summer Operating License Condition 2.C. (18), Fire Protection System, was identified for failure to install battery pack emergency lighting units, in accordance with the approved V.C. Summer Fire Protection Program, in 13 areas (access and egress routes included) where manual operator actions were required to support post-fire safe shutdown. This finding had a potential to impact the licensee's ability to shut down the plant in the event of a loss of power to normal lighting during a fire. The finding was of very low safety significance because it did not affect fire detection, fire suppression, or fire barriers.
Inspection Report# : 2001009*(pdf)*

Significance: G Jun 24, 2000
Identified By: NRC
Item Type: FIN Finding
PLANT WAS PLACED IN ELEVATED RISK LEVEL
The licensee removed the B trains of Component Cooling Water (CCW) and charging from service during preventative maintenance on a CCW valve without recognizing that this placed the plant in an elevated risk level as defined in the licensee's safety function matrix. As a result, provisions of Operations Administrative Procedure (OAP)-102.1, "Conduct of Operations Scheduling Unit," Revision 3, concerning evaluating the configuration and obtaining the General Manager's approval were not met. Since there was no actual loss of safety function with A train CCW and charging available and operable, this issue was determined to be of very low safety significance. No violation occurred since the licensee complied within the time constraints of the applicable technical specification limiting conditions for operation and the procedure will not be required by regulations until the revised sections of the Maintenance Rule (10 CFR 50.65) become effective in November 2000.
Inspection Report# : 2000004*(pdf)*

Significance: N/A Jun 15, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
INADEQUATE 10 CFR 50.59 SAFETY EVALUATION
The licensee's 10 CFR 50.59 safety evaluation performed to incorporate the results of the Transient Stability Study of the Offsite Power System into the Updated Final Safety Analysis Report (UFSAR) Section 8.2.2.2 did not provide an adequate technical basis to support the determination that an unreviewed safety question did not exist. Specifically, the 10 CFR 50.59 did not address the increase in the probability of occurrence of a malfunction of the loss of voltage relay for Case Study Six of the Transient Stability Study; and it did not provide an adequate technical basis to support the conclusion in the UFSAR that the requirements of 10 CFR 50, Appendix A, General Design Criterion 17 would still be met for the grid conditions evaluated for Case Study Six. This is a violation of 10 CFR 50.59 and is in the licensee's corrective action program as PIP 0-C-00-0569. Based on the changes in the loss of offsite power frequency and core damage frequency, this issue was of very low safety significance.
Inspection Report# : 2000003*(pdf)*

Significance: G Jun 15, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
FAILURE TO TRANSLATE INTO APPROPRIATE PROCEDURES DESIGN BASIS LIMITING VALUES
The licensee failed to translate into appropriate procedures and/or acceptance criteria (1) the 105 F design basis limiting value for the diesel generator (DG) intercooler water heat exchanger outlet temperature and (2) the requirement to derate the DGs if the intercooler water heat exchanger outlet temperature exceeded the 105 F value. The issue was of very low safety significance because the licensee's operability evaluation concluded that, with a derating factor applied, the DGs were still operable. This issue was determined to be a violation of 10 CFR 50, Appendix B, Criterion III, Design Control and is in the licensee's corrective action program as PIPs 0-C-00-0603 and 0-C-00-0629.
Inspection Report# : 2000003*(pdf)*

Significance: G Jun 15, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
DESIGN INFORMATION WAS NOT CORRECTLY TRANSLATED INTO A CALCULATION
Design information (e.g., DG design heat load, instrument uncertainty) was not correctly translated into calculation DC07610-002, Revision 1. The calculation incorrectly concluded that one of the two 50% capacity ventilation fans per DG could maintain the associated DG rooms below the Technical Specification limit of 120 F during DG operation for an outside ambient temperature of up to 95 F. Based on the design heat load, one DG ventilation fan could maintain the associated DG rooms below 120 F for an outside ambient temperature of up to only 79.4 F. The issue was of very low safety significance because there were no instances identified where one DG ventilation fan was taken out of service and the associated DG was still considered to be operable. This issue was determined to be a violation of 10 CFR 50, Appendix B, Criterion III, Design Control and is in the licensee's corrective action program as PIP 0-C-00-0570.
Inspection Report# : 2000003*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Significance: G Dec 23, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
IMPROPER IMPLEMENTATION OF TECHNICAL SPECIFICATION RADIATION PROTECTION PROGRAM REQUIREMENTS
The inspectors identified a non-cited violation for failure to adhere to a radiation protection procedure as required by TS 6.11, "Radiation Protection Program." On October 26, 2000, electronic dosimeters (ED) were used as radiological controls for scaffold construction activities in a residual heat removal heat exchanger room. Contrary to a health physics procedure, ED dose rate alarm setpoints were established at 300 millirem per hour (mrem/hr) rather than greater than the 400 mrem/hr general work area dose rates adjacent to the residual heat removal heat exchangers. As a result workers were not properly responding to dose rate alarms. The finding was of very low safety significance because an overexposure did not result, a substantial potential for such an exposure did not exist and the licensee's ability to assess worker's dose was not compromised.
Inspection Report# : 2000006*(pdf)*

Public Radiation Safety

Significance: G Jun 24, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
ATMOSPHERIC EFFLUENT MONITOR CALIBRATIONS DID NOT MEET 10 CFR PART 20.1501(b) REQUIREMENTS
As of April 10, 2000, selected atmospheric effluent process monitor calibrations did not meet 10 CFR Part 20.1501(b) requirements. Specifically, secondary calibration sources in-use since the early 1990's for the particulate and gaseous channel detectors were not traceable to the original primary detector calibrations. Evaluations of the effect of geometry and fabrication differences between the original, vendor-supplied sources and the current secondary calibration sources identified a potential 25 percent bias in expected detector response. Based on the identified bias, the current detector responses for monitoring radioactive material concentrations and for establishing set-point values were determined to be conservative. An additional example of a previously issued non-cited violation (50-395/99006-03) was identified. This additional example is in the licensee's corrective action program as CERs 99-1170 and 99-1172. Since effluent releases did not result in doses exceeding Appendix I to10 CFR Part 50 design criteria nor 10 CFR 20.1301 concentration limits, this finding was considered to be of very low safety significance.
Inspection Report# : 2000004*(pdf)*

Physical Protection

Miscellaneous

Significance: G Mar 23, 2001
Identified By: NRC
Item Type: FIN Finding
Problem Identification and Resolution Annual Inspection
No findings of significance were identified. The implementation of the corrective action program (CAP) was acceptable with concerns noted.

Management oversight was evident in all aspects of the program, and trending was extensive with an appropriate focus on human performance. The licensee was generally effective at identifying problems and placing them into the CAP. Self-assessment by the CAP department was very limited and not well documented. The licensee generally evaluated individual problems and established acceptable schedules for implementing corrective actions appropriately. Corrective actions were generally implemented in a timely manner. The apparent cause determinations appeared to accurately identify why the equipment problems occurred. The inspectors determined that the licensee properly classified discrepant conditions. The inspectors found that the scope and depth of corrective actions assigned by the licensee were generally appropriate for the severity and risk significance of the problems identified. Two issues identified during this inspection concerned the effectiveness and timeliness of corrective actions associated with previous NRC-identified Non-Cited Violations (NCVs). In addition, the inspection team observed that the Primary Identification Program (PIP) process was not effectively using the Repetitive Condition portion of the PIP database. The identification of repeat problems was dependent on the memories of individuals involved in the PIP process, rather than being retrievable from the PIP database. Interviews of plant personnel indicated that they felt free to input safety issues and conditions adverse to quality into the CAP. A safety conscious work environment was evident at Summer.
Inspection Report# : 2001006*(pdf)*

Significance: G Dec 23, 2000
Identified By: Licensee
Item Type: NCV NonCited Violation
FUEL HANDING BUILDING NEGATIVE PRESSURE EXCEEDED TECHNICAL SPECIFICATION REQUIREMENT
Technical Specfication (TS) 4.9.11.d.3 surveillance requirement states that the spent fuel ventilation shall maintain the spent fuel area at a negative pressure greater than or equal to 1/8 inches water gauge relative to the outside atmosphere during irradiated fuel movement and during crane operation with loads over the pool. Contrary to that requirement on October 16, 2000, the licensee discovered that fuel movement had occurred without the proper fuel handling building negative pressure. The failure to meet this TS requirement is documented in the licensee's corrective action program as PIP 0-C-00-1455.
Inspection Report# : 2000006*(pdf)*

Significance: G Dec 29, 2001
Identified By: NRC
Item Type: FIN Finding
Temporary Backup Diesel Air Driven Compressor Installed In The Instrument Air System Without Design Control Documents
A finding was identified for having installed a backup diesel driven air compressor in the instrument air system without design control documents, i.e., without temporary or permanent plant modification documentation. References to this compressor implied it was "temporary" even though it had been installed in the plant since approximately 1982. The finding was determined to be of very low safety significance because no significant adverse impacts had been experienced during the time period it has been installed and its performance was being monitored under the maintenance rule program.
Inspection Report# : 2001004*(pdf)*

Significance: N/A Dec 29, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
Failure To Process Procedure Revisions In Accordance With Administrative Procedure For Procedure Review And Approval
Technical Specifications 6.8.1.a and Regulatory Guide 1.33, Appendix A, Section 1.e, requires procedures be implemented covering the procedure review and approval process. On October 2, 2001, the licensee identified that procedure changes were approved without all the applicable provisions of SAP-139, "Procedure Development, Review, Approval and Control," being met. This issue has been documented in the licensee's corrective action program under Problem Identification Program reports 0-C-01-1700, 1722 and 1925. (No Color)
Inspection Report# : 2001004*(pdf)*

Last modified : March 27, 2002