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SUBJECT: Annual 10CFR50.59 rept for 1990,including summaries of changes to procedures &/or plant mods which change plant as described in FSAR.

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FEB 27 1991

Letter Number: HO-910015 (O) 10CFR50.59

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SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
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REPORT IN ACCORDANCE WITH 10CFR50.59

Gentlemen:

In accordance with 10CFR50.59, the following report is submitted for the year of 1990. This report contains brief summaries of changes to procedures and/or plant modifications, which change the plant as it is described in the FSAR. There were no tests or experiments conducted during this interval, which are not described in the FSAR and require reporting in this report.

Very truly yours,

R. B. Richey
Vice President
Harris Nuclear Project

MGW:gcm

Enclosure

cc: Mr. S. D. Ebneter (NRC - RII)
Mr. J. E. Tedrow (NRC - SHNPP)

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MEM/HO-9100150/1/OS1

Change to Plant as Described in the FSAR

Title: PCR-000214, Addition of HVAC to Waste Processing Building (WPB) Control Room Locker Area and Toilet Facility.

Functional Summary:

This plant modification installed a bathroom and locker facility adjacent to the WPB Control Room. FSAR Section 9.4.3 discusses the WPB Control Room HVAC System which will now supply air to the added bath/locker facility.

Safety Summary:

The WPB Control Room HVAC system is a non-safety system and is not required for safe shutdown of the plant. The system provides cooling to the WPB Control Room and pressurization for radiation protection of operators. This change does not affect the systems ability to perform its intended function.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Section 9.4.3 and Figure 9.4.3-4

Change to Plant as Described in the FSAR

Title: PCR-001546, Use of Nukon Fiberglass Insulation Inside Containment

Functional Summary:

This plant modification approved the replacement of the metal reflective insulation inside containment with Nukon Fiberglass Insulation (Owens-Corning Fiberglass Corp.) on a one-for-one basis, as deemed necessary to replace defective insulation.

Safety Summary:

In a letter dated December 8, 1978, the NRC staff accepted the use of Nukon insulation inside nuclear containments. "Based on quantitative and qualitative tests performed by or for Owens-Corning Fiberglass, the staff concluded that the Owens-Corning Fiberglass Corporation's nuclear containment insulation system (Nu'k'on) is capable of retarding heat loss from piping and equipment in containment areas, and that the overall integrity of the blankets will not be adversely affected by the conditions found during the lifetime of the plant. It was concluded that during a loss-of-coolant accident, the Owens-Corning Fiberglass insulation system is not expected to interfere with the operation of the emergency recirculation system." The staff's acceptance was based on Topical Report OCF-1 (dated December, 1978), developed by Owens-Corning Fiberglass Corporation which adequately addressed the six concerns stated below.

1) Release of airborne particles leading to a radiation health hazard in service;

2) Stress corrosion cracking of the austenitic stainless steel surfaces that comes in contact with the insulation;

3) Deterioration of the thermal properties during normal plant operation, complicating operation and control of the plant;

4) Presenting a fire hazard in the containment area that could interfere with safe operation of the plant;

5) Interference with the emergency spray system in the event of a LOCA;

6) Blocking of pressure relief ports in the event of an accident;

Additional plant specific analysis was conducted to confirm that Nukon insulation does not pose any additional threat to containment sump screen blockage at SHNPP.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Section 6.2.2

Change to Plant as Described in the FSAR

Title: PCR-001887, Reconfiguration of the Fuel Handling Building (FHB) Fuel Pool A to Allow Increased Storage of BWR and PWR Fuel Assemblies.

Functional Summary:

This plant modification allows the reracking of the FHB Fuel Pool A to allow for storage of BWR and PWR fuel assemblies. Pool A is being transformed from a new fuel storage area for PWR fuel only into a composite PWR and BWR irradiated fuel storage area. This change also allows the contents of the Spent Fuel Pool B to be transferred into the A pool in order to perform liner repairs on the B pool.

Safety Summary:

Fuel Pool A is designed for the storage of new and spent PWR fuel. Since the 11X11 BWR rack modules are interchangeable with 7X7 PWR rack modules it is acceptable to store spent BWR fuel in the A pool. The A pool meets all of the design and performance requirements as the B pool and can be used as a spent fuel pool with no changes to its cooling or purification capability.

Rearrangement of the racks in the A pool has no effect on the maximum stored criticality since the individual racks are designed to maintain a subcritical array regardless of rack arrangement or boron concentration. The maximum intended heat load from the proposed rack arrangement would be less than the heat load if all PWR fuel were stored in the A pool since the PWR fuel constitutes a greater heat load when compared with the BWR fuel.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Section 9.1

Change to Plant as Described in the FSAR

Title: PCR-001993, Primary Sample Panel '1A' Ventilation Modifications.

Functional Summary:

The Primary Sampling System (PSS) is designed to collect fluid and gaseous samples contained in the Reactor Coolant System and Safety Injection System. It is also designed to collect fluid samples from the Boron Thermal Regeneration System, Chemical and Volume Control System, Steam Generator Blowdown System, residual heat removal heat exchangers, and a gas sample from the volume control tank and main steam. The PSS provides samples in two sampling rooms in the Reactor Auxiliary Building, and brings them to a common location in the sampling rooms via 1A and 1B Primary Sample panels for analysis by the plant operating staff.

This plant modification increased the air flow rate exhausted from the Primary Sample Panel 1A in order to assure proper capture velocity of contaminants generated during sampling. The air flow rate exhausted was increased from 300 CFM to 1000 CFM with all hood doors open. This increase in air flow was accomplished by reducing air flow quantity to be exhausted from the service water discharge pipe tunnel without adversely affecting the pipe tunnel design space temperature. Ductwork internal to the primary sample panel 1A from the sample vessel enclosure to the common exhaust header was increased from three inch to seven inch diameter.

Safety Summary:

This modification affects two Q-Class E systems. 1) The Reactor Auxiliary Building (RAB) Ventilation System (exhaust side) and, 2) the Primary Sampling System (ventilation portion only). Neither of the two systems are initiating or mitigating systems.

The subject air flow changes do not effect structural integrity of the seismic designed portion of ductwork of the RAB Normal Ventilation System as the operating pressure remains unchanged.

This change does not increase levels of airborne contamination (radioactivity) released via the RAB vent stack nor does it result in releases via unmonitored release points. The revision does result in more positive capture of airborne contaminants generated during sampling which increases safety of operations personnel. The reduction in the exhaust flowrate from the service water discharge pipe tunnel does not result in space temperature increases above current design.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Figure 9.4.3-1

Change to Plant as Described in the FSAR

Title: PCR-002252, Snubber Reductions

Functional Summary:

This plant modification abandons in place mechanical snubbers No. CS-H-963 and CS-H-968 from the Chemical and Volume Control System in the Reactor Auxiliary Building Elevation 261'-0. The snubbers were abandoned as part of a snubber reduction effort to reduce maintenance cost and man-rem exposure during testing and repair work.

Safety Summary:

This modification does not impact the function or operation of the Chemical and Volume Control System. The stress analysis and structural acceptability show that the abandonment of the snubbers is acceptable.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Therefore, no unreviewed safety questions exists.

FSAR Reference:

Table 3.9.3-16

Change to Plant as Described in the FSAR

Title: PCR-002290, Waste Gas Analyzers Replacement

Functional Summary:

Each Hydrogen Recombiner Package of the Gaseous Waste Processing System (GWPS) includes a Gas Analyzer System. This plant modification replaces the existing Bendix Hydrogen and Oxygen Analyzer with functionally equivalent Teledyne analyzers for both non-safety A and B trains of the Catalytic Hydrogen Recombiner Package.

The analyzer changeout facilitated installation of new flow control panels, rework of the existing tubing from the hydrogen recombiner skids to the gas analyzer racks and an increase in the compressor suction line size to minimize system backpressure. Also, the new Teledyne analyzers cannot be damaged during no-flow conditions (system shut-down), therefore, deleting the need for low flow de-energization of the analyzers.

Safety Summary:

This plant modification incorporates changes to improve the GWPS performance and reliability. The electrical portion of this system is fed from a non-ESF supply which is not required to operate during an emergency shutdown. The instrumentation and control functions of the GWPS as described in FSAR Section 11.3.2.2.2 have not been affected by the Teledyne Analyzer changeout.

Control signal failure probability has been decreased due to the deletion of the low-flow analyzer shutdown, which is not required for the Teledyne Analyzers. Leaving the Teledyne Analyzers energized during system shutdown or no-flow conditions will have no adverse impact on the analyzers themselves nor system operation or availability.

The increase in the compressor suction line size has no adverse affect on the system or plant operation. The line is designed and constructed in accordance with the original codes. The two lines being moved are designed and constructed to the original codes. The relocation will not adversely affect any system. The redesigned lines are not routed over any safety related equipment. No new crossties between equipment or systems are being made by this plant modification.

The GWPS performs no function related to the safe shutdown of the plant. This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Section 11.3.2 and 9.5

Change to Plant As Described in the FSAR

Title: PCR-002297, Corrosion Product Sampler Installation

Functional Summary:

This plant modification to the Secondary Sampling System installs iron and copper corrosion monitors on the following sample lines:

a) Condensate pump discharge
b) High pressure heater drains
c) Feedwater to the steam generators

These monitors are installed to determine the volume of corrosion products entering the steam generators. The monitors also provide an indication of the source of the majority of corrosion products.

Safety Summary:

The corrosion products monitors are designed to meet or exceed the secondary sample system design pressures and temperatures with one exception. The feedwater to the steam generators sample point design pressure is 2000 psig while the corrosion monitor design pressure is 1500 psig. A pressure relief valve set at 1500 psig maximum is installed per this modification on the corrosion monitor inlet sample line to prevent monitor overpressurization in the event the sample line pressure exceeds 1500 psig. The secondary sample system is Quality Classification E which is assumed to fail during a seismic event and release any sample liquid to the equipment drains for processing. Failure of the corrosion monitors will not increase the consequences of a radioactive release over what is presently analyzed. The corrosion monitors are designed using stainless steel materials to minimize corrosion and erosion. The table on which the corrosion monitors are mounted is designed to capture any leakage from the monitors and direct it to the equipment drain.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Section 9.3.2.2.2

Change to Plant as Described in the FSAR

Title: PCR-003701, Removal of Flow Switches FS-7001A and FS-7001B from the Environmental Qualification Program.

Functional Summary:

This plant modification was issued to remove Flow Switches FS-7001A and FS-7001B from the Environmental Qualification (EQ) Program. FS-7001A and FS-7001B function as redundant, safety-grade instrumentation monitoring the component cooling water (CCW) flow via a common header (3CC6-201SN-1) from all three reactor coolant pumps. If CCW flow decreases to the low flow setpoint, alarms are initiated at the monitor light boxes (MLB-4A and MLB-4B) and ERFIS messages for Points FRC-7001C and FRC-7001D are generated. Low CCW flow to the RCP oil coolers requires Operator action to restore the CCW flow or to take other steps, as deemed appropriate by Operations. The flow switches are located in the Reactor Auxiliary Building above Elevation 236' on Line 3CC6-201SN-1. This location makes it difficult for plant technicians to perform EQ mandated calibrations.

Safety Summary:

As detailed above, FS-7001A and FS-7001B monitor CCW flow from the reactor coolant pumps bearing oil coolers. The flow switches are safety grade and during normal operation, the location of FS-7001A and FS-7001B is mild for both temperature and radiation. Therefore, the only time the flow switches are subjected to a harsh environment is post design basis accident (DBA). If there is a spurious loss of CCW, not associated with a DBA, FS-7001A and FS-7001B will function as designed, and there will be no change in their operating environment.

In the event of a DBA the containment will be placed in Phase "A" isolation either by the initiation of Safety Injection or manually from the Main Control Board. Phase "A" isolation does not isolate the CCW from the reactor coolant pumps (RCPs). This allows these pumps to operate during safety injection and will also allow the operators sufficient time to shutdown the RCP's prior to Phase "B" isolation which does secure the component cooling water. The RCP's however, are not safety related devices and as such cannot be considered to be available following a DBA. In addition, there is no postulated accident in which the RCPs are utilized for accident mitigation. Therefore, the basis for removing FS-7001A and FS-7001B from the EQ Program is, they will perform their safety function prior to an accident in a mild environment. They are not required to operate after a Design Basis accident when a harsh environment is created.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR REFERENCE: Table 3.11.0-7

Change to Plant as Described in the FSAR

Title: PCR-003754, Chemical and Volume Control System (CVCS) Heat Exchanger Performance Thermometer Deletion.

Functional Summary:

This plant modification removed temperature indicators associated with the CVCS, due to ALARA concerns.

Temperature indicators TI-01-CS-7241, TI-01-CS-7243, and TI-01-CS-7244 are located in high radiation areas and present ALARA implications during ILRT and routine calibration activities. These Dresser dial thermometers provide local indication of CVCS Regenerative and Excess Letdown Heat Exchanger performance; however, this temperature monitoring is not utilized for performance trending due to local exposure rates. Temperature elements TE-01-CS-0123, TE-01-CS-0139, and TE-01-CS-0140 provide Main Control Room monitoring of Heat Exchanger outlet temperature making the subject indicators expendable.

The subject indicators have been removed and threaded pipe plugs installed in the existing thermowells with necessary sealant.

Safety Summary:

The affected thermometers are installed at the CVCS Regenerative and Excess Letdown Heat Exchangers, monitoring inlet/outlet temperature. These devices, providing performance related local indications, render no control functions and are not necessary for Plant Process Display or Post Accident Monitoring.

This modification, deleting unnecessary instrumentation, does not degrade plant safety considering its effect on accident-initiating systems, accident-mitigating systems, or key safety considerations.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Figure 9.3.4-1

Change to Plant as Described in the FSAR

Title: PCR 004192, Replacement of Turbine Trip Condenser Vacuum Switches.

Functional Summary:

This plant modification was necessary due to the poor performance characteristics of the turbine trip condenser vacuum switches PS-4131AV thru DV and the fact that Westinghouse revised the vacuum trip setpoint. New turbine vacuum pressure switches and circuitry have been installed.

Specifically, this modification replaces the existing United Electric pressure switches with Static '0' Ring (SOR) pressure switches which have proven more reliable, adds a separate pressure switch for pre-trip alarm and revised control circuitry such that a setpoint change occurs at an increase of approximately 60% power from the low setpoint (5.0 InHg) to the high setpoint (7.5 InHg).

Safety Summary:

Presently during the summer months reactor power reduction is required to prevent a turbine trip on low condenser vacuum. This has been attributed in part to the reduction in cooling tower efficiency and miscellaneous water chemistry make up. Per Westinghouse recommendation, at approximately 60% power the condenser vacuum turbine trip setpoint should be increased from 5.0 InHg to 7.5 InHg to allow the unit to operate at full power production.

This has been achieved by installing a relay which will energize at approx. 60% power and switch the dual Hi-Lo SOR pressure switch from the low setpoint to the high setpoint. New alarm pressure switch PS-4131EV will also be switched via the same relay to provide operators pre-trip warning at 4.0 InHg below 60% power and 6.5 InHg above 60% power. The additional alarm relay will replace one of the DEH fluid low pressure trip switches.

This enhancement does not effect the original design intent of the low condenser vacuum trip circuit as discussed in FSAR 10.2.5 but allows for increased unit reliability and availability.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Figures 10.2.2-08, 10.2.2-10

Change to Plant as Described in the FSAR

Title: PCR-004506, Deactivation of the Chlorine Detection System.

Functional Summary:

This plant modification deactivated the chlorine detection system. Technical Specification 3.3.3.7, Amendment 8, stated that two independent chlorine detection trains be operable whenever liquid chlorine is present at the onsite chlorine storage area in quantities greater than 20 lbs. SHNPP no longer stores chlorine in large quantities on site, therefore, this system is no longer required. Amendment ten (10) of the SHNPP Technical Specifications deleted the Chlorine detection System from Technical Specifications.

Safety Summary:

The Chlorine Detection System consisted of two independent chlorine detector trains with each train consisting of a detector at each Control Room Area Ventilation System intake (both normal and emergency) and a detector at the chlorine storage area.

The storage area detectors alarm and isolate the control room in the event of a release of chlorine at the storage area. CP&L does not store large quantities (i.e., quantities greater than 20 pounds) of liquid chlorine onsite at Harris. Therefore, the accidental onsite release of such a small quantity of chlorine would not affect the plant operators. As such, deletion of the storage area chlorine detectors will not increase the consequences of an accidental onsite release of chlorine. The deactivation of the Chlorine Detection System will also avoid inadvertent control room isolations.

The chlorine detectors located at the Control Room Area Ventilation System intakes are intended to provide protection in the event of an accidental offsite release of chlorine. A probabilistic risk assessment (PRA) was performed to determine the probability of an accidental chlorine release in the vicinity of Harris. The results of the analysis showed that offsite chlorine release accidents have such a low probability that they are not considered to be credible events.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Sections 1.8, 2.2, 3.1, 6.4, 7.2, 7.3, 9.4, 9.5

Change to Plant as Described in the FSAR

Title: PCR-004695, Removal of the Main Steam Power Operated Relief Valve (PORV) Actuators from the Environmental Qualification (EQ) Program.

Functional Summary:

The Main Steam PORVs, located in the Main Steam Tunnel, utilize an Electro-Hydrologic actuator manufactured by Paul Monroe-Enertech. This plant modification removed these actuators from the Harris Plants EQ Program. The requirement for these actuators to be environmentally qualified was re-evaluated by this modification. It was determined that the subject actuators are not required to mitigate any FSAR Chapter 15 analyzed event that could result in an elevated temperature or pressure in the area of the steam tunnel.

Safety Summary:

Since these actuators are not required to mitigate a Chapter 15 event that could cause an elevated temperature or pressure in the steam tunnel, it is unnecessary to maintain their EQ status.

This change does not increase the probability or consequences of analyzed accidents, no introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Table 3.11.0-2

Change to Plant as Described in the FSAR

Title: PCR-004930, Instrument Air Check Valve.

Functional Summary:

This plant modification installs a 3" 150# carbon steel welded swing check valve in the Instrument Air (IA) System. The check valve is installed in the main supply header into the Radiation Control Areas (RCA) (i.e., Reactor Auxiliary Building, Containment Building) to prevent any reverse flow in the system flow path.

During routine plant shutdown surveillance testing, improper work activities by test personnel caused radiological contamination of various piping sections of the IA system. This situation created immediate concerns due to IA system flow path contamination inside the RCA, along with potential problems regarding leakage paths outside the RCA via the IA supply header which crosses the RCA boundary.

The immediate problems were resolved by flushing the affected piping until samples showed contamination levels within acceptable levels inside the RCA boundary and by restricting use of the Emergency Breathing Air System. The potential problem regarding IA system leakage paths outside the RCA boundary is being addressed by the installation of the subject check valve.

Safety Summary:

The IA system, a part of the Compressed Air System originates in the Turbine Building and provides "instrument quality" air via piping to the Turbine Building, Reactor Auxiliary Building, and Containment Building.

The Compressed Air System (CAS) is not required for the initiation of any engineered safety feature systems, safe shutdown system, or any other safety-related system. Therefore, the CAS is considered non-nuclear safety except for the containment penetrations and the valve accumulators.

A major piping or component failure in the Turbine Building could cause rapid depressurization of the IA flow path, thereby creating a reverse flow to outside the RAB and RCA. This event could cause release of radioactive contamination to the environment. The installation of the subject check valve provides means to prevent this type of occurrence.

During and after an accident there are no air-operated valves that require cycling to bring the plant to safe shutdown. The accident analyses does not assume the instrument air system to be operable and does not take credit for the system.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Figure 9.3.1-3

Change to Plant as Described in the FSAR

Title: PCR-004984, Emergency Service Water (ESW) System Minimum Flows/Flow Balancing

Functional Summary:

Engineering Evaluation PCR-004984 established revised minimum service water flow requirements for various ESW loads and evaluated flow balances for acceptability based on the new minimum flows. As previously designed, no margin existed in the configuration of the ESW system to allow for degradation due to the highly conservative flow requirements. The new flow values were established for the worst-case condition with the ESW pumps aligned to the Main Reservoir with a level of 205.7 feet mean sea level with a water temperature of 95°F (Tech. Spec. 3/4.7.5).

A summary of required FSAR Table 9.2.1-1 changes is as follows:

Component	Old Value	New Value
Component Cooling Water Heat Exchanger	12,000 gpm	9,000 gpm
Standby Diesel Generator Coolers	1,250 gpm	900 gpm
Containment Fan Coolers	3,000 gpm	2,850 gpm
Reactor Auxiliary Bldg. HVAC Chillers	2,500 gpm	2,420 gpm

Safety Summary:

The ESW system is an accident mitigating system. The reduced flow rates still provide adequate margin for accident mitigation. This change does not increase the probability of consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. No unreviewed safety question exists.

FSAR Reference:

Table 9.2.1-1

Change to Plant as Described in the FSAR

Title: PCR-005021, Engineering Evaluation of Emergency Core Cooling System (ECCS) Flow Inconsistencies.

Functional Summary:

In December of 1989, Westinghouse informed Carolina Power & Light Company of inconsistencies found between the Harris plant ECCS flow rates assumed in the input to the Westinghouse supplied Loss-Of-Coolant Accident (LOCA) Analyses used to demonstrate compliance with the requirements of 10CFR50.46 and the flows allowed by the Harris Technical Specifications.

Specifically, the issue of concern was that the Harris Technical Specifications allowed for 31 gpm of reactor coolant pump (RCP) seal injection flow at normal charging pump conditions (at 2250 psia) versus an assumed 24 gpm (at 2250 psia) used in the generation of ECCS flow data for the Harris Safety Analyses. Since these flow values correspond to specific seal injection line resistance, it was seen that the seal injection line resistance, associated with the Technical Specifications, was actually less than that assumed in the calculations. Therefore, additional charging/SI flow would be pumped through the seal injection line potentially resulting in a reduction of flow injected into the core.

Since the potentially lower ECCS flow to the reactor core would ultimately impact the FSAR, Chapter 15, Lost-of-Coolant Accident Analyses provided by Westinghouse; an evaluation of the discrepancy was performed by Westinghouse.

Safety Summary:

The evaluation showed that the increase in peak cladding temperature (PCT), due to the reduction in safety injection flow to the core was within the bounds of the PCT limits as defined in the Harris Technical Specifications.

Based on the foregoing evaluation, it is determined that, pursuant to the criteria specified in 10CFR 50.59, the existence of the discrepancy between the seal injection line resistance allowed by the Technical Specifications and that assumed in the safety analysis does not involve an unreviewed safety question.

Because of the effects of steam/water condensation in the RCS loops, the computer model has been shown to be sensitive to reductions in safety injection flow. As a result FSAR Table 6.2.1-36 (Double Ended Pump Suction Guillotine Min. SI Reflood Mass and Energy Releases) and Table 6.2.1-41 (Double Ended Pump Suction Guillotine Min. SI Post-Reflood Mass and Energy Releases) are being revised to reflect new release data.

FSAR Reference:

Tables 6.2.1-36 and 6.2.1-41

Change to Plant as Described in the FSAR

Title: PCR-005157, Secondary Protection to ARP-19B Electrical Containment Penetration Circuits.

Functional Summary:

Electrical Containment Penetration circuits, safety related and non-safety related, are to be protected against overcurrent to prevent penetration conductor damage. In all cases, the penetration circuit protection consist of a primary and back-up (secondary) disconnecting device which can each limit the maximum I^2t at the penetration to a value less than that required for thermal damage to the penetration conductor.

During a comparison, between Maintenance Surveillance Test MST-E0007 "120/208 VAC Molded Case Circuit Breaker Test" and electrical Calculation 30-PKR "Electrical Penetration Protection (Reg. guide 1.63), it was determined that the existing 30 amp breaker did not provide sufficient protection against instantaneous short circuit current to prevent possible damage to the penetration.

This modification adds two dual element time delay fuses, Bussman type FRN-R20 to the penetration circuits. Existing fuse holders in ARP-19B (SB) were used and only wiring the internal jumpers was required to implement this modification. New jumper cables will be added and some will be replaced as a result of this modification.

The new fuses will provide sufficient conductor protection for the electrical penetrations.

Safety Summary:

The type of fuse selected is a Buss type FRN-R which is already used for protection on other electrical penetrations. The existing spare fuse holders in the ARP-19B are compatible with the FRN-R Fuse. The circuit is not degraded by the addition of the new fuses. It provides proper coordination between primary and secondary protection of the electrical penetrations in the event of circuit fault currents.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR References:

Section 8.3.1 and 16.3

Change to Plant as Described in the FSAR

Title: PCR-005264, New Fuel Dry Storage.

Functional Summary:

This plant modification installs four 6 x 10 fuel racks in the new fuel inspection pit to allow storage of new fuel during and after receipt inspection. New fuel is already being stored in the new fuel inspection pit in its shipping containers during the receipt inspection process, on a temporary basis. This change will not subject the fuel to any new hazards; it will just be staying in the fuel inspection pit longer.

Safety Function:

Installation of the four 6 x 10 fuel racks in the fuel inspection pit was thoroughly evaluated. The pit slab was structurally evaluated and the seismic loading of the racks were considered.

The current SHNPP Facility Operating License requires that fresh fuel be stored with a minimum of 12 inches edge-to-edge between adjacent assemblies (when fuel is outside its shipping container or approved storage rack location). The use of four racks with fuel in every-other-cell of every-other row (15 assemblies per rack) satisfies this licensing requirement.

The license also requires that new fuel assemblies be stored in such a manner that water will drain freely from the assemblies in the event of flooding/draining of the fuel storage area. The rack cells each drain freely thru holes in the rack baseplate at each cell location. NUREG-0612 makes it clear that heavy load drop accidents are a concern only for "spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal" (Section 1.1). Consequently, new fuel damage due to load drop accidents is not a safety concern.

Placing four 6 x 10 racks in the new fuel inspection pit for new fuel dry storage (loading every-other-cell in every-other-row) with 15 new fuel assemblies per rack does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exits.

FSAR Reference:

Section 9.1.1 and 9.1.4

Change to Plant as Described in the FSAR

Title: PCR-005331, Installation of Corrosion Coupon Racks.

Functional Summary:

This plant modification installs corrosion coupon racks on the Reactor Auxiliary Building Component Cooling Water (CCW) System, Waste Processing Building CCW System, and the Boron Thermal Regeneration (BTRS) Chilled Water System. The racks are required to monitor the effectiveness of the corrosion inhibitor added to these systems.

Safety Summary:

The corrosion coupon racks are installed on non nuclear safety portions of the CCW System. Failure of the coupon racks will not deter the operation of the safety portion of the CCW System. Corrosion rack components are designed to meet or exceed the Reactor Auxiliary Building and Waste Processing Building CCW design pressures and temperatures.

The coupon rack is installed on the BTRS Chilled Water System which is a non-nuclear safety system assumed to fail during an accident situation. This rack will not affect the operation of any safety system. All corrosion coupon rack components are designed to meet or exceed the chiller system design pressure and temperatures.

This change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Figures 9.2.2-1, 9.2.10-1, 9.3.4-4

Change to Procedure as Described in the FSAR

Title: HPP-509, Operation of the UFV-260, Underwater Filter/Vacuum Unit.

Functional Summary:

The purpose of Plant Procedure HPP-509 is to address the control, setup, operation, and maintenance of the Tri-Nuclear Underwater Filter Vacuum Unit, Model UFV-260. The vacuum will primarily be used in the spent fuel pools at the Harris Plant to support receipt of spent fuel from CP&L's Robinson and Brunswick Nuclear Plants.

Safety Summary:

Operation of this vacuum is in no way related to any FSAR Chapter 15 initiating event. Although the unit will be operating in the spent fuel pools, the only plant systems that the UVF-260 will interface with are the fuel handling crane systems. The total vacuum package weighs approximately 600 lbs. (wet), which is well under the maximum capacity of the 3 cranes located in the fuel handling building.

The Chapter 15 fuel handling accident, as well as technical specifications assume that there is 23 feet of water above the fuel at all times. FSAR Section 9.1.3 states that siphoning of the new and spent fuel pools via piping or hose connections to these pools is precluded by the location of the penetrations, limitations on hose length, and termination of piping penetrations flush with the liner. This vacuum utilizes a 100' suction hose. This length of hose could be configured in such a way to allow siphoning. The procedure does not allow the hose to break the surface of the water while the unit is in operation. This administrative control (HPP-509) precludes any siphoning which could lower the water level in a spent fuel pool. In addition the procedure does not allow the unit to be operated in a pool containing irradiated fuel.

The change does not increase the probability or consequences of analyzed accidents, nor introduce a different type of accident or equipment malfunction than already evaluated in the FSAR. Thus, no unreviewed safety question exists.

FSAR Reference:

Section 9.1.3