CLINTON POWER STATION

10CFR50.59 REPORT

FOR

MAY 1991 THROUGH MAY 1992

DEFINITION OF ACRONYMS FOR DOCUMENTS EVALUATED

ACN	-	Advance Change Notice
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society of Testing and Materials
CR	-	Condition Report
CPS	-	Clinton Power Station
ECN	-	Engineering Change Notice
EPIP	-	Emergency Plan Implementing Procedures
FA	-	Field Alteration
FECN	-	Field Engineering Change Notice
HVAC	-	Heating, Ventilation, and Air Conditioning
IEEE	-	Institute of Electrical and Electronic Engineers, Inc.
NFPA	-	National Fire Protection Association
PDR	-	Procedure Deviation for Revision
P&ID	-	Piping and Instrumentation Diagram
TM	-	Temporary Modification
TPD	-	Temporary Procedure Deviation
USAR	-	Updated Safety Analysis Report
S	-	Supplement
R	-	Revision

CLINTON POWER STATION

10CFR50.59 REPORT

FOR

MODIFICATIONS

FROM MAY 1991 THROUGH MAY 1992

INSTALL FLANGES IN AUXILIARY STEAM SYSTEM

Document Evaluated: FA ASF017 Log Number: 91-0030

Flanges were installed between the auxiliary steam reboilers and their associated condensate drain tanks to facilitate removing the reboilers from service for maintenance. This modification was released for operation for the "A" reboiler only. The installed flanges do not affect operation or performance of the associated components. The change adds no new failure modes as other piping connected to the reboilers is also flanged.

REPLACE OBSOLETE AUTOMATIC RECIRCULATION VALVE WITH A NEW MODEL IN THE AUXILIARY STEAM SYSTEM

Document Evaluated: FA ASF018 Log Number: 91-0088

The electrode boiler feed pumps in the auxiliary steam (AS) system are provided with minimum flow valves. The present low pressure automatic recirculation control (LARC) valves are obsolete and were replaced with automatic recirculation control (ARC) valves. This is a change to the facility as described in USAR Figures 3.6-1 and 9.5-6. The function and sizing of the ARC valve is the same as the LARC valve. No malfunctions of the AS system are evaluated in the USAR.

PROVIDE CAPABILITY TO ADJUST COOLING WATER FLOW TO UPPER AND LOWER BEARINGS OF THE REACTOR RECIRCULATION PUMP MOTORS

Document Evaluated: FA CCF010 Log Number: 92-0009

The reactor recirculation pumps provide forced circulation of coolant through the reactor pressure vessel core through jet pumps integral to the reactor vessel. Each reactor recirculation (RR) pump motor has an upper and lower bearing provided with coolers. Previously, there was no means for balancing the cooling water flow between the upper and lower bearings. An investigation showed that the cooling water flow through the coolers exceeded the maximum design flows. This change provided a flow meter and a flow control valve at the outlet of each of the upper and lower bearing coolers on both of the RR pump motors. This change required a revision to USAR Figures 3.6-1 and 9.2-3 and the description of the RR system. Neither the RR pump motors nor the component cooling water system are required to operate during design basis accidents. This change will extend the operating life of the coolers by providing means to control flow through them. No new type of failure has been introduced in the USAR, nor has there been an increase in the consequences or potential for a failure previously evaluated.

UPDATE USAR TO REFLECT CHANGES TO OPERATION OF MSIV LEAKAGE CONTROL SYSTEM INBOARD AND OUTBOARD ROOM COOLERS

Document Evaluated: FA C-F054, FECN 24378 Log Number: 90-0133

Previous design changes required the Main Steam Isolation Valve (MSIV) Leakage Control System (LCS) inboard and outboard rooms, high pressure core spray (HPCS) system room, and combustible gas control system (CGCS) equipment room coolers to operate continuously during normal plant operating conditions. The temperature switches which started the room coolers were not qualified and were subsequently deleted. This change deletes drawing notes which indicate that the temperature switches will be replaced. It was determined that the respective room coolers need not run continuously during normal plant conditions. The MSIV-LCS inboard and outboard room temperatures were calculated to increase from 104°F to 108°F and 115°F respectively. The HPCS and CGCS room temperatures were determined to be within design limits. The equipment qualification packages were revised. Where the qualified life of equipment has been reduced, the affected equipment will be replaced at a higher frequency.

REPLACE THE DIVISION I BATTERY

Document Evaluated: FA DCF004 Log Number: 91-0124 & R1

Battery 1DC01E provides a reliable backup source of DC power for Division 1 equipment. Due to design of the existing battery and its limited design margin, it was necessary to increase the capacity of the battery. The existing battery was rated at 1138 ampere-hrs. The new battery is rated at 1708 ampere-hrs. The design change was reviewed for electrical load impact, structural adequacy, room temperature, hydrogen evolution, room access, fire protection, the effects of voltage drops and short circuit currents on existing components, relay settings and circuit breaker replacement. The new battery has cast lead battery posts with a copper insert, to provide structural strength, which the previous battery did not have. Problems have been encountered in the past with electrolyte attacking the copper insert causing a decline in cell voltage. New manufacturing techniques have corrected this problem. It is judged that the probabilities of a malfunction involving a decline in cell voltage have not increased. A revision to the safety evaluation allowed a partial release on the Division 1 nuclear system protection system inverter. The inverter was to be placed back in service and supplied power from the Division 1 battery charger 1DC06E, via Division 1 125-volt DC bus 1DC13E, while the Division 1 battery was being replaced.

CHANGES TO EMERGENCY DIESEL GENERATOR DOCUMENTATION

Document Evaluated: FA DGF026 Log Number: 91-0100

Each diesel generator (DG) contains an expansion tank for cooling water which is rated for 50 psi. The expansion tank provides a reservoir for cooling water as it expands. Each expansion tank has a level switch which will energize an annunciator at a predetermined low level. The expansion tank also contains a pressure cap which allows filling of the expansion tank and pressure release between 6.5 and 8 psi. The low level annunciator is not affected. A condition report identified that the expansion tank level switches for the Division I and II DGs (1LS-DG285, 286, 287, 288) were not in the seismic qualification (SQ) program even though they are seismic category I. During the condition report investigation, it was found that the above level switches were connected to safety-related 1E buses with no supporting documentation.

Field Alteration DGF026 implemented the following changes:

1. Allows non-1E Level Switches 1LS-DG285, 286, 287, 288 to be connected to 1E buses.
2. The expansion tank level switch for the Division III DG is required as a pressure boundary to prevent draining of the Division III DG expansion tank. The switch already meets the requirements for the classification change.
3. For Division I and II DGs, the design maximum pressure for the expansion tank level switches was revised from 50 psi to 16 psi. Since the fill cap for the expansion tank relieves pressure between 6.5 and 8 psi, this change will allow a 100% margin in the pressure rating of the switches.
4. USAR Table 9.5-13 was revised to include the Division I and Division II level switches, 1LS-DG285, 286, 287, 288.
5. USAR Table 9.5-13 will delete the Maintenance and Test Frequency column. Currently, the USAR Table 9.5-13 Maintenance and Test Frequency column indicates "Test frequencies will not exceed one year". The column is no longer necessary because IP's Diesel Generator Qualification Program delineates the preventive maintenance requirements for the diesels.

This evaluation indicates that these changes do not increase the probability or consequences of a malfunction of the diesel generators nor do they create a new type of malfunction.

CLOSE ISOLATION VALVE ON CO_2 FIRE SUPPRESSION STORAGE TANK

Document Evaluated: FA FPF011 Log Number: 91-0011

A carbon dioxide (CO_2) fire suppression system is provided for the three emergency diesel generator rooms. The system design conforms with National Fire Protection Association (NFPA) Standard NFPA 12, "CO_2 Extinguishing Systems". There are two CO_2 storage tanks connected in parallel to the main discharge header. This field alteration changes the normal position of the isolation valve on one of the tanks from normally open to normally closed. This change also adds a chain wheel operator so that the normally closed valve can be opened quickly. Testing has shown that with the isolation valve closed the system provides sufficient quantity and concentration of CO_2.

DEACTIVATE MAIN CONTROL ROOM FIRE PROTECTION MONITORING OF THE BLUE WAREHOUSE

Document Evaluated: FA FPF028 Log Number: 91-0084

This modification deactivates the main control room monitoring of the fire protection annunciation and alarm logging of the blue warehouse fire protection system. The change was made to remove the monitoring of the warehouse fire protection from the responsibility of the control room operators, and place it with site security similar to the other facilities outside of the protected area. The responsibility of monitoring the blue warehouse is placed where it can be more directly handled. The change in the fire protection monitoring responsibility will not affect any of the plant nuclear safety-related systems. The design change involves minor wiring changes to delete alarms and to eliminate inputs to the alarm circuits.

DELETE NOTE FROM USAR FIGURE REQUIRING SPECIFIC CHECK VALVE

Document Evaluated: FA FPF031 Log Number: 91-0064

A change was made to drawing M05-1039, sheet 6 (same as USAR Figure 9.5-1, Sheet 6, Note 6). This change was to delete the requirement that a specific model of backflow preventer be used on the discharge side of the jockey fire pump. This change allows the use of other equivalent types. This change was made because a condition report identified that the existing backflow preventer did not match the specification. The existing model is equivalent to the model specified by the note.

PROVIDE EMERGENCY POWER TO CHEMISTRY LABORATORY

Document Evaluated: FA LLF006 Log Number: 90-012

This field alteration installed emergency power to receptacles in the chemistry laboratory which are utilized for post accident sample analysis following a loss of coolant accident (LOCA). This change makes it possible for sample analysis following a LOCA that is accompanied by a loss of offsite power (LOOP). The specific change entailed connecting the transformer which provides power to these receptacles to the Division II diesel generator motor control center (MCC) 1B. Since the receptacle circuit is not Class 1E, the connection is being made through acceptable isolation devices. This additional load was evaluated by revising the diesel generator load calculation. During this revision, it was determined that, due to more recent input from the respective vendors and the mathematical corrections, the total loading on all three diesel generators has been decreased. The total loading does not exceed the name plate rating. Fuel oil storage requirements per technical specifications are not affected.

INSTALLATION OF GE8B FUEL BUNDLES

Document Evaluated: FA NBF010 Log Number: 91-0115

Field alteration NBF010 evaluated the use of the GE8B fuel design. GE6 and GE7B fuel designs are currently used. GE8B fuel has a higher helium backfill pressure which allows a higher maximum linear heat generation rate (MLHGR) (14.4 kilowatt/foot [kw/ft] vs. 13.4 kw/ft) and allows higher batch average burnup (38000 megawatt-day/metric ton [Mwd/MT] vs. 33000 Mwd/MT). In addition, GE8B fuel has a gadolinia-rich zone near the top of the bundle, which improves reactivity shutdown margin. GE8B fuel has been generically reviewed by the NRC and has the same form, fit and function as GE6 and GE7B fuels for reload design. In addition, over 5000 GE8B fuel bundles have been used in other reactors. The reason for the change is that GE8B improves the margin to MLHGR, improves cold shutdown, and reduces fuel costs. The consequences of accidents are not altered because GE8B, GE6, and GE7B fuels have the same design basis.

NRC-approved methods were used to evaluate: MLHGR, minimum critical power ratio (MCPR), maximum average plant linear heat generation rate (MAPLHGR) before each cycle for normal operation; anticipated operational occurrences; and loss of coolant accidents (LOCA) events to ensure that there is no increased fuel failure. In addition, cold shutdown and reactor stability are evaluated each cycle to ensure that there is no effect on fuel failures. Regarding defective fuel during normal operations, based on historical data, the GE8B fuel pin failure rate is 0.99999 which is an improvement over the GE7B fuel pin failure rate of 0.99994. Therefore, the consequences of failures are not changed.

Regarding the fuel bundle drop accident, GE8B fuel has the same number of fuel rods, the same uranium mass, and the same design basis as GE6 and GE7B fuels. Therefore, this design change does not alter the consequences of a fuel bundle drop accident. The basis for the conclusion is that the bundle burnup is consistent with the burnup used for the Updated Safety Analysis Report, Section 12.2.1, and the associated source term for the bundle.

Regarding an anticipated transient without scram (ATWS) event, there is an additional consideration that has to do with Emergency Guidelines, Appendix C. The change affects the maximum subcritical banked withdrawal position (MSBWP). The MSBWP is "00" for GE8B fuel compared to "02" for GE6 and GE7B fuels. This change ensures that there is no effect on fuel failure if there is a potential for an ATWS and the consequences are not altered.

Maximum core uncovery time limit (MCUTL) and minimum alternate reactor pressure vessel pressure (MARFP) in the Emergency Procedure Guidelines, Appendix C are affected because MLHGR increases from 13.4 kw/ft to 14.4 kw/ft. When multiple failures were considered, new MCUTL and MARFP were calculated to ensure that there is no effect on fuel failures during extraordinary events and the consequences of failures are not altered.

Regarding effects on fuel failures for very high bundle burnup, the evaluation concluded that there was no significant effect except for increased iodine levels for the fuel bundle drop accident. However, the batch average discharge burnup will not approach the burnup value of 29,200 Mwd/MT which is used in the analysis until the end of Cycle 5. Assessments are planned in order to determine the effect of increased burn-up capability of GE8B fuel on the source term for future cycles.

Because the helium backfill pressure in the fuel pin is higher than in the old designs, cladding stress at system pressure is reduced and heat transfer from the pellets and out of the pins is improved. Improved heat transfer will decrease fission gas release to the gap between the fuel pellet and the cladding which lowers peak pin pressure during operation. This permits the MLHGR to be increased to 14.4 kw/ft without exceeding the 1% cladding strain safety limit. This design change has no effect on the bases of the technical specifications. However, as part of the reload design, the Core Operating Limits Report will be changed to reflect 14.4 kw/ft for the GE8B fuel.

REPLACE RCIC TURBINE EXHAUST FLANGED JOINT WITH A WELDED CONNECTION

Document Evaluated: FA RIF010 Log Number: 91-0110

The reactor core isolation cooling (RCIC) system turbine exhaust line contained a flanged connection that would be exposed to containment atmosphere following drawdown of the suppression pool after a design basis loss of cooling accident. Corrective actions included an Appendix J, Type B leak rate test during the second refueling outage and replacement of the flange with welded pipe during the third refueling outage. This change eliminated the flanged connection, which eliminates the possible leakage path and the need for Appendix J testing of this penetration. The piping modification design conforms to the design criteria which is ASME Section III, Class 2, seismic category I. This change does not affect RCIC system operation.

EXPAND CONFIGURATION OF THE SERVICE AIR SYSTEM

Document Evaluated: FA SAF006 Log Number: 90-0117

This field alteration expands the service air (SA) system to include existing piping installed during the early construction phase of the plant. This piping will be used to provide an alternate path for air to be supplied to the SA header from an outside source. This equipment is upstream of the instrument air system. The design of the plant is such that a failure of SA will not compromise any nuclear safety-related system or component and will ot prevent the safe shutdown of the reactor.

INCREASE THE LOW POWER SETPOINT PRESSURE

Document Evaluated: FA TGF008 S3 Log Number: 91-0063

This field alteration changed the main turbine steam admission logic from full arc (where all four turbine control valves throttle in parallel) to partial arc where only two valves open at low power, the third as power increases, and the fourth opens only at high power. This change increases overall plant efficiency and shifts the relationship between reactor power and first stage turbine pressure (first stage turbine pressure is higher for a given reactor power). Reactor pressure and the steam pressure upstream of the turbine control valves are the same for a given reactor power as before the change. First stage turbine pressure is used as a measure of reactor power for the rod pattern controller.

To compensate for the increased first stage pressure, this supplement increases the low power setpoint from 124.5 psig to 138 psig and the lower allowable value from 101.5 psig to 115 psig. The new setpoint is based on calculations and data from post maintenance testing.

The plant technical specifications are being reworded to define the low power setpoint as a pressure setpoint using data from TGF008 testing rather than as a percent of rated thermal power. Because the USAR describes the setpoint in terms of reactor power (20%), there is no change to the USAR description. The new setpoints include conservatism to account for instrument accuracy, calibration uncertainties, and drift during the interval between calibration. The setpoint change has also been reviewed to ensure that it does not result in spurious rod blocks during normal operating conditions.

CHANGE THE COAGULANT USED IN THE TREATMENT OF RAW WATER IN THE MAKEUP WATER SYSTEM

Document Evaluated: FA WMF020 S1 Log Number: 91-0060

This change modified the coagulant used in the treatment of raw water in the makeup water system from sodium aluminate to a blend of polyaluminium chloride and cationic polymers. The change is being made to improve the reliability of water treatment equipment since the replacement coagulant is less viscous and does not precipitate aluminum oxide which clogs lines, instruments, and pumps. Water chemistry will not be adversely affected by this change. The reactor grade water quality requirements currently in place will not be revised by this change.

REPLACE SERVICE WATER FLOW CONTROL VALVE

Document Evaluated: FA WSF008, S2 Log Number: 91-0016

The plant service water (WS) system provides cooling water to the plant chilled water (WO) system chillers. Problems were experienced with the WS flow control valve (1WS079A) at the outlet of chiller 0WO02CA. The existing valve was not capable of withstanding the high differential pressure experienced when service water was at low temperatures and low flow. The valve was being damaged by cavitation. This change replaced the valve with a new type of valve better designed to control the pressure drop without cavitation damage. This change also provided for a larger diameter instrument air line to the operator for this valve. Since this operator only required an intermittent supply of air this change did not impact the instrument air system. The WO system is not required to achieve or maintain safe shutdown.

INSTALL PRESSURE TAPS IN PLANT SERVICE WATER SYSTEM TO VALIDATE FLOW BALANCE

Document Evaluated: FA WSF009 & S1 Log Number: 91-0046, 92-0044

This change installed pressure taps at various points in the plant service water system. The plant service water system provides cooling water required by station auxiliary equipment during normal station operation it is also the supply to the shutdown service water system during normal station operation and shutdown. The installation of pressure taps at selected locations in the system will allow data to be taken for verifying system flow balance calculations. The installation of these pressure taps does not impact the performance of the system.

INSTALL FLANGED PIPING IN SERVICE WATER PIPING TO FACILITATE CHILLER MAINTENANCE

Document Evaluated: FA WSF013 Log Number: 91-0061

Flanged piping was installed in the plant service water (WS) system to facilitate maintenance of the plant chilled water (WO) system 'E' chiller. The WO chillers are not safety related, also there are no safety design bases for the WS system. If WS becomes unavailable, the source of cooling water is transferred to the shutdown service water system pumps automatically. Failures of the WO chillers are not evaluated in the USAR, and this change does not introduce a new failure mode. This change does not affect the flooding analysis since leakage from the spool piece would be no worse than that already considered for this area.

DECREASE SETPOINT OF MAIN GENERATOR HYDROGEN COOLERS RELIEF VALVES

Document Evaluated: FA WSF014 Log Number: 91-0050

The main generator hydrogen coolers are cooled by plant service water (WS) and are provided with thermal overpressure protection relief valves. This change decreases the relief valve setpoint from 170 psig to 130 psig. The 170 psig was only achievable if plant service water was isolated while heat was being added to the cooler. The maximum operating pressure of the cooler is 150 psig. Thus, the cooler could have experienced a pressure greater than its design pressure at the original setpoint. Also, tube sheet repairs resulted in reducing the design pressure to approximately 148 psig. Failure of the hydrogen coolers is not evaluated in the USAR. The failure of a hydrogen cooler would be enveloped by existing analyses in the USAR.

REPLACE PLANT SERVICE WATER HEAT EXCHANGER FLOW REGULATING VALVES

Document Evaluated: FA WSF015 Log Number: 92-0027

The plant service water (WS) system provides cooling water to various systems and components in the plant. The component cooling water (CC) system is an intermediate heat exchange loop between the WS system and systems which are potentially radioactively contaminated, such as the fuel pool cooling and cleanup system. Valves 1WS018A and B regulate flow through the CC heat exchangers to maintain the CC at its desired temperature. The valves have been experiencing cavitation producing vibration and noise, and constant breakdown of these valves. This modification replaces these valves with new aspirator valves. The aspirator acts as a vacuum breaker which minimizes the cavitation inside the valve body. Support modifications and other minor piping changes were included. This change is expected to improve the operation of these valves and does not adversely impact any equipment malfunction or accident analysis.

REMOVE EDUCTORS AND ASSOCIATED SMALL BORE PIPING FROM THE SPENT RESIN TANK

Document Evaluated: FA WXF008 Log Number: 90-0054

This change removed eductors, spargers and associated piping from the spent resin tank. This change was made because the eductors, spargers and associated piping became plugged with resin during tank recirculation. This change allows the spent resin tank to be recirculated without causing plugging of the recirculation line. No failures of the spent resin tank are evaluated in the USAR. No new type of malfunction or accident is created.

DELETE NONEXISTENT TEST CONNECTION FROM BREATHING AIR SYSTEM DRAWINGS

Document Evaluated: FECN 27170 Log Number: 91-0070

The breathing air (RA) system provides breathing quality air to stations throughout the plant. It was identified that a test connection shown on the RA piping and instrumentation drawing (P&ID) had not been installed in the field. This design change deletes the test connection from the P&ID. The RA system performs no safety related function.

ALTERNATE COATING FOR DIESEL FUEL OIL STORAGE TANKS

Document Evaluated: Mod DG-062 Log Number: 92-0053

The diesel generator storage tanks and day tanks for Divisions 1, 2 and 3 are coated with a corrosion inhibitor. The coatings specified in USAR Section 9.5.4.2 are no longer available. The original paint manufacturer has specified an equivalent coating, which is compatible with the exising coating and will not react with diesel fuel oil or other fuel oil additives. The new coating is designed to handle aggressive solvents and oils.

ALLOW REACTOR OPERATION THROUGH CYCLE 4 WITH EIGHT OF TWENTY-EIGHT SHROUD HEAD STUD ASSEMBLY BOLTS REMOVED

Document Evaluated: Mod NB-028 Log Number: 92-0057

This change evaluated reactor operation through the end of cycle 4 with eight of twenty-eight shroud head stud assembly bolts removed and with three new bolts of a modified design installed. The shroud head stud assembly clamps the shroud head and steam separator assembly to the shroud. The studs are disengaged to allow for the removal of the separator assembly. During installation of the separator assembly, the proper torque is applied to the stud remotely, using a bolt which extends from the stud to the top of the steam separators. After the stud is torqued, the bolt is locked in place by a spring-loaded locking collar which has internal splices that mesh with external splices on the bolt near he head. This prevents the bolt and stud from rotating during service. The bolt has no other function during reactor operation. During the second refueling outage, four of the twenty-eight shroud head stud assembly bolts were found to have sufficient wear of the locking splices to warrant their removal since a potential loose part could result. Continued operation through cycle 3 with these four bolts removed was justified in a safety evaluation reported in the 1991 annual report. Due to continued wear of the 24 remaining bolts during cycle 3, it was necessary to remove seven additional bolts during RF-. In order to meet bolt spacing criteria assumed in the vendor's analysis three new bolts of a modified design were installed. The wear resulted from feedwater flow impinging on the bolt shafts. Ultimately, this wear could render the locking collar assembly inoperable and significantly weaken the bolt shaft. Also, several other bolts and collars have slight wear of the locking splices but are still capable of performing the locking function. Analysis showed that this condition (eight of twenty-eight shroud head stud assembly bolts removed and three new bolts of modified design installed) would not have an adverse effect on operation or response to accidents and transients. Shroud head lift, stresses, and loose parts were considered.

ADDITION OF A FLUSHING CONNECTION TO THE SX-RH CROSS TIE

Document Evaluated: Mod SX-032 Log Number: 92-0069

A cross tie between the shutdown service water (SX) system and the residual heat removal (RH) system is routed such that portions of the line have trapped silt and restrict flow. The line is not used during normal conditions. A flushing connection is required to prevent silt build-up. The tee will be blind flanged during normal operation. The only time this cross tie is used is under post design basis conditions should reactor pressure vessel flooding be needed. The addition of the flushing connection will ensure its ability to function.

ADDITION OF MANUAL ISOLATION VALVES TO THE DRAIN LINES FOR THE REHEATER DRAIN TANKS

Document Evaluated: Mod TD-007 Log Number: 92-0073

This modification added a manual isolation valve to the reheater drain tank turbine drain (TD) line. This change provides double isolation which greatly improves the isolation capability required during normal plant operation and allows for increased flexibility for the maintenance. The entire TD piping system is constructed to ANSI B31.1 Power Piping code and is categorized as non-seismic in the design specification. Impacts on existing supports, piping and equipment as applicable were evaluated.

CLINTON POWER STATION

10CFR50.59 REPORT

FOR

TEMPORARY MODIFICATIONS

FROM MAY 1991 THROUGH MAY 1992

MAIN CONDENSER SCALE INHIBITOR CHEMICAL TREATMENT

Document Evaluated: TM 91-007 Log Number: 91-0055

This temporary modification installed a chemical feed system during the third refueling outage. This system feeds scale inhibitor to the main condenser circulating water system pump suction bay. A hard calcium carbonate scale was found in the main condenser tubes during a second refueling outage inspection. The scale impedes heat transfer and causes a midsummer generating capacity loss of 6 MWe. Chemical treatment is required to prevent further scale buildup and generating capacity loss. The inhibitor being used is phosphoric acid. The added weight of the feedpump skid was evaluated and determined not to affect the seismic qualification of the screenhouse which is Seismic Category I. Interaction between the phosphoric acid and the sodium hypochlorite (currently used as a biocide) was evaluated and determined not to be a problem because of the short contact time in a once-through system. Effects on plant materials were evaluated. Also, IP determined that there would be no detrimental effect on the environment due to use of this chemical.

OPTIMIZE TIMING OF REACTOR WATER CLEANUP SYSTEM FILTER PRECOATING PROCESS

Document Evaluated: TM 91-013 Log Number: 91-0053

This temporary modification changed the times for various steps in backwashing/precoating phases of the reactor water cleanup (RT) system filter/demineralizers. The sequence and timing of the precoating process are not discussed in the USAR in detail; however, USAR 5.4.8.2 states that the time to take a filter off-line, backwash and precoat is less than one hour. This change allows the time to exceed one hour as required to optimize filter performance. This is expected to enhance the performance of the RT filters by ensuring that the filter is evenly precoated over its entire length, by reducing the amount of resin required to precoat one RT filter, and by enhancing capacity to absorb some ions (mainly chromates). The time required to take a filter off-line, backwash, and precoat is not part of the design bases.

INSTALL PIPE SURFACE TEMPERATURE DETECTORS ON STEAM CYCLE SYSTEMS PIPING

Document Evaluated: TM 91-014 Log Number: 91-0113

Thirty-four steam cycle valves were instrumented to monitor heat dissipation which has the potential to significantly reduce plant electrical power output when leakage is experienced. This type of monitoring is referred to in the industry as a Potential Loss Monitoring Program (PLMP). Eight of these valves were instrumented via temporary modification 90-069 for the pilot PLMP during the second refueling outage. The remaining twenty-six valves were instrumented during the third refueling outage to expand the scope of the pilot PLMP, for the purpose of further technical evaluation of this program. Pipe surface temperature monitoring resistance temperature devices were installed by this temporary modification. The monitoring system is entirely nonintrusive in nature.

ISOLATE INSTRUMENT LINE TO PRESSURE SWITCHES WHICH CONTROL MOISTURE SEPARATOR/REHEATER VALVES

Document Evaluated: TM 91-015 Log Number: 91-0059

One of two pressure switches on an instrument line from the main turbine cross-around steam line failed, causing a steam leak in the turbine building. A change was made to lift a lead interrupting a logic signal from the two pressure switches in order to isolate the instrument line to the pressure switches. As a result, a pressure transmitter on this instrument line was also isolated. Cross-around steam pressure is used as an indication of power level. One pressure switch is set so that the moisture separator/reheater (MSR) reheater drain tank emergency drain valves open below 20% power. The other pressure switch is set so that, below 10% power, the MSR moisture separator drain tank emergency drain valves open; the main steam-to-MSR isolation valves close; and the MSR scavenging steam block valves close. The MSR emergency drain valves fail open or automatically close at low power to purge air from the MSR during startup and to dry the MSR during turbine shutdowns. This temporary modification required a lead to be lifted to prevent these automatic actuations while the pressure switches were isolated. The pressure transmitter provides input for control of the MSR main steam control valves. However, since these valves were manually controlled, there was no impact as a result of isolating this transmitter.

INSTALL BYPASS PIPING AROUND INOPERABLE ISOLATION VALVE FOR OFFGAS HYDROGEN ANALYZER

Document Evaluated: TM 91-016 Log Number: 91-0058

This temporary modification bypassed one of the main condenser offgas (OG) system hydrogen (H_2) analyzer isolation valves (N66-F085B). The H_2 analyzers measure H_2 concentration in the OG stream. There are two redundant analyzers in parallel. Sample flow is provided by connections to the OG process piping and to the main condenser. Condenser vacuum is used to draw the sample stream through the analyzers. Each analyzer has two isolation valves on the vacuum side. One isolation valve is on the analyzer itself and the other was added during construction. In this circumstance, the "B" analyzer was operable but its isolation valve would not open. This temporary modification installed piping from the "B" analyzer to a location between the two isolation valves of the "A" analyzer. This provided an effective flow path for the "B" analyzer, making it operable. The isolation valve could not be repaired during plant operations since this could cause a loss of condenser vacuum, resulting in a plant trip. The temporary piping is constructed to the same codes and quality as the original piping.

DISCONNECT INTERFACE WIRING BETWEEN LIQUID RADWASTE DISCHARGE RADIATION MONITOR AND PANEL IN RADWASTE OPERATIONS CENTER

Document Evaluated: TM 91-017 Log Number: 91-0066

The liquid radwaste discharge monitoring instrument (OPR04D) measures the radiation concentration of the liquid radwaste discharge into the plant service water discharge header. Its main function is to provide a signal on high radiation level to close the liquid radwaste discharge isolation valve. Various controls and annunciators are provided in the main control room, the radiation protection office and the radwaste operations center and are described in USAR Sections 11.5.2.2.6 and 7.7.1.19.4. This temporary modification disconnected the wiring between the monitor and the panel in the radwaste operations center to eliminate electrical noise that has caused the monitor to be declared inoperable. Sufficient controls and annunciations remain in the main control room and radiation protection office to provide for operation of liquid radwaste discharges. The capability for the liquid radwaste monitor to initiate closure of the discharge isolation valve upon high radiation alarm was retained.

INSTALLATION OF PRESSURE GAUGES AT THE CIRCULATING WATER PUMP DISCHARGE

Document Evaluated: TM 91-028 Log Number: 91-0087

Pressure gauges were installed on existing capped root valve lines on the discharge piping of each circulating water system pump (three pumps). This was done to provide pressure data on the circulating water system. The data will be used to calculate condenser thermal performance. Since USAR Figure 10.4-3 shows the lines for these valves as capped (1CW008A, B, and C), this is a change to the facility as described in the USAR. It was judged that the temporary modification would not increase the probability of a loss of condenser vacuum since failure of the modification would not cause an appreciable loss of circulating water system flow.

INSTALL NEW SODIUM HYPOCHLORITE INJECTION SYSTEM

Document Evaluated: TM 91-031 Log Number: 91-0093

Temporary sodium hypochlorite injection equipment for the circulating water and service water systems was originally installed under temporary modification 88-044. A new sodium hypochlorite storage tank and associated isolation valves and piping were installed to replace the tanks and valves installed in the original temporary modification because they had developed leaks. Although this change does not affect the facility as described in the USAR, a 10CFR50.59 safety evaluation was performed because of potential interaction with safety-related equipment and structures in the screenhouse. The sodium hypochlorite addition reduces biofouling and the intrusion of corbicula into the circulating water and service water systems. The new installation was analyzed for the effects of spills and seismic and wind loadings on the screenhouse slab. Failure of this temporary modification will not impact the fire protection or the shutdown service water system because of separation and the provision of a berm to contain spills.

PROVIDE COOLING WATER TO THE "A" FIRE PUMP DIESEL DURING POST-MAINTENANCE TESTING

Document Evaluated: TM 91-032 Log Number: 91-0094

The "A" firepump is the normal source of cooling water for the diesel driven fire pump. During post-maintenance testing of the diesel engine, the normal cooling water supply was disconnected and it was necessary to provide a supply of cooling water from an alternate source. This temporary modification connected a cooling water supply to the diesel engine from a drain valve on the service water system. Hoses were routed through the nonsafety-related areas of the screenhouse and connected to the pressure regulating valve on the supply side of the diesel engine heat exchanger. The exchanger requires approximately 30 GPM of cooling water flow. The service water system flow was not significantly affected. During this maintenance effort, the "B" fire pump was operable, and pump "C" was available as a back-up. An operator was stationed in the vicinity of the open service water system drain valve in the event of a hose break.

BLOCK CLOSED RELIEF VALVE TO PROTECT PERSONNEL WORKING IN THE CONDENSER

Document Evaluated: TM 91-036 Log Number: 91-0116

This modification blocked closed relief valve 1B21-F408 located on the main steam (MS) supply line to the auxiliary steam (AS) system, which discharges to the main condenser. This was done to protect personnel working in the main condenser during an outage. The upstream MS valves were closed so that this portion of the system was not exposed to MS pressure. Pressure surges originating from the AS reboilers would be relieved by the relief valves on the reboilers.

OXYGEN INJECTION SYSTEM

Document Evaluated: TM 91-037 Log Number: 91-0106

A system for injecting oxygen in the condensate pumps suction piping was installed. This was done to minimize corrosion of the carbon steel and low-alloy steel surfaces in the condensate and feedwater systems. Studies have indicated that oxygen concentrations between 20-50 μg/L allow a stable oxide layer to form on the piping surfaces. Prior to this modification, feedwater oxygen level was typically 18 μg/L. This system was designed to bring that level up to 40 μg/L. During the time the injection system is in operation, data on pipe wall thinning and crud formation will be taken. Based on the industry data, oxygen injection will minimize the formation of new corrosion products and retard corrosion product release to feedwater and subsequently to the reactor. The installation was evaluated for effects of missiles and fires, and resulted in no new potential equipment failures important to safety.

TEMPORARY REPLACEMENT OF TWO REACTOR WATER CLEANUP SYSTEM PIPE SUPPORTS

Document Evaluated: TM 91-040 Log Number: 91-0122

This temporary modification converted two reactor water cleanup (RT) system pipe supports from constant supports to rigid supports. This was done to support installation of temporary shielding on the RT piping in the drywell during the third refueling outage. The temporary shielding was installed only during Mode 4 (cold shutdown) and Mode 5 (refueling) and was removed prior to the end of the outage. An evaluation determined that all pipe stresses were within ASME Section III allowable values and that all pipe supports and auxiliary and structural steel were structurally adequate.

DRAIN THE DIVISION II DIESEL GENERATOR FUEL OIL STORAGE TANK

Document Evaluated: TM 91-045 Log Number: 91-0131

During the third refueling outage, the Division II diesel generator (DG) fuel oil storage tank was temporarily modified to drain the tank. Also, an air mover was attached to the normal flame arrester connection. These changes were done while the DG was out of service for maintenance. Since the temporary modification was installed while the DG was out of service for maintenance; the fuel oil system was restored to design configuration; and the fuel oil was sampled and analyzed prior to restoring the tagout, an unreviewed safety question was not involved.

INSTALL MONITORING EQUIPMENT FOR TROUBLESHOOTING SERVICE AIR COMPRESSOR

Document Evaluated: TM 91-050 Log Number: 91-0111

This temporary modification installed four pressure transmitters and one monitor to a service air compressor. The service air system has three compressors that supply service air and instrument air for maintenance, instrumentation and controls throughout the plant. This temporary modification installed tees or connections at different tubing locations on one compressor, to allow data collection to help determine why this compressor has had pressure surge problems. All installed devices were rated for system pressure, voltage, and amperage. The operation of the temporary modification monitoring system did not impact the compressor operation. The two remaining service air compressors were not affected, nor was there an increase in the probability of occurrence or consequence of a loss of instrument air.

PLANT SERVICE WATER SUPPLY FOR CONDENSER TUBE CLEANING EQUIPMENT

Document Evaluated: TM 91-051, R1 Log Number: 92-0012, R1

This temporary modification installed a temporary water supply from the plant service water (WS) system to the condenser tube cleaning pumps used during the third refueling outage. These pumps shoot scrapers through the main condenser tubes to clean out all soft and hard deposits. Temporary hoses were attached to seven drain valves in the WS system. Revision 1 of the temporary modification installed two additional hoses attached to WS drain valves. These hoses were removed when cleaning was complete, and the configuration was restored to its normal state. The condenser tube cleaning equipment uses approximately 150 GPM of water when operating which is negligible compared to the total flow to the affected components; therefore, none of the operating characteristics of these components were affected. In the event of a shutdown service water (SX) system automatic start, all but one drain valve would have been isolated from the SX system. This event would not have affected the operation of the drywell chillers, and would not have depleted the Ultimate Heat Sink water inventory.

DISABLE ROD CONTROL AND INFORMATION SYSTEM ROD BLOCK

Document Evaluated: TM 91-052 Log Number: 91-0112

The intermediate range monitors (IRM) are neutron detectors used to monitor, record, and provide signals to both the reactor protection system (RPS) and the rod control and information system (RC&IS) during reactor startup, heatup, and shutdown. The IRM channels produce a trip of both the RPS and RCIS rod block when placed in the inoperable position. To meet the requirements of the technical specifications, the channel must be placed in the tripped position creating a scram signal for that channel. This charges the scram logic to one-out-of-three since one trip signal is already present. However, since the rod block occurs when any one channel is in the tripped condition, there is a continuous rod block present. This temporary modification placed the IRM C channel in bypass for the rod block signal while not affecting the scram signal. The bypassing of the a single IRM channel reduces the associated channel trip logic to one-out-of-one. However, the bypassing of the IRM C rod block signal is allowed within technical specifications.

The reason for the temporary modification is that IRM C is reading erratically (0 to 125%) and is generating both reactor scram and rod block signals. To allow the plant to startup while satisfying the technical specifications, requires only six operable IRMs; therefore, IRM C rod block is being bypassed. The continuous rod withdrawal evaluation in USAR 15.4.1.2 is not considered credible during startup. USAR Chapter 7 lists the rod block as part of the power generation design bases and not the safety design bases. No other IRM channel can be tested with the temporary modification installed since testing would satisfy the two-out-of-four trip requirements causing a scram. The evaluation showed that failure of the temporary modification due to a seismic event would generate a rod block.

REMOVE THE CURRENT FLOW COMPARATOR INPUT TO THE PROLONGED LOSS OF STATOR COOLING TURBINE GENERATOR RUNBACK CIRCUIT

Document Evaluated: TM 91-064 Log Number: 91-0128

This temporary modification removed one of the three inputs to the prolonged loss of stator cooling turbine generator runback circuit. This circuit protects the generator from damage due to loss of cooling water to the generator stator. The prolonged loss of stator cooling runback is described in the USAR, although the inputs to this circuit are not. Several spurious turbine generator runbacks occurred due to failure of the current flow comparator. With the remaining inputs to the runback circuit capable of performing their function, the turbine generator will still trip if the runback is not completed within a set time.

FABRICATE MOUNTING BRACKET AND WIRING HARNESS FOR ROD POSITION INFORMATION SYSTEM POWER SUPPLY MODULE

Document Evaluated: TM 91-067 Log Number: 91-0130

This temporary modification replaces a mounting bracket and wiring harness for a failed power supply in rod control and information system position indication circuit. The mounting of the temporary power supply module is identical to the original and will not become a missile during a seismic event. Wiring sizes of the new harness were also evaluated as acceptable.

BLOCK OPEN CONTAINMENT ISOLATION VALVES FOR SERVICE AIR, INSTRUMENT AIR, BREATHING AIR, AND EQUIPMENT/FLOOR DRAINS

Document Evaluated: TM 91-069 Log Number: 91-0121

This temporary modification blocked open several air-operated containment isolation valves using stem collars so service air, instrument air, breathing air, and equipment and floor drain flow could be provided during the third refueling outage. The air supply to these valves was removed while the instrument air system was out of service for maintenance. The stem collars were installed to prevent the valves from closing due to the loss of air. Also, jumpers were installed for the equipment and floor drain sump pump circuitry to prevent the sump pumps from tripping. These containment isolation valves were not required to be operable in Mode 4 (cold shutdown) or Mode 5 (refueling and core alterations) during the period when this temporary modification was installed.

DEFEAT FUEL HANDLING PLATFORM INTERLOCKS

Document Evaluated: TM 91-070 Log Number: 91-0126

The normal operating mode of the fuel handling platform is for moving new and spent fuel between the inclined fuel transfer system (IFTS) cart and the spent fuel pool. These transfers are completed under water. Interlocks are provided to prevent refueling bridge crane movement through the gate between the spent fuel pool and the IFTS transfer pool if the refueling mast is not centered on the gate. Other interlocks prevent the IFTS cart movement into the IFTS transfer pool area unless the upender is in the vertical position. This temporary modification set up the fuel handling platform to move new fuel from the storage vault to the spent fuel pool. For this operation, the fuel handling mast is stowed in a horizontal position, and the auxiliary hoist, rather than the mast, is used to transport new fuel. The new fuel is lifted vertically until the bottom of the fuel bundle is above the top of the fuel pool. The fuel is then moved laterally over the IFTS transfer pool area and then lowered to its temporary location in the spent fuel pool. This temporary modification defeated the interlocks in preparation for the new fuel transfer. Also operation from the auxiliary hoist/monorail pendant would normally require a second operator to press the bridge override pushbutton at the main control panel. A jumper was installed to allow operating the bridge from the auxiliary monorail pendant without the need for the second operator. During this operation, the new fuel was not moved over spent fuel.

LOWER SETPOINT FOR FUEL POOL COOLING AND CLEANUP PUMP MOTORS COOLING WATER LOW FLOW TRIP

Document Evaluated: TM 92-002 Log Number: 92-0028

During the 4160 volt 1A bus outage during the third refueling outage (RF-3), the service water (SX) system and component cooling water (CC) system were removed from service at the same time. During this time, shutdown service water, which is colder than CC system water, was used to cool the fuel pool cooling and cleanup (FC) system pump motors. Technical specifications require that reactor vessel water in the upper containment pool be kept above 68°F to maintain the reactivity shutdown margin. The USAR requires that spent fuel pool water also be kept above 68°F for the same reason. Furthermore, because RF-3 was conducted during the spring, the lake water (shutdown service water) was colder than normal. Because of the colder temperature of the SX water, it was necessary to reduce SX flow to the FC heat exchangers to prevent cooling the pools below 68°F. Pool temperatures and SX flows were monitored and controlled administratively. This temporary modification specifically lowered the trip set points on the FC pump motor's cooling water flow indicating switches from 25 GPM to 8 GPM in order to prevent spurious trips of the pumps with the reduced SX flow. Calculations demonstrated that 8 GPM is adequate to remove the heat load with the cooling water at or below 50°F, which was the case while this temporary modification was installed.

TEMPORARY ELECTRICAL BACKFEED FROM SWITCHYARD THROUGH THE MAIN POWER TRANSFORMERS TO THE UNIT AUXILIARY TRANSFORMERS

Document Evaluated: TM 92-004 Log Number: 92-0005

Due to a failure of the "B" phase main power transformer (MPT), it was necessary to backfeed electrical power from the switchyard through the MPTs to the unit auxiliary transformers to test the replacement MPT. This temporary modification removed the reverse power relays, adjusted the power imbalance relays to give maximum protection, installed a jumper to keep the turbine running on its turning gear, and installed a jumper to keep the turbine control valves closed. The generator was off-line and physically disconnected from the isophase bus ducts during this installation.

BLOCK OPEN OFFGAS DESICCANT DRYER INLET VALVE

Document Evaluated: TM 92-011 Log Number: 92-0017

The off-gas system has two desiccant dryers in parallel. The dryers dry the off-gas effluent to approximately -50°F dew point prior to going to the gas cooler and charcoal beds. Moisture is removed from the off-gas to prevent freezing and blocking the charcoal beds. One desiccant dryer is in service while the other dryer is being regenerated or is in standby. During a normal transfer between the inservice and standby dryers, the "B" dryer inlet valve, 1N66-F012B, failed to open. This temporary modification blocked this valve open. Operators used the manual inlet isolation valve 1N66-F121B to manually isolate/open the dryer. These valves are shown on USAR Figure 3.6-1, Sheet 75. Administrative controls were implemented to ensure proper operation of the 1N66F121B valve.

RELOCATE HYDROLASERS USED FOR LEAK RATE TESTING DURING RF-3

Document Evaluated: TM 92-012 Log Number: 92-0023

Two high-pressure hydrolasers are used during refueling outages to conduct inservice inspection leak rate tests. This temporary modification relocated the hydrolasers to the northwest area of the fuel building, elevation 755', during RF-3. This installation was reviewed for its effect on the Fire Protection Evaluation Report (FPER), for possible effects of water spray due to hose failures, for internal flooding, and for structural and seismic considerations.

BLOCK OPEN OFF-GAS DESICCANT DRYER INLET VALVE

Document Evaluated: TM 92-013 Log Number: 92-0019

During a normal transfer between the inservice and standby dryers, the inlet valve of the "A" dryer failed to fully open. This temporary modification blocked valve 1N66-F012A open. A second manually-operated valve was used to isolate the "A" dryer during its regeneration cycle. A similar problem occurred on the "B" dryer inlet valve four days prior to this occurrence. Administrative controls were placed on operation of the manual inlet isolation valve. Failure of operations to properly manipulate the valve could cause a loss of off-gas flow, and subsequently, a loss of condenser vacuum which is an accident evaluated in the USAR (Section 15.2.5.). A loss of vacuum caused by operator error in manipulating this valve would be less severe than that evaluated in USAR 15.2.5 because the vacuum decay would be slower.

INSTALL WINDOW ON DRYWELL AIR PARTICULATE MONITOR

Document Evaluated: TM 92-018 Log Number: 92-0022

The drywell air particulate monitor provides a secondary method for detecting a leak in the reactor coolant pressure boundary within the drywell. The air particulate sample panel filters out representative samples of gaseous and effluent particles, trapping the particles on filter paper. A gear drive assembly advances the paper, moving the sample to a beta scintillation detector. As originally installed, the paper path was totally enclosed and not visible without opening up the panel. On several occasions the paper advance mechanism failed. Because the failures could not be observed, they were not discovered until the next equipment disassembly, which was typically the preventative maintenance task to replace the filter paper. The equipment was therefore inoperable without the knowledge of the operators and in violation of the technical specifications. This temporary modification installed a clear viewing window in the cover of the compartment for the supply filter paper reel. This change allows direct observation of the reel which can only rotate if the paper is moving. This change allows identification of a failure in time to effect a repair and avoid a violation of technical specifications. The temporary modification was evaluated for seismic and environmental considerations and found acceptable.

MOBILE TOOL DECONTAMINATION FACILITY

Document Evaluated: TM 92-023 Log Number: 92-0030

This temporary modification consisted of a mobile tool decontamination facility during the third refueling outage. The facility was self-contained and was composed of two 40-foot by 8-foot vans and one 40-foot flatbed trailer. The system cleaned radioactively contaminated tools by means of a high velocity delivery of carbon dioxide (CO_2) pellets. The vans contained the pellet cleaning enclosures, ventilation equipment, pelletizer, air drying equipment, electrical distribution equipment, and a clean storage area. The flat bed trailer held a 14-ton CO_2 storage tank. The facility was set up outside the power block at the southeast corner of the fuel building. The safety evaluation addressed: the possibility of airborne releases of gases and particulates; the potential risks to main control room habitability and diesel generator operability due to rupture of the CO_2 tank or failure of components such as the safety relief valve; effects on secondary containment; additional loads on plant structures; seismic events; tornados; and fire protection.

INSTALL REACTOR WATER LEVEL TRANSMITTER SIMULATOR ON ANALOG TRIP MODULE

Document Evaluated: TM 92-032 Log Number: 92-0047

During the third refueling outage (RF-3), due to maintenance activities, it was necessary to remove the sensing line to reactor water level transmitter 1B21-N080D. This temporary modification disconnected the sensing line and installed a transmitter simulator on the analog trip modules (ATM) 1B21-N680D and 1B21-N683D. USAR Figure 5.1-3 shows the connection between the transmitter and the ATM. This ATM provides actuation signals to the reactor protection system (RPS) and the containment and reactor vessel isolation control system (CRVICS). This instrumentation is covered by Technical Specifications 3.3.1-1.4, 3.3.1-1.5, and 3.3.2-1.5.C; however, the temporary modification was installed only during modes 4 or 5 when this instrumentation is not required to be operable. Normally the logic for this trip is two-out-of-four. Without installing the transmitter simulator, this channel would be in a tripped condition meaning that the logic would be one-out-of-three. This would make the trip more susceptible to inadvertent isolations of RHR shutdown cooling. By installing the transmitter simulator and inserting a simulated signal, the logic was effectively two-out-of-three. This change inhibited automatic isolation of several residual heat removal (RHR) outboard containment isolation valves including some in the shutdown cooling loop. Manual isolation was still available using an isolation pushbutton or valve handswitches.

REVISE SEQUENCE OF VENTING THE REACTOR WATER CLEANUP SYSTEM FILTER/DEMINERALIZERS

Document Evaluated: TM 92-034 Log Number: 92-0054

The reactor water cleanup system filter/demineralizers remove solid and dissolved impurities from reactor coolant. The filter/demineralizer units are pressure precoat type filters using a filter media and mixed ion-exchange resins. Problems were experienced with the sequence of venting the units prior to precoating. Previously the vent valve closed first, allowing entrapped air to become pressurized in the filter/demineralizer dome area. Subsequently, when the precoat return valve opened, the precoat tank would overflow creating a contamination area at the 803-foot elevation of the containment building. The problem was controlled administratively by stationing an operator to manually throttle the precoat return valve. This temporary modification corrected the problem by revising the length of various timers so that the filter/demineralizers vent valve closes last. The timing sequence is given in USAR Figure 5.4 19, Table II. Another change disconnected the precoat tank dust collector, which was originally installed to remove nuisance dust particulates from a previous precoat material which is no longer used.

DRAIN THE DIVISION III DIESEL-GENERATOR FUEL OIL STORAGE TANK

Document Evaluated: TM 92-035 Log Number: 92-0051

A temporary modification was installed which provided a method for draining the Division III diesel generator (DG) fuel oil storage tank so that the tank could be cleaned during the third refueling outage. Also installed was an air mover attached to the normal flame arrester connection. These changes were made while the DG was removed from service for maintenance. An unreviewed safety question was not involved since the temporary modification was installed only while the DG was removed from service for maintenance, the fuel oil system was restored to design configuration, and the fuel oil was sampled and analyzed prior to restoring the DG to an operable status.

CONTINGENCY CROSSTIE OF DC ELECTRICAL BUSES DURING DIVISION 1 BATTERY REPLACEMENT

Document Evaluated: TM 92-037 & R1 Log Number: 92-0050 & R1

Modification DCF004 replaced the Division 1 125-volt DC battery with a new, larger battery during RF-3. While the battery was being replaced, the Division 1 DC loads were powered by the battery charger. A failure of the battery charger at this time would result in a power loss on the 1A bus. This would result in power being lost to the normal supply of the Division 1 nuclear system protection system inverter, several reactor core isolation cooling system valves, and the 125-volt DC 1A distribution panel.

This temporary modification was installed as a contingency plan in the event of a battery charger failure. It consisted of a crosstie between the 1A and 1F buses so that the 1F battery charger could be used as a secondary source of power for the 1A bus. The connection complies with the requirements for circuit protection for 1E buses connected to non-1E sources. Administrative controls were in place to maintain loads on the 1F bus within the bus design capability. This change did not involve a change to the technical specifications. It was installed while the plant was in mode 4 or 5 when only one division was required to be operable.

A revision to TM 92-037 allowed the operational (non-contingency) use of the temporary modification. The temporary modification was utilized to ensure that a primary and secondary source of power were present at the Division 1 bus. This was done by connecting the Division 1 battery charger in parallel with the non-divisional 1F battery and battery charger upon installation of the crosstie, and then turning off the Division 1 battery charger. This ensured that power was maintained at the Division 1 bus. The parallel connection of these power sources was used for the limited time required for switching the circuit breakers. The non-divisional 1F bus has 17,200 amps of available fault current. With the Division 1 charger also connected to the Division 1 bus during this crosstie, an additional 375 amps of fault current is introduced for a total of 17,575 amps. This value of fault current remains within the ratings of the breakers on the Division 1 bus, except for the breaker located in compartment 13A. This spare breaker was not used during this temporary modification installation since all loads on the Division 1 bus were shed with the exception of the 125-volt DC distribution panel and the emergency lighting cabinet. The expected load for this crosstie includes running the Division 1 diesel generator.

DRYWELL TEMPERATURE MONITORING (Elevation 800-Foot Azimuth 22-Degrees) TO DETERMINE LOCALIZED HOT SPOTS.

Document Evaluated: TM 92-040 Log Number: 92-0071

Condition Report (CR) 1-92-03-016 identified damage to an instrumentation cable (1RI16C) for a reactor core isolation cooling system check valve. Also, CR 1-89-01-016 and 1-90-10-070 identified damage to cables 1RI16C and 1RI16J during the first and second refueling outages. This temporary modification implemented the action plan to monitor temperature profiles in the drywell (elevation 800-foot, azimuth 22-degrees). This temperature monitoring system included cables and resistance temperature devices. Hardware will be mounted on the junction boxes and conduits. The temperature will be monitored from the containment building through permanent cables routed through an exiting drywell penetration. This change has no impact on any permanently installed temperature elements.

INSTALLATION OF BLIND FLANGE ON CONTROL ROOM HEATING VENTILATION AND AIR CONDITIONING (HVAC) CHARCOAL BED DELUGE LINE

Document Evaluated: TM 92-043 Log Number: 92-0060

This temporary modification installed a blind flange on the 0VC075B filter unit charcoal bed deluge line to make the control room HVAC train "B" operable. This change allowed maintenance to be performed on valve 1SX076, assuring that the control room HVAC "B" train remained operable. A blind flange on the 0VC075B charcoal bed deluge connection impaired automatic deluge fire protection capability, but compensatory measures were in place per CPS procedures. Hourly fire watches of this area were performed. The blind flange ensured leak tightness of the control room ventilation system filter unit.

BLOCK OPEN VALVES 1IA005, 1IA008, 1SA029, AND 1SA032

Document Evaluated: TM 92-050 Log Number 92-0076

This modification blocked open drywell isolation valves 1IA005, 1IA008, 1SA029 and 1SA032, on the service air and instrument air systems during the third refueling outage. This was done to perform maintenance on the limit switches. These valves fail closed when control power is lost. These valves were blocked open with a stem collar. By mechanically blocking open these valves, the associated systems could still be utilized. This change disabled the automatic isolation of the valves as mentioned above, however, these valves were not required to be operable during this mode of operation.

BLOCK CLOSE VALVE DURING NITROGEN PURGE OF CHARCOAL ADSORBERS

Document Evaluated: TM 92-054 Log Number: 92-0082

This temporary modification blocked closed inlet valve 1N66F131 to the refrigerated absorber vault in the offgas system. The purpose for maintaining the inlet valve closed was to allow purging of the charcoal adsorbers with nitrogen while eliminating the potential for inadvertent introduction of oxygen into the charcoal adsorbers until the purging operation was completed. The valve was restored to its normal state once the purging operation was completed. The change allowed contrclled purging of the adsorbers. The system was operated in the bypass mode as described in USAR Section 11.3.2.6.3. The nitrogen purge was maintained per CPS Procedure 3215.01. The nitrogen flow prevented backflow of offgas into the charcoal adsorbers. Overall, system operation remained per design as described in the USAR Section 11.3.2.1.6.3..

CLINTON POWER STATION

10CFR50.59 REPORT

FOR

PROCEDURES AND DOCUMENTS

FROM MAY 1991 THROUGH MAY 1992

HELIUM LEAK TESTING

Document Evaluated: CPS Procedure 2800.11 R4, 2800.11D003 R0 Log Number: 92-0080

This procedure revision allows the use of Sulfur Hexafluoride (SF_6) to identify condenser tube leaks with the main condenser and circulating water system in service. For any single injection, SF_6 will be injected into the circulating water system at a rate not to exceed 15 standard cubic feet per minute for one and one-half minutes. If condenser tube leakage exists, the SF_6 gas will be detected in the offgas system stream through the use of an SF_6 gas detector. If condenser tube leakage is detected, one condenser waterbox will be isolated and drained at which time helium will be utilized as the tracer gas to identify which tubes are leaking. Condenser tube leak detection, using SF_6, has been utilized at numerous other nuclear facilities. The nuclear steam supply system vendor evaluated the use of SF_6 for condenser leak testing at CPS and recommended the method described above. Chemistry sampling of reactor coolant will monitor conductivity, chloride and sulfate levels to ensure these parameters remain within the appropriate levels specified in CPS procedures.

RAISE MAIN TURBINE BYPASS LOAD LIMIT FROM 105% TO 110%

Document Evaluated: CPS Procedure 3004.01 R12, PDR 91-C266 Log Number: 91-0062

During normal plant operation, steam pressure is controlled by the turbine control valves which are positioned in response to the pressure regulator signal. The turbine control valve demand signal is limited to that value required to fully open the turbine control valves. Thus, if the pressure control system requests that additional steam flow be relieved from the reactor when the control valves reach wide open, a control signal will cause the main turbine bypass valves to open. The limit for the pressure setpoint is called the load limit. This procedure revision raises the load limit setpoint from 105% to 110%, during periods when the moisture separator/reheaters are not available. At 100% reactor power, with the moisture separator/reheaters not in service, more steam is sent to the high pressure turbine. To prevent the main turbine bypass valves from opening with the turbine control valves fully open, the load limit was raised. This change allows increased steam flow to the main turbine within its design capability as described in USAR Section 10.2. Turbine control valve close stroke time is not affected by this change. The main turbine will still be operated below the flows and pressures used in the accident analyses.

OPERATE FEEDWATER PUMP MINIMUM FLOW CONTROLLER IN MANUAL ONLY

Document Evaluated: CPS Procedure 3101.01 R9 Log Number: 91-0372

USAR Section 10.4.7.5 states that the reactor feedwater pumps are provided with flow controls that automatically regulate pump recirculation. The pumps require a minimum flow of 5000 GPM each. To provide minimum flow during periods of low feedwater flow, a minimum flow recirculation line with discharge to the main condenser is provided. Controls are provided which allow operation in either automatic or manual. Because of problems encountered with the automatic mode, operations has opted to operate the system with the minimum flow controller always in manual. This change deletes the USAR description of the FW pump minimum flow control as automatic. This USAR change reflects a change to the operating procedure that states automatic operation is not allowed. None of the USAR Chapter 15 accident analyses are affected by this change.

PROVIDE INSTRUCTIONS FOR SUPPLYING CYCLED CONDENSATE TO CONTROL ROD DRIVES DURING PLANT OUTAGES

Document Evaluated: CPS Procedure 3304.01 R14 Log Number: 91-0125

This procedure change allows the use of cycled condensate system water to provide cooling water to the control rod drive (CRD) mechanisms. Continuous flow of high quality water is desired to cool the mechanical seals. Normally, the CRD pump oil coolers are cooled by the turbine building closed cooling water system which is cooled by the plant service water (WS) system. During plant refueling outages, when it is necessary to shut down the WS system to perform maintenance, the CRD pumps must be shut down since their oil coolers will not be supplied with cooling water. This procedure change allows high quality water to flow through the CRD mechanisms during WS system outages through a cross connection to the cycled condensate system. Flexible hoses with in-line filters are used to make the cross connection. If a hose were cut or were to burst, installed check valves would prevent backflow from the reactor vessel.

USE OF LOW PRESSURE COOLANT INJECTION (LPCI) FLOWPATH FOR SHUTDOWN COOLING

Document Evaluated: CPS Procedure 3312.01 R17 Log Number: 92-0043

The residual heat removal (RHR) system is designed to remove decay heat from the reactor during shutdown conditions in the shutdown cooling (SDC) mode. Either RHR loop can be used depending on the decay heat load and the required cool down rate. In the normal lineup of an SDC loop, suction is taken from the reactor recirculation (RR) "B" pump inlet line, pumped through the respective RHR heat exchanger (HX), and returned to the reactor vessel through one of the two feedwater (FW) lines. CPS Procedure 3312.01 was revised to allow the use of the LPCI injection lines for SDC return to the reactor vessel during the second refueling outage (RF-2). The configuration would allow one RHR loop to be operating in its normal SDC lineup with return to the reactor vessel through one of the FW lines. The other loop would be lined up as a backup shutdown cooling system with return through its respective LPCI injection line. This would enable the second FW line to be shutdown for maintenance. The safety evaluation documented the availability of the LPCI flowpath during future refueling outages.

In late 1981 or early 1982, the Kuo Sheng (BWR-6) plant experienced damage to an in-core instrument tube and surrounding fuel bundles due to jet impingement on the core while using LPCI injection piping for SDC. A change was made during CPS construction to add flow deflectors at the injection nozzle, to replace the in-core instrument tube with a strengthened tube and to instruct operators to use LPCI only for accident or emergency situations. Allowing operation of SDC through an LPCI injection line in which suction is taken from the suppression pool requires several restrictions to be placed on using the LPCI lines for SDC. The time allowed for operations in this mode was limited to less than 96 hours. In addition, inspection of the six fuel assemblies in the immediate vicinity of the injection nozzle would be required prior to startup if this lineup were used. Also, flow rates were restricted below 6000 GPM.

PROVIDE FOR ALTERNATE SHUTDOWN COOLING METHOD USING UPPER CONTAINMENT POOL MAKEUP DUMP LINES

Document Evaluated: CPS Procedure 3312.02, R3 Log Number: 92-0031

This procedure revision adds another method for alternate shutdown cooling (ASDC). This method, which is used during refueling outages with the reactor pressure vessel (RPV) head removed, provides for returning water from the RPV through the upper containment pool dump lines to the suppression pool. The previous ASDC methods enabled water to be returned to the suppression pool using three of the main steam relief valve discharge lines. Any one of the five emergency core cooling system pumps may be used to take suction from the suppression pool and inject into the RPV.

Several prerequisites assure safe operation in the new alternate mode. No fuel assemblies are allowed to be stored in the upper pools. No fuel assemblies or control rods are allowed to be handled within the RPV. All refueling operations in the containment building are suspended. The suppression pool water level is lowered to accommodate the additional volume of water. If the high pressure core spray pump is used, it is aligned to the suppression pool to prevent addition of water from the reactor core isolation cooling storage tank to prevent overflow of the drywell weir wall. Since fuel movement is not allowed while this flow path is in use, there is no impact on the analysis of a dropped fuel bundle in the containment or any other USAR analysis.

ALTERNATE SUPPRESSION POOL COOLING USING FUEL POOL HEAT EXCHANGERS

Document Evaluated: CPS Procedure 3318.02, R1 Log Number: 92-0014

This procedure provides an additional method of decay heat removal from the reactor vessel. This procedure provides for an abnormal suppression pool cooling configuration in which suppression pool water is pumped through one of the two fuel pool cooling (FC) heat exchangers. This was originally reported in IP's 1990 50.59 annual report. This latest safety evaluation was performed to satisfy the prerequisites and limitations of this procedure which require an analysis of decay heat levels, the expected heatup rate of the spent fuel pools, and the time required to restore the systems to their normal configuration prior to each use of the procedure. This evaluation showed that there was adequate heat removal capability using one FC heat exchanger during the third refueling outage (RF-3). Conservative limits are provided to assure safe operation.

DISABLE REACTOR WATER CLEANUP SYSTEM PROTECTIVE CONTROL FEATURES DURING A DIVISION I BUS OUTAGE

Document Evaluated: CPS Procedure 3509.01C001 R0 Log Number: 92-0039

Two protective features of the reactor water cleanup system (RT) will be bypassed during a Division I bus outage. These protective features are designed to trip the pumps on closure of the outboard containment isolation valve or on low flow indication. The purpose of this change is to allow continuous running of the RT pumps. These features may be bypassed in plant operational modes 4 or 5 with no handling of irradiated fuel in secondary containment, no core alterations, and no operations with a potential for draining the reactor vessel. Division II isolations and trips will continue to function during this outage; however, these isolations are not required to be operable during the operational modes in which this modification is implemented.

DISABLE AUTOMATIC TRIP OF REACTOR WATER CLEANUP PUMPS DURING A NUCLEAR SYSTEM PROTECTION SYSTEM DIVISION II BUS OUTAGE

Document Evaluated: CPS Procedure 3509.01C002 R0 Log Number: 92-0038

The control system for reactor water cleanup system pumps is designed to trip the pumps if the inboard containment isolation valve closes. This modification allows installation of a jumper to prevent the pump trip when the control power is not available during a Division II bus outage. The Division I interlock will still be operable. This modification is only allowed in operational modes 4 or 5 with no handling of irradiated fuel in secondary containment, no core alterations, and no operations with a potential for draining the reactor vessel.

TEMPORARY ELECTRICAL POWER TO FIRE PROTECTION PANELS DURING A 4160 VOLT BUS OUTAGE

Document Evaluated: CPS Procedure 3514.01C005 R0 Log Number: 92-0036

This procedure allows the installation of a temporary power supply to fire protection panels while the 4160 volt bus 1A1 is taken out of service for maintenance. If the temporary power source were to fail, a loss-of-AC-power alarm will sound in the main control room and the power source will automatically transfer to a 24-hour battery backup.

PROVIDE TEMPORARY POWER TO FIRE PROTECTION PANELS DURING A 480 VOLT SUBSTATION OUTAGE

Document Evaluated: CPS Procedure 3514.01C010 R0 Log Number: 92-0037

This procedure allows the installation of a temporary power supply to fire protection panels during an outage of the 480 volt substation 1A. If the temporary power source were to fail, a loss-of-AC-power alarm will sound in the main control room and the power source will automatically transfer to a 24-hour battery backup.

DIESEL GENERATOR OVERSPEED TRIP TEST

Document Evaluated: CPS Procedure 3882.01 R1 Log Number: 92-0033

The Division I and II emergency diesel generator overspeed trip test procedure was revised to allow tripping one of the engines while the other engine continues to operate. This change allows testing the overspeed trip setpoint for each engine to ensure both engines are in the required overspeed trip range. Each engine is equipped with an overspeed trip. When the trip is actuated, crosstie logic also trips the other engine. This revision provides for disabling the crosstie logic so that the overspeed trip on each engine can be tested. This test method was used at startup and was recommended by the vendor. The diesel generator will not be operable, per Technical Specifications, during this configuration.

CHANGE TO STANDBY LIQUID CONTROL SYSTEM DRAIN PATH

Document Evaluated: CPS Procedure 9015.01 R31, PDR 91-0411 Log Number: 91-0108

The surveillance procedure for ensuring the operability of the standby liquid control system was revised to allow waste water to be drained to the containment building drains and treated in the liquid radwaste treatment system. Tests have shown that boron levels following liquid radwaste treatment system processing are below levels that would be of concern in the reactor water. This change was evaluated against an event at another plant where boron was not removed by demineralizers prior to reaching the reactor vessel. The liquid radwaste treatment consists of a two-step process of evaporation followed by demineralization. In this process, it is highly unlikely that significant amounts of boron will contaminate the reactor coolant.

REACTOR CORE ISOLATION COOLING (RCIC) OPERABILITY CHECKLIST PROCEDURE

Document Evaluated: CPS Procedure 9054.01 R27, 9054.01D001 R29 Log Number: 91-0097

The reactor core isolation cooling (RCIC) system surveillance procedure was revised to incorporate changes made in the inservice inspection program and the quick start section for RCIC turbine startup. The changes made to the procedure allow the cycling of the RCIC pump minimum flow valve, 1E51-F019, at 1000 psig discharge pressure rather than 300 psig. This valve is designed to open at a differential pressure of 1400 psi. The turbine governor system is capable of controlling turbine speed and system flow within limits with the 1E51-F019 valve open or closed.

BIOCIDE TREATMENT OF WASTE SLUDGE LINERS

Document Evaluated: CPS Procedure CPS-P-03-001 Log Number: 92-0025

This procedure provides a method to disinfect a waste sludge liner with a 0.5 percent concentration of glutaraldehyde prior to commencing the routine dewatering. Sample analysis of liners has indicated the presence of biologically-produced methane and carbon dioxide gas. Following treatment, the biocide is removed as the liner is dewatered. The removed liquid is distilled, filtered, and demineralized. Then it is sampled to determine suitability for return to cycled condensate. The chemistry group has total organic carbon detection capability. The glutaraldehyde content in the processed water is determined prior to return to the cycled condensate system. The equipment necessary for biocide addition is compatible with the materials of the equipment in which it is used. Continuous gas monitoring will be employed during biocide treatment as a precautionary measure.

SUPPLEMENTAL EVALUATION OF RADWASTE SOLIDIFICATION SYSTEM

Document Evaluated: CPS Procedure PCP-03-003 Log Number: 91-0132

The CPS radwaste disposal process uses a vendor-supplied mobile dewatering and solidification system. A safety evaluation, prepared in 1989, covered implementation of this process. The safety evaluation was supplemented in 1991 to address the waste delivery system, the radwaste overhead bridge crane, the turbine building crane, the electrical supply, and the air supply. This safety evaluation also included a realistic analysis of a liner drop event. The analysis demonstrated that the limits of 10CFR20 would not be exceeded by this event.

REACTOR PRESSURE VESSEL (RPV) STEAM DRYER CRACK

Document Evaluated: CR 1-89-01-162 R2 Log Number: 92-0066

During the first refueling outage (RF-1), a crack in a reactor vessel steam dryer was discovered. At that time, the crack extended approximately 6 7/8 inch from the bottom end of a channel weld. Inspections performed during the second refueling outage (RF-2) showed that a 1/4-inch ligament at the bottom of the weld was cracked. In addition, it was determined that the upper end of the crack had grown approximately 3/8 inch. It was also judged that the separation between the channel plate and skirt had increased from that measured during RF-1. Comparison between the RF-2 and RF-3 inspection results has shown that the crack growth rate has slowed. The growth during cycle 3 was found to be only approximately 1/8 inch. Based on a comparison of the RF-1, RF-2 and RF-3 inspection results, it is not anticipated that growth beyond acceptable limits will occur during the fourth operating cycle.

REVISE USAR TO REFLECT ADMINISTRATIVE CHANGES, USE OF SINGLE-USE RESINS AND WX SYSTEM OPERATIONAL CHANGES.

Document Evaluated: CR 1-90-03-087 R0 Log Number: 92-0088

The change revises the USAR to reflect current CPS practices for the use of single-use resins. The change also adds source term information to reflect the use of toss-away resins and the analysis comparison to 10CFR20 limits. For regeneration and non-regeneration modes of operation, the radiological consequences remain approximately equal to or less than the phase separator or the concentrate waste tanks.

CLOSE DIVISION II SHUTDOWN SERVICE WATER PUMP MINIMUM FLOW LINE CONTROL VALVE

Document Evaluated: C[illegible]-01-020 R1 Log Number: 92-0055

Valve 1SX173B provides a minimum flow bypass of the division 2 residual heat removal (RHR) system heat exchanger for the shutdown service water (SX) pump. Post maintenance testing during the second refueling outage indicated leakage past the valve seat. The valve could not be replaced during the RF-3 outage due to scheduling constraints. The valve will remain closed until it can be replaced during RF-4. The continuous flow through the RHR heat exchanger will prevent microbiologically induced corrosion (MIC). The flow rate is below the velocity at which the heat exchanger tubes would be damaged. There will be no significant impact on service life since there is no heat load on the shell side of the heat exchangers during normal plant operation. The pump operation was evaluated, and although the pump motor would be subjected to a higher starting current for a longer time, there is still adequate margin between the motor starting current and the over-current trip relay setting. Also, the estimated amount of leakage through the valve (<100 GPM) will not impact the effectiveness of heat exchangers supplied with cooling water by SX.

SHIELDS SOIL SERVICE FACILITY EVALUATION

Document Evaluated: CR 1-91-04-022 R0 Log Number: 92-0087

USAR Sections 2.1, 2.2 and 6.4 were revised to document the analysis which evaluated the possibility of exposure of main control room personnel to hazardous chemicals in the event of a release of anhydrous ammonia from the Shields Soil Service Facility located in the vicinity of Clinton Power Station. No equipment, systems or parameters are affected by this change.

RADWASTE EVAPORATOR BODY FOAM PROBE REMOVAL

Document Evaluated: CR 1-91-07-016 R0 Log Number: 91-0082

This change removed three evaporator vapor body foam probes and replaced them with blank flanges. These probes and the associated electronics had previously been retired in place. An evaluation determined that the probes were deteriorating and damage could occur to system components if they were not removed. The probes did not perform any control function. The replacement blank flanges are equivalent to the flanges installed with the probes and will maintain the integrity of the tanks.

TORQUE SWITCH SETPOINTS OF MOTOR OPERATED VALVES 1FP051, 1FP053 AND 1FP054

Document Evaluated: CR 1-92-03-036 R0 Log Number: 92-0072

The purpose of this evaluation is to document the acceptance of the existing motor operators and torque switch setpoints for fire protection system containment isolation valves, 1FP051, 1FP053, and 1FP054 until completion of the NRC Generic Letter 89-10 evaluation. Design values for stroke times and flow rates will not be altered as a result of this condition. Existing technical specification requirements are still being met and the valves will function as intended to meet the licensing basis.

LOST PARTS FROM FUEL HANDLING GRAPPLE INTO THE REACTOR VESSEL

Document Evaluated: CR 1-92-03-064 R0 Log Number: 92-0065

During the third refueling outage, stainless steel fragments of an electrical connector were accidently dropped into the reactor vessel during refueling activities. The fragments were not recovered and are assumed lost in the reactor vessel. The analysis report shows no potential interference with control rod operation or with the core flow system. There is a very remote chance that the fragment will cause fuel clad fretting failure after prolonged operation. This remote chance is from the fragments entering the bundle from either the upper or the lower tie plates. The presence of the stainless steel fragments in the core is not a concern since stainless steel is approved for use in the reactor, and therefore, is not a concern for potential adverse chemical reactions.

DISCREPANCY BETWEEN THE USAR AND THE DESIGN CRITERIA FOR THE FUEL POOL COOLING (FC) HEAT EXCHANGERS

Document Evaluated: CR 1-92-03-076 R0 Log Number: 92-0076

The plant configuration meets the design criteria requirements of having component cooling water system (CC) flow on the shell side and fuel pool cooling and cleanup system (FC) flow on the tube side of the heat exchanger. USAR Sections 9.1.3.2 and 9.1.3.1.2 were changed to reflect the correct configuration of the FC system interface with the CC system. The USAR was corrected to show that component cooling water flows on the shell side of the FC heat exchangers, and FC water flows through the tube side. The plant configuration meets all the design criteria requirements.

LOST PARTS FROM QUARTZ LAMP LENS IN THE REACTOR PRESSURE VESSEL (RPV)

Document Evaluated: CR 1-92-05-007 R0 Log Number: 92-0075

During the reassembly of the RPV, glass particles from a quartz light lens were found in the reactor cavity grating area and on the RPV flange. Several pieces, totaling approximately five square inches of the lens, were not recovered and are assumed lost in the RPV. The grating area around the RPV flange was flooded at the time of the break. The steam separators and the steam dryer were installed at the time of discovery which prevented the pieces from falling directly into the core region. This evaluation therefore assumes that all of the missing fragments fell into the annular downcomer region of the reactor. If the glass pieces had not fallen into the reactor, they would have entered the fuel pool cooling and cleanup (FC) system. The FC system is designed to remove suspended particles and impurities. Therefore, it is expected any glass fragments entering that system would be removed.

The conclusion is that reactor operation is safe with the glass fragments remaining in the vessel. The analysis performed shows that there is no concern for fuel bundle flow blockage since any glass fragments will be quickly reduced to small, sand-like grains by coolant flow. These grains will not inhibit flow through bundles, but will be dispersed in the reactor coolant and eventually collected by the reactor water cleanup system. Should any of the glass pieces migrate to the lower plenum and move to the core region, their area would be too small to cause fuel bundle flow blockage. Due to the small size and brittleness of the glass fragments, it is judged that they are incapable of causing interference with control rod operation. The presence of these glass fragments in the reactor is not a chemical hazard to the fuel or plant operation because the glass will break up into small grains, and the oxide composition of these grains will remain chemically inert at reactor coolant temperature. Therefore, there is no potential for adverse chemical reaction.

INSTALL BARRIER TO PROTECT DRYWELL PERSONNEL FROM RADIATION DUE TO DROPPED FUEL BUNDLE

Document Evaluated: ECN 9730 Log Number: 92-0026

This change provides a temporary barrier to be installed in the reactor cavity during refueling outages. The barrier consists of several platforms of metal grating which are mounted to the reactor head insulation bolts. The barrier system is named removable guaranteed accidental refueling drop shielding or (REMGARDS USAR Section 9.2 is being revised to reflect the REMGARDS installation. The design and installation of the REMGARDS modules have been analyzed for a seismic event and for the dynamic impact of a dropped fuel bundle. During normal operations, the REMGARDS modules will be stored in the steam separator pool. Storage of the modules in the pool was also analyzed for a seismic event. Storage of the modules in the separator pool will not decrease the amount of water available for an upper pool dump since the modules will be stored below the level of the upper pool dump lines.

EMERGENCY CLASSIFICATION PROCEDURE

Document Evaluated: EPIP EC-02, R3, ACN 4/3 Log Number: 92-0070

This procedure (EPIP EC-02) was revised to enhance the recognition and classification of emergency conditions and also, to make emergency action levels consistent with other approved CPS documents. This change does not involve the design or operation of plant equipment or the operations training of personnel required to operate plant equipment. This procedure only provides guidelines for classifying an accident. The revision to this procedure meets the requirements of 10CFR50.54q and 10CFR50 Appendix E.

EMERGENCY OPERATING PROCEDURE REVISIONS

Document Evaluated: Emergency Operating Procedures (EOPs) Log Number: 92-0059

In response to an NRC Region III Notice of Violation and open items in NRC Inspection Report 50-461/91006, several changes were incorporated in the Clinton EOPs and support procedures. These changes corrected deficiencies identified in the inspection report, and were consistent with Revision 4 to the Emergency Procedure Guidelines.

RELOCATE EMERGENCY RESPONSE HEADQUARTERS SUPPORT CENTER

Document Evaluated: EPIP HQ-01, R3, ACN 4/1 Log Number: 91-0091

IP has relocated the emergency response Headquarters Support Center to a new location at the IP Plaza in downtown Decatur. This is a change to the CPS Emergency Plan which is section 13.3 of the USAR. This change has no impact on plant operations and continues to meet the requirements of 10CFR50.54(q) criteria.

CHANGE AUTHORIZED REPRESENTATIVE FOR CERTIFYING THE ACTIVATION OF A LICENSED OPERATOR FROM VICE PRESIDENT TO DIRECTOR - PLANT OPERATIONS

Document Evaluated: NTD 2.13, R3, ACN 4/2 Log Number: 91-0079

10CFR55.53(f) requires that an authorized representative certify that the qualifications and status of a licensed individual, who has not maintained an active status, are current and valid prior to resuming license functions. Previously, USAR Section 13.2.2.1.3.A required the Vice President to perform this function. This change requires the Director-Plant Operations to approve the certification. This change does not impact 10CFR55.59 requalification.

SUPPLEMENTAL RELOAD LICENSING REPORT FOR RELOAD 3, CYCLE 4

Document Evaluated: Report 23A7144 Log Number: 92-0064

General Electric report 23A7144 documents the results of reload licensing analyses for the next (fourth) operating cycle. Included in this document are the plant conditions assumed in the analyses; fuel bundle types in the core and their proposed locations; applicable margin improvement and operating flexibility options; and core and transient analysis results. In a proprietary supplement, are maximum, average planar, and linear heat generation rate (MAPLHGR) values as a function of average planar exposure for each unique axial region (lattice) of the new fuel bundles to be irradiated for the first time in the fourth operating cycle.

184 irradiated fuel bundles were replaced by bundles of a new type, GE8B. Field alteration NBF010 documents the design changes and associated engineering reviews for using the GE8B fuel at CPS. The new bundles are standard, pre-pressurized, barrier-clad, S-lattice fuel bundles which will be described in a future update to GE document NEDE-31152P, the fuel assembly description document that is referenced by the GE Standard Application for Reactor Fuel (GESTAR).

The GE8B fuel bundle design retains many of the features of the previously-loaded GE7B fuel. Several changes have been made to accommodate higher fuel burnup, including: increased helium pre-pressurization; reduced clad-to-pellet diametral gap with an increased pellet diameter; increased fuel density; and increased fission gas plenum volume above shortened active fuel columns in gadolinia-bearing rods. The first three changes will improve thermal conductivity in order to reduce fuel temperature and thereby reduce the rate of fission gas release from the fuel. The increased fission gas plenum volume will accommodate the lower thermal conductivity of gadolinia-bearing fuel pellets. Because of the potential for higher bundle-average enrichments and an associated more-negative void coefficient of reactivity, GE8B incorporates a high-flow upper tie-plate to reduce pressure drop in the two-phase flow region of the bundle, thereby maintaining an adequate thermal-hydraulic stability margin. In order to improve the axial power distribution and cold shutdown reactivity margin, gadolinia is axially varied in the bundle. Axially-varying gadolinia was previously used to improve the power distribution and reduce the inventory of shallow control blades.

The technical information in report 23A7144 has been incorporated into the Clinton USAR by reference. In addition, the CPS Emergency Plan, which is referenced in Appendix D and section 13.3 of the USAR, is affected because the Emergency Classification Guide (Table 4-4a) is impacted by a change in maximum subcritical banked withdrawal position due to the loading of GE8B fuel. For the same reason, the Emergency Operating Procedures, which are referenced in section 13.5.2.1.3 of the USAR, are also impacted.

These changes are being made to meet cycle energy requirements at increa discharge burnups while operating in accordance with plant technical specifications. Technical specification changes have already been made which established the Core Operating Limits Report (COLR) and removed cycle-specific parameter limits from the technical specifications. In addition, a request to change the technical specification bases to reference the COLR for the MAPLHGR multiplier for single recirculation loop operation has been submitted to the NRC.

The core operating limits established for the fourth operating cycle meet the requirements of the technical specifications since the GE8B fuel design has been incorporated into GESTAR. Fuel storage requirements in Technical Specification 5.6 are not impacted for GE8B fuel since the basis of this section is reactivity rather than enrichment. Technical Specification 5.3 states that "each fuel rod shall have a nominal active fuel length of 150 inches." In GE8B fuel bundles, each fuel rod containing gadolinia has a reduced active fuel length in order to extend the length of the fission-gas plenum. This generic design change has been reviewed and approved by the NRC.

ADDENDUM TO THE SUPPLEMENTAL RELOAD LICENSING REPORT FOR RELOAD 3, CYCLE 4

Document Evaluated: Report 23A7144 Log Number: 92-0081

Safety evaluation 92-0064 documents the evaluation of reload licensing analyses for reload-3, cycle-4. Included in this evaluation are the plant conditions assumed in the analyses; identified fuel bundle types in the core and their proposed locations; applicable margin improvement and operating flexibility options; and core and transient analysis results.

The Core Operating Limits Report is the cycle-specific document which provides the operating limits for the following parameters: the average planar linear heat generation rate; the core flow and power dependent minimum critical power ratio (MCPR); and the linear heat generation rate. In accordance with Technical Specification 6.9.1.9, these limits are determined using previously reviewed and approved analytical methods, and are established so that all limits of the safety analyses are met.

IP was informed that the non-applicability of the misoriented bundle accident to CPS may no longer be valid. This is a generic problem for all S-lattice plants. Calculations were performed for the change in critical power ratio (delta-CPR) for the misoriented bundle accident for CPS. The resulting delta-CPR of 0.13 for GE8B fuel exceeds that for both the 100°F loss of feedwater heating event and the rod withdrawal error. Since these are the limiting events for cycle-4 operation, the MCPR operating limit may be impacted.

Proper orientation of fuel assemblies in the reactor core was verified during RF-3. Several different visual indications of correct orientation ensure that any misoriented assembly would be readily identifiable during core loading verification. While the generic issue is being discussed with the NRC, the Core Operating Limits Report was revised with a modified, flow-dependent MCPR limit curve for the GE8B fuel to accommodate this increased delta-CPR. Specifically, the portion of the curve corresponding to rated and near-rated core flow is being moved up to 1.20 from 1.18.

OFFSITE DOSE CALCULATION MANUAL, REVISION 8

Document Evaluated: ODCM, Rev. 8 Log Number: 91-0076

The Offsite Dose Calculation Manual (ODCM) provides the methodology to assure compliance with the radioactive effluent dose limitations stated in 10CFR20, 10CFR50 Appendix A, 10CFR50 Appendix I, and 40CFR190. The primary requirements for the CPS Radiological Environmental Monitoring Program (REMP) are also set forth in the ODCM. The REMP conforms to the guidance of 10CFR50 Appendix 1.1. The ODCM was revised to delete the general surveillance requirement in Section 1.3.1 which stated that the combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval. The revised surveillance interval will allow an extension not to exceed 25 percent of the specified surveillance interval. This change is consistent with the Technical Specification surveillance requirement performance requirements as amended in CPS Technical Specification Amendment No. 53, approved November 1, 1990.

The following changes were also made in Revision 8 to the ODCM: Paragraph 2.2.1 was deleted and the criteria describing representative sampling for batch releases of radioactively contaminated water from CPS were added to Table 2.3-1 notations. Figures 2.1-2, 2.5-1, 2.5-2, 2.5-3, and 3.3-1 were updated with editorial changes for ease of reading or more accurate descriptions. Section 2.5.2 was revised to provide clearer guidelines on the amount of radioactivity in liquid radwaste that may be stored in temporary liquid radwaste hold-up tanks. Table notation "g" for Table 3.4-1 was revised to clarify sampling requirements of charcoal and particulate on the station heating, ventilation and air conditioning and standby gas treatment system stacks. In Table 3.9.2-1, note "1" was added to clearly identify what is required to consider a particulate or iodine sampler operable. Thermoluminescent dosimeter sampling locations CL-95, CL-96 and CL-97 were added to Figure 5.0-4 and designated as control locations. Requirement 2.a of Section 7.2 was revised to require total container volume vice container volume be reported for each class of solid waste shipped offsite during the report period in the Semiannual Radioactive Effluent Release Report per Appendix B of Regulatory Guide 1.21.

ESTABLISH AN OUTAGE PERSONNEL STAGING AREA IN THE RADWASTE BUILDING AT THE 762 FOOT ELEVATION

Document Evaluated: Staging Area Log Number: 92-0020

The 762 foot elevation of the radwaste building was utilized as a personnel staging area during the third refueling outage (RF-3). This evaluation allowed the increase in combustible fire loading. USAR Appendix E - Fire Protection Evaluation Report, Section 3.7.4.3, was affected due to the increase in the fire load from office equipment and other appurtenances necessary for personnel occupancy. There are no safety-related cable trays or equipment located in this fire zone. The additional fire hazard is acceptable with additional administrative controls established per CPS Procedure 1893.01, "Fire Protection Impairment Reporting".

TELEPHONE SYSTEM UPGRADE

Document Evaluated: USAR 1.2.8.17, 9.5.2.2, and T9.4-32 Log Number: 92-0103

The main solid-state switchboard for the plant telephone system was moved from the Service Building to the Nuclear Support Building. The telephone systems are powered from a 12 kV loop outside the plant protected area instead of a non-1E motor control center (MCC). A 52 volt DC battery provides backup power instead of a Division 1 power supply. The USAR Sections 1.2.2.8.17, 9.5.2.2 and Table 9.4-32 were revised to reflect these changes and also include that IP owns and maintains the telephone systems instead of a local phone company.

PROCESS DIAGRAMS AND OTHER FIGURES

Document Evaluated: USAR 1.7.4 Log Number: 92-0068

The USAR was clarified to say that process diagrams and other figures shown in the USAR, while representative of typical plant design and operations, may not be current. Current design basis documents and drawings should be referred to when reviewing and evaluating the design. It further states that the annual updates to the diagrams and figures will be provided when permanent configuration changes are made. This change is an administrative change.

DOCUMENTATION OF POST-LOCA HYDROGEN GENERATION

Document Evaluated: USAR 1.8, Regulatory Guide 1.7 Log Number: 92-0058

USAR Section 1.8 was clarified to identify that specific analytical data pertaining to post-LOCA hydrogen generation in the containment and drywell are subject to change. The postulated post-LOCA hydrogen generation concentrations during the operational life of the plant are to be documented in design baseline calculations rather than in the USAR. The specific data points and values in the USAR which were utilized in, or were a result of, the hydrogen generation analysis will not be removed from the USAR. However, this information in the USAR is only representative of analytical methodology and results. CPS procedures and baseline documentation ensure the limitation of post-LOCA hydrogen concentrations inside the containment and drywell in accordance with Regulatory Guide 1.7.

USAR SECTION 1.8, REGULATORY GUIDE 1.33, Revision 2

Document Evaluated: USAR 1.8, Regulatory Guide 1.33 Log Number: 92-0042

USAR Section 1.8 was revised to take exception to the ANSI N18.7 requirement of biennial procedure review and identify programmatic controls that are equivalent to or better than the biennial review process. The change was made to reduce the procedure review workload when the only requirement for this review is the biennial requirement.

USAR SECTION 3.1.2.2.5

Document Evaluated: USAR 3.1.2.2.5 Log Number: 92-0093

The USAR Section 3.1 was revised to reflect that residual heat removal check valves 1E12-F050A/B and 1E12-F053A/B are not equipped with position indication in the main control room. These valves perform the function of a third isolation barrier and are therefore not required to have open or closed indication. This is a documentation change only. No physical changes are being made to the plant. No systems, equipment or parameters will be affected by this change.

TYPE "B" LEAK TESTING FOR RCIC HEAD SPRAY LINE PIPING FLANGED CONNECTION

Document Evaluated: USAR 6.2.6.2 Log Number: 91-0044

The reactor core isolation cooling (RCIC) system piping to the reactor head is provided with a flanged connection to facilitate reactor pressure vessel head removal. This change adds this connection to the USAR listing of items for which type "B" primary leakage testing is specified per 10CFR50, Appendix J. This is a corrective action required by Licensee Event Report (LER) 90-18. This testing is performed when RCIC is not required to be operable.

RESTRICT TESTING OF UNTESTED ISLANDS IN NUCLEAR SYSTEM PROTECTION SYSTEM TO THOSE AFFECTING FUNCTIONAL LOGIC

Document Evaluated: USAR 7.2.1.1.4.8 Log Number: 92-0045

The nuclear system protection system (NSPS) is provided with a self test system (STS), which automatically tests the NSPS logic continuously; however, there are some components of the NSPS logic circuits which are not tested by the STS. The untested portions are referred to as untested islands (UTIs). CPS committed to the NRC to perform manual testing of the UTIs. Manual testing involves removing the circuit cards from the NSPS panels. Removing the circuit cards has resulted in greater impact on equipment and operations than those failures that the tests are designed to avert. There have been three Licensee Event Reports (LERs 88-009, 89-012, and 90-015) caused by removing or inserting circuit cards in the NSPS panels since testing began in 1987. A failure modes and effects analysis (FMEA) was performed which indicated that continued testing of several UTI types is not necessary. For these UTIs, it has been determined that those failure modes which are not detectable by STS either do not inhibit the safety function of the NSPS logic or are detectable during the performance of system or instrumentation surveillance tests currently required by the CPS Technical Specifications. The types of UTIs no longer subject to manual testing are: self-test coupling capacitors, logic seal-in or latching circuits (20 millisecond delay and 250 millisecond delay), and power-on initialization circuits.

Because the above UTIs are no longer being manually tested, USAR Section 7.2.1.1.4.8 is being revised to state that special manual tests are only required to detect those failures which could prevent the NSPS from performing its intended safety functions. Since UTI testing is not addressed by the technical specifications, this change does not alter any technical specification. Testing needed to meet the requirements of logic system functional tests will continue to be performed as required. Some of the failure modes of these UTIs could result in spurious operation of safety equipment. However, elimination of the requirement to periodically remove the associated circuit cards has the potential to improve the performance of the associated equipment by eliminating the potential for spurious trips while removing or inserting the cards in the NSPS panel.

CLARIFY THE RATED VOLTAGE FOR MOTORS OF MOVS

Document Evaluated: USAR 8.3-10 Log Number: 92-0061

In 1991, an NRC inspection evaluated the methodology at CPS for evaluating the capability of motor operated valves (MOVs) to achieve the required thrust. The results of this inspection are documented in NRC Report 50-461/91019. The inspection evaluated the actions being taken in response to Generic Letter 89-10 (Safety-Related Motor Operated Valve Testing and Surveillance). The USAR was clarified to state that continuous duty motors are capable of producing full load torque at 75% of rated voltage, and Class 1E motors for motor-operated valves are designed to open or close the valve against specified differential pressure, without damage, at 90% of rated voltage.

REVISE USAR TO RECOGNIZE THE POTENTIAL FOR NEUTRON ACTIVATION OF CORROSION INHIBITOR CHEMICALS IN THE PLANT CHILLED WATER AND DRYWELL CHILLED WATER SYSTEMS

Document Evaluated: USAR 9.2.8 Log Number: 91-0105

The USAR was revised to acknowledge the presence of low radioactivity levels (10E-7 to 10E-6 microCi/ml.) within the drywell ventilation (VP) and plant chilled water (WO) systems and to institute requirements for the review of any changes to the amount or type of corrosion inhibitor added to these systems. The radioactivity levels resulted from neutron activation of the corrosion inhibitor which was added to the WO and VP systems. The USAR is being changed to acknowledge that the corrosion inhibitor in the WO and VP has been neutron activated inside the drywell. The radiation levels resulting from the activation do not pose a concern to individuals working on these systems or a contamination problem in the case of spills or leaks. The proposed change to the USAR does not introduce an unreviewed safety question, increase the probability or consequences of an accident or malfunction, or introduce a new type of accident or malfunction.

DELETE OPERATIONAL CRITERIA FOR THE SERVICE BUILDING VENTILATION SYSTEM FROM THE USAR

Document Evaluated: USAR 9.4.12.1.2.6 Log Number: 91-0098

The service building ventilation system descrition in USAR Section 9.4.12 was revised to remove the statement "The system maintains the offices at 75°F ±2°F and 45 ±5% relative humidity (RH) in the summer and 35 ±5% RH in winter". This change allows the personnel working in the office areas of the service building to adjust the room temperatures for their own personal comfort.

DIVISION III DIESEL GENERATOR AIR COMPRESSOR TAGOUT

Document Evaluated: USAR 9.5.6.2 Log Number: 92-0091

The diesel-driven air compressor (1DG03CA) has been removed from service for extended periods of time because of check valve problems. At the time of the safety evaluation, the air receiver inlet check valve (1DG172) to the air receiver associated with the diesel would not open. Normally, when either the motor-driven air compressor (1DG03CB) or the diesel-driven compressor starts, both air receivers (1DG06TA,B) are pressurized equally. In the present configuration, the air receivers require manual equalization to maintain the pressure in both receivers. This is done by momentarily opening (for approximately 30 seconds) the air receiver cross connect valve (1DG631) per CPS Procedure 3506.01. The equalization process ensures that both air receivers maintain their safety design basis capability of five successive starts without recharging the air receivers.

The USAR states that the Division III diesel generator has two air supply subsystems. Each subsystem consists of one air receiver connected to one starting motor train, and each air receiver is charged by an individual air compressor.

USAR Section 9.5.6.2 will be changed to clarify that the compressors are connected to a common air dryer and the air receivers can be charged by either or both air compressors. Also, the equalization valve may be used to manually equalize the pressure between the air receivers.

INCREASE VOLUME OF RESIN AND ALLOW RATIOS OTHER THAN 1:1 FOR CATION/ANION USED IN LIQUID RADWASTE DEMINERALIZERS

Document Evaluated: USAR 11.2.1 2 Log Number: 92-0001

USAR Section 11.2.1.2 was revised to state that a total of 90 cubic feet of anion and cation resin will be used in each liquid radwaste treatment system demineralizer rather than stating a specific ratio. CPS does not regenerate resin, therefore, the change will increase the ratio of anion to cation resin resulting in a decrease in the ionic concentration in the waste stream. This change will not increase the concentration of radioactive source terms above that used in the analysis of the ruptured concentrated waste tank because the source terms for each radwaste waste stream are traced to ensure they remain below those assumed in the accident analysis.

REVISION TO RADWASTE CONTAINER DROP ACCIDENT ANALYSIS

Document Evaluated: USAR 11.4.1.6 Log Number: 92-0056

The CPS mobile radwaste solidification system was revised to allow waste to be dewatered or solidified in large volume waste containers. Operationally, this results in fewer lifts made with the radwaste crane, but the loads are heavier. Some loads must be handled with the turbine building crane instead of the radwaste crane. Section 11.4 of the USAR was revised to reflect this change. An analysis of a ruptured liner was performed. It demonstrated that the limits of 10CFR20 are not exceeded. Also, the radionuclide concentrations which may be anticipated to occur throughout plant operation including fuel failure were used in the calculation in order to determine the maximum concentrations which approach the 10CFR20 limits. These concentrations are periodically monitored by CPS 10CFR61 compliance program. These radionuclide concentrations are also being incorporated for reference into USAR Table 12.2-12.

LIQUID RADWASTE TANK FAILURE ANALYSIS

Document Evaluated: USAR 15.7.3 Log Number: 92-0035

USAR Section 15.7.3 provides an analysis of a postulated radioactive release due to a liquid radwaste tank failure. The original calculation was revised to base the analysis on the rupture of the concentrated waste tank instead of the phase separator tank. The concentrated waste tank was chosen for analysis because it has the highest average radioactivity that could become airborne. Although the radionuclide inventory for the phase separator tank is greater than that for the concentrated waste tank, the radionuclides in the phase separator are ionically bound to the phase separator resins, and therefore, less likely to go airborne than those in the concentrated waste tank. The methodology used to perform the calculation for whole body and inhalation doses at the exclusion area boundary and in low population zones reflects a more realistic, but still conservative calculation. The methodology used in these calculations is acceptable for dose calculations.

REVISED CRITERIA FOR FIRE-RATED SEALS IN ELECTRICAL CONDUIT

Document Evaluated: USAR APP E 3.1.2.2.10, 9.5.1.1.d Log Number: 91-0101

This change revised CPS's position, regarding periodic inspection and installation of internal fire-rated conduit seals, which states that all penetrations through fire barriers such as mechanical piping, electrical conduit and cable trays will be sealed with fire-rated seals. USAR Section 9.5.1.1.d was also revised to include reference to the "Conduit Fire Protection Research Program", Final Report, which provides the technical justification for revising internal conduit fire seal installation and inspection practices.

The report was accepted by the NRC in a letter, dated October 23, 1989 titled, "Review of Draft Safety Evaluation of Conduit Fire Seal Topical Report for Proprietary Contract," and was substantiated with a safety evaluation and technical evaluation report. The NRC staff evaluated the research program in the safety evaluation and technical evaluation report and concluded there is justification for the revised conduit sealing criteria. The report provides guidelines regarding sealing of conduits penetrating fire barriers. The guidelines are consistent with the proposed change.

Other justification for acceptability of the change includes: past Quality Assurance (QA) and Quality Control (QC) inspections of internal conduit seal material installation during the construction phase, and continued QA material installation inspections during the operational phase; maintenance work control process; engineering (configuration control) design change and associated design review process; limited accessibility and small cross-sectional area of material exposed; seal material qualification for the plant life; and limited possibility for propagation of fire from one fire area to another.

CORRECT DISCREPANCIES IN RELIEF VALVE DISCHARGE LOCATIONS

Document Evaluated: USAR F3.6-1, F6.2-133, F9.1-4 Log Number: 91-0071

USAR Figures 3.6-1 and 9.1-4 were revised to correct the location for the discharge of relief valves 1FC085A and 1FC08FB, which discharge to the suppression pool via the suppression pool makeup (SM) system steam separator storage pool dump lines. Also, USAR Figures 3.6-1 and 6.2-133 were revised to correct the discrepancy for the location of relief valve 1SM003A. As the actual relief valve discharge locations are adequate there is no increase in the probability or consequences of any malfunction or accident analyzed in the USAR.

REACTOR WATER CLEANUP SYSTEM PROCESS DIAGRAMS

Document Evaluated: USAR F5.4-17, S1 & 2 Log Number: 92-0067

USAR Figure 5.4-17 was revised to reflect the current design maximum flow rate for the reactor water cleanup system of 450,000 pounds per hour (907 GPM at 110°F shell inlet temperature) is acceptable during startup and shutdown conditions. The changes were the result of concerns that the operation of the reactor water cleanup system in the recirculation mode would provide higher reactor water temperature to the feedwater system piping, which has been analyzed for a lower operating temperature. This recirculation mode of operation precludes thermal stratification in the bottom of the vessel when the reactor recirculation pumps are not operating. Administrative controls were added to the operating procedure to ensure that the feedwater piping is not heated up above 435°F.

AS-BUILT BREATHING AIR SYSTEM PIPING AND INSTRUMENTATION DIAGRAM (P&ID)

Document Evaluated: USAR F9.3-3 Log Number: 90-0071, R1

This USAR change corrected Figure 9.3-3 to as-built conditions for the breathing air (RA) system. The changes consisted of correcting a pipe diameter and a reducer location. This change did not affect the system components and did not increase the consequences or probability of occurrence of RA failures evaluated in the USAR.

CORRECT SEISMIC CLASSIFICATION OF THE REACTOR WATER CLEANUP PUMPS

Document Evaluated: USAR T3.2-1 Log Number: 91-0114

USAR Table 3.2-1 was revised to change the seismic category of the reactor water cleanup (RT) pumps from seismic category I to N/A. Although the RT pumps are not seismic category I, the nuclear steam supply system design criteria and ASME Section III require the consideration of seismic loading in the design of the pump.

BATTERY LOAD REQUIREMENT CALCULATIONS

Document Evaluated: USAR T8.3-8, 9, 10, & 11 Log Number: 92-0090

USAR Tables 8.3-8, 8.3-9, 8.3-10 and 8.3-11 for Division 1, 2, 3, and 4 were changed to reflect a change to the 125 volt battery load amperage requirements. These values were adjusted based on a design review of battery load calculations. The calculation reviews were made to provide greater detail and documentation support for the analyzed battery loads. The revised loads remain within the design capability of the batteries.

REVISE DIVISION III BATTERY LOADS

Document Evaluated: USAR T8.3-10 Log Number: 91-0073

The values in USAR Table 8.3-10, for the Divison III 12 volt battery load requirements, are being revised as a result of a general design review of the battery loads. The major change was the incorporation of the starting current value for the Division III diesel-engine- powered air compressor starter motor. The revised loads are within the design capacity of the battery. A change to Technical Specification 4.8.2.1.d.2.C is being processed but is not required prior to implementation of this USAR change since the changes are still in compliance with the Technical Specifications.