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Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
1990 Report of Facility Changes, Tests and Experiments

Gentlemen:

Enclosed is the 1990 Report of Facility Changes, Tests and Experiments for Waterford 3 which is submitted pursuant to 10CFR50.59. This annual report covers the period from June 19, 1989 through June 18, 1990.

If you have any questions regarding this report, please contact D.A. Rothrock at (504) 739-6693.

Very truly yours,

R.F. Burski

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Entergy Operations Inc.
Waterford 3 SES
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REPORT OF FACILITY CHANGES, TESTS AND EXPERIMENTS

FOR 1990 PER 10CFR50.59

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Waterford 3 SES

1990 report of Facility Changes, Tests and Experiments

SUMMARY

This report provides the Waterford 3 Facility Changes made pursuant to 10CFR50.59(a)(1). The report covers the period from June 19, 1989 through June 18, 1990. None of the items in this report represent an unreviewed safety question.

The report identifies 105 Facility Changes (17 changes under the Design Change Package program, 27 under procedure changes and 61 special safety evaluations)

DESIGN CHANGE PACKAGES (DCPs)

Design Change Package, DCP-3027

DCP-3027, Safety Injection Tanks Narrow Range Level Indication (Revision 0)

Description of Change

The redundant narrow range level transmitters on each of the four safety injection tanks (SIT) share the same reference leg. DCP-3027 will separate each transmitter from the common reference leg and provide redundant and separate reference legs. One of the narrow range level transmitters will be retubed to the reference leg for the wide range level transmitter.

Reason for Change

The design change is to prevent recurrence of a TS violation caused by inadequate design. The TS violation resulted because both narrow range level transmitters for each SIT are connected to the same reference leg. The reference legs on the SITs are wet leg (filled). When a leak in the reference leg on one of the SITs occurred, the level appeared to increase in the tank. The violation occurred because the tank level was drained below TS limits due to the inaccurate reading of the level indication.

Safety Evaluation

This design change involves separating the narrow range level transmitters from sharing the same reference legs. This will allow two redundant narrow range level indications for each SIT. The SIT level transmitters are used to assure that the tank levels remain within the TS limits. The redundancy that this design change will provide will allow operations to notice any difference in level indication before a problem could develop. The use of the transmitters are not being modified. The transmitters will still be used to monitor the level of the SITs. This modification will allow a more reliable indication of the SIT levels. The level indications are not required for a safe shutdown. The level indications are use to ensure that the levels remain within TS limits to mitigate the consequences of a design basis event. This design change will improve the ability to monitor the level in each of the SITs and thereby reduce the possibility of a malfunction. Since the reference legs of the narrow range level transmitters will be separated, a problem involving one of the level indicators, transmitters or reference legs will be easily noticed. This design change is to reduce the possibility of a malfunction of safety-related equipment. The redundancy that this provides will reduce the possibility of an error due to equipment malfunction. This change will not involve any change to the TSs; however, the independent narrow range indications for each SIT will ensure that the TSs are maintained.

Design Change Package, DCP-3059

DCP-3059, Condenser Vacuum Pump Effluent Radiation Monitors (Revision 0)

Description of Change

This change revises the Condenser Off-gas Radiation Monitoring system by deleting system redundancies and providing a permanent sample cooling and conditioning skid. Radiation monitor PRM-IRE-0001 is being deleted. The function of this monitor will be relocated to monitor PRM-IRE-0002. Sample cooling will be provided by a self contained package chiller.

Reason for Change

This is a system improvement that will improve reliability and replace the need for potable water currently being used as a cooling medium.

Safety Evaluation

The design basis Steam Generator Tube Rupture (SGTR) bounds the possible accidental release of radiation through the turbine building vent. Other smaller primary to secondary leaks result in far less radioactive gas inventory in the steam system. SGTR may last fifteen minutes before a low pressurizer pressure causes an automatic reactor trip and subsequent main steam line isolation upstream of the condenser. A less serious SGTR may not result in a reactor trip, and operator action to isolate the leaking steam generator may occur thirty minutes after the SGTR. The wide range monitor modification takes all of the functions of the narrow range monitor without sacrificing sensitivity or performance. No greater or lesser amount of radioactive gases exit the turbine building vent after a SGTR. The wide range monitor change does not eliminate a backup narrow range monitor control function because one never existed, nor was there a requirement for a backup.

The new design retains the previous conditions where monitor power is lost in a LOOP and diversion valves fail as is. Although the feature does exist to manually connect the monitor to the EDGs in a LOOP, the condenser vacuum pumps will not operate; therefore, monitoring is not required. Under the worst conditions, SGTR and LOOP scenario, if the monitor becomes manually loaded on to an emergency power supply, the non-safety-related solenoid sample isolation valves would remain closed and the motorized diversion valves would not automatically reposition. This occurrence would leave the monitor in a useless mode. In the worst case SGTR scenario, TS controls on secondary chemistry provides more general public dose protection than the off-gas diversion system. The bulk of the effluent leaves through other pathways not measured by this monitor.

The order of events during an accident prohibits any but the smallest quantities of radioactive gas to enter the off-gas system because of main steam isolation features. During the early phases of a SGTR accident, the reactor control system would attempt to replace the lost Reactor Coolant System (RCS) inventory. The bulk of the plant systems, including power production, would continue, so normal plant conditions and responses to radioactivity in the off-gas is expected. A concurrent LOOP causes bypass valve failure sending steam to the astmosphere terminating radioactive gas flow to the condensor. The new design does not rearrange any safety-related system or delete functions which could create a new accident.

The wide range monitor skid only changes to adopt the narrow range monitor output function (diversion valve control). A narrow range monitor is part of the wide range monitor skid. Further, the new sample conditioning system and revised sample line routing increases the reliability of the wide range monitor by reducing the probability of moisture intrusion into the monitor.

A wide range off-gas monitor failure results in consequences no different than a power failure to the non-safety-related, "fail - as is" diversion valves. A monitor failure causes a loss of automatic control (diversion) and effluent measurements no different than the current design.

This design change does not add, delete, or revise any safety-related components. All equipment modified by this change is located in the turbine building. Therefore, the possibility of a malfunction of equipment important to safety is not created by the implementation of this change.

The margin of safety comes from adherence to regulations requiring condenser off-gas monitoring. Because radiation monitor PRM-IRE-0002 can perform all functions presently performed by radiation monitor PRE-IRE-0001, the present regulatory requirements are not affected by this change. The TSs list the important radiation monitoring functions in the plant. The only listing relevant to the Main Condenser Evacuation System (MCES) falls under the Effluent Accident Monitor heading. The wide range monitor PRM-IRE-0002 can accommodate that without PRM-IRE-0001. The TS lists lower limits of detection (LLD) for MCES monitoring. Monitor PRM-IRE-002 contains a low range detector which will monitor the LLD.

Design Change Package, DCP-3080 (Rev 0)

DCP-3080, Diverse Reactor Trip System (DRTS) / Diverse Emergency Feedwater Actuation System (DEFAS) to comply with 10CFR50.62 (ATWS) Requirements and Pressurizer Pressure Signals to QSPDS for Regulatory Guide 1.97 Compliance

Description of Change

This DCP covers the design details for DRTS and DEFAS to comply with 10CFR50.62 requirements (ATWS rule). The package also covers design changes for wide range pressurizer pressure signals to QSPDS for Regulatory Guide 1.97 compliance.

Reason for Change

These changes are being made to comply with 10CFR50.62 (ATWS Rule) and Regulatory Guide 1.97.

Safety Evaluation

The DRTS and DEFAS will not increase the probability of accidents previously evaluated in the FSAR in that these systems are secondary to existing systems and do not contribute to the initiating events for the accidents analyzed in the FSAR. Class 1E isolation devices are installed. A loss of power to the DRTS or the DEFAS or the application of a single, active failure will not introduce a plant transient.

The implementation of DRTS and DEFAS will not increase the consequences of an accident previously evaluated in the FSAR. The DRTS and DEFAS are designed so that any single failure will not cause these systems to actuate. Assuming the worse case scenario of a main steam line break (MSLB) and a single active failure, the systems will not inadvertently actuate and will not increase the consequences previously analyzed. The DRTS and DEFAS setpoints are set beyond the existing settings of the Plant Protection System. The system design incorporates Class 1E isolation devices, single failure criterion and enhanced logic to assure that DRTS and DEFAS are initiated upon failure of the PPS. The system precludes inadvertent actuation on open circuit failures since the initiation relays require energization to actuate. A two-out-of-two logic is required to actuate the DEFAS; thus, precluding inadvertent actuation on a single active failure. The design features of the DRTS and DEFAS coupled with augmented quality controls in the procurement, design, installation and operation of the system provide adequate assurances that a malfunction different from that already analyzed in the FSAR is not credible.

The DRTS and DEFAS systems will not create different accidents than those previously evaluated in the FSAR. These systems are secondary alternate systems for tripping the reactor and turbine with the ability to actuate emergency feedwater in a diverse manner. Therefore the effects of implementing these systems do not change the evaluations as defined for the safety-related primary systems. The new DRTS and DEFAS equipment has been evaluated for electrical, physical, and functional interactions with previously evaluated safety equipment. The design of the new systems use class 1E isolation devices and conservative actuation setpoints to assure that the installation of the new equipment will not increase the probability of previously evaluated safety equipment failure. In addition, the DRTS and DEFAS uses diverse design features to preclude the potential for common mode failures in both the DRTS and DEFAS and the safety grade PPS.

As an assurance that the automatic initiation of the DEFAS will not take place except under conditions indicative of an ATWS, the DEFAS is interlocked with the DRTS such that initiation of the DEFAS is permitted only upon actuation of DRTS. The DRTS and DEFAS will use high quality equipment procured, designed and installed in accordance with the quality assurance requirements of Generic Letter 85-06. Testing can be accomplished during power operations to assure that DRTS and DEFAS logic will operate properly; however, this does not verify the sensor or final actuation device. This will be done on the same frequency as the PPS surveillance test (quarterly). The systems will be tested from sensor to final actuation device each refueling outage. This test program coupled with control room indication of DRTS and DEFAS circuitry trouble assure that impending failures of DRTS and DEFAS equipment will be detected in a timely manner.

The DRTS and DEFAS are designed to actuate for mitigatory purposes for anticipated operational occurrences (AOO) scenarios only after failure of the existing protection systems. The DRTS and DEFAS are sufficiently independent and diverse to assure that equipment and margins covered by the plant's TSs are not impaired by the implementation of these systems. In addition, the implementation of DRTS and DEFAS will not impact the plant's accident analyses which form the basis of the TSs.

Design Change Package, DCP-3088

DCP-3088, Transfer System Proximity Switches (Revision 0)

Description of Change

The fuel transfer machine limit switches and cable assemblies are being replaced. The switches will be replaced by maintenance free proximity switches. This change will not change the operation of the transfer machine.

Reason for Change

This design change package replaces eight underwater limit switches and cable assemblies in the fuel transfer machine due to repeated malfunctions.

Safety Evaluation

The affected equipment was not previously evaluated in the FSAR. This modification replaces existing mechanical limit switches with proximity limit switches and does not increase the probability of an accident. No credit is taken for components or subsystems of the fuel handling equipment to mitigate the consequences of the postulated fuel handling accident. The described change will not change the operation or possible failure mode of the transfer machine.

The transfer machine is a non-safety, non-seismic system that does not contribute to the probability of a malfunction of equipment important to safety. The change will not change the operation of the transfer machine and will create no new possibilities of a malfunction or contribute to the consequences of a malfunction of equipment important to safety. The transfer machine does not contribute to any margin of safety as defined in the bases of the TSs.

Design Change Package, DCP-3097

DCP-3097, Fuel Alignment Plate (Revision 0)

Description of Change

The design modification involves two changes to the Fuel Alignment Plate (FAP) thimble flow path. The first is made to the thimble throat area and consists of enlarging sections of the throat using electric discharge machining (EDM) in a "splined" pattern. The "lands" are designed to preserve the function of thimble centering which facilitates thimble re-insertion into the fuel assembly after refueling. The removal of material provides a larger flow area in the throat and a throat-to-tube flow area ratio which is closer to unity, hence reducing differential flow velocities and vortex formation.

The second FAP design change consists of the insertion of a flow restricting "plug" into the alignment plate thimble well. This plug reduces flow in the thimble throat such that vortex formation is terminated. Each thimble well is sized and an appropriate dimension plug selected for fit. The plug is pressed into position with approximately 8,000 pounds force. The self-locking design chevrons prevent plug removal. (Plugs can be removed using EDM methods).

Reason for Change

Local reactor coolant flow patterns between the in-core instrument (ICI) thimble and the instrument tube at the FAP induce thimble vibration. The intent of this modification is to alleviate the flow induced vibration by enlarging the flow area at the entrance to the instrument tube and installing a flow restrictor plug at the thimble well in the bottom of the fuel alignment plate. These changes reduce the flow and alter the adverse flow patterns thereby eliminating the source of unacceptable thimble vibration.

Safety Evaluation

The probability or consequences of an accident previously evaluated in the FSAR will not be increased because the modifications do not affect the fit or function of the FAP or ICI thimbles. The thimble guide tube is a separate piece, slipfit into the fuel alignment plate and held in position by "straps" attached to the control element assembly (CEA) shroud. Therefore, the enlargement of the thimble guidetube "throat" does not affect the structural integrity of the FAP. The possibility of the plugs shattering is very small because the plugs are press fit to the ICI guide tube and the plug materials are made of ASME SA-479 Type 304 stainless steel bar stock. The upper end fitting from beneath, and the instrument guide tube from above will keep the new plug from migrating out of the FAP. Damage to fuel or other RCS components because of large parts from the plug moving through the RCS is not possible. Small parts of the plug will not form because of installation procedures and material selection. Further, stresses in the FAP caused by any accident is insufficient to relieve the forces holding the plugs in place.

The probability or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because, as noted earlier, the modifications do not affect the fit or function of the FAP or ICI thimble. Analysis performed by Combustion Engineering shows that the transient response would be adequate with a slight redistribution in flow. The additional weight of 56 plugs to the FAP is 200 pounds and will be distributed uniformly within the FAP. The minimal additional weight is well within the tolerance of the Upper Guide Structure (UGS) weight allowed in the seismic calculation. This modification will minimize the flow velocity at the thimble throat locations, thus stopping the vibration and the resulting wear. The result is elimination of a possible malfunction of equipment important to safety.

The change does not degrade the performance of a safety system in the FSAR. There are no changes to protective boundaries and no impact on the TSs. There is no increase in the probability of exceeding a safety limit.

Design Change Package, DCP-3101

DCP-3101, Reactor Coolant Pumps (RCPs) Platforms and Ladders (Revision 0)

Description of Change

This design change installs stairs from the top of the "D" ring to the upper portions of RCPs 2A and 1B, platforms for RCP 1B and RCP 2A, a catwalk over a whip restraint, a platform over the heating, ventilation, air conditioning (HVAC) duct located in the east "D" ring which interconnects the access platform for RCP 2A with the catwalk for RCP 2B and relocates the fire protection sprinkler lines for RCP 1B and 2A.

Reason for Change

The platform and stairs have been designed to accommodate the removal of the RCPs, motors and the motor enclosures.

Safety Evaluation

The stairs and platforms for RCPs 2A and 1B and the platform over the HVAC duct do not perform any safety-related function. The supporting steel for these structures has been designed as Seismic Category I. The HVAC duct located in the East "D" ring is classified as non-safety seismic. The support steel for the duct is classified as Seismic Category I. The platform over the HVAC duct will also be attached to the same support members for the duct. Since the new platform is designed Seismic Category I, it will not degrade the design function of the duct supports nor impact the functional operation of the HVAC system. The fire protection sprinkler lines which require relocation are classified as non-safety seismic. To satisfy the existing design criteria, the new pipe is identical to the old, and the new supports are seismically designed and qualified in accordance with the original specification. The additional amounts of structural steel and process piping in the reactor containment building (CB) will be included in the tracking of the containment Net Free Volume.

The addition of the platforms and stairs provide no safety function and meet seismic design requirements. Thus, the addition will not increase the probability or consequences of an accident previously evaluated in the FSAR. The probability or consequences of a malfunction are unaffected for the same reasons. There is no equipment being added by this change that has a safety function, therefore the possibility of a malfunction of equipment important to safety different than already evaluated in the FSAR is not created and the margin of safety will not be reduced.

Design Change Package, DCP-3106

DCP-3106, HVAC Containment Radial Duct Modification For Polar Crane Lifting Clearance (Revision 2)

Description of Change

This change installs a removable section of HVAC ring header duct above the maintenance hatch and a laydown area above the number one steam generator framing. It also relocates lights, conduits, polar crane chock stops, and existing area radiation monitoring equipment. Two isolation dampers are provided at each end of the fixed ring header.

Reason for Change

The intent of the modification is to facilitate removal of the RCP motors.

Safety Evaluation

The installed equipment is non-safety, seismically restrained and thus will not increase the probability of an accident. The relocated safety-related equipment performs only a monitoring function which will not increase the accident probability. The changes made to the HVAC ring header have no impact on the functional operation of the system or the safe operation of the plant. The reduced section and dampers have been designed according to the FSAR's classification. The support framing has been designed as a Seismic Category 1 structure. Isolation dampers, used when the section of duct is removed during outages, shall be equipped with a lock open device to ensure the dampers are held open during normal operation.

The installed equipment is not required for accident mitigation. The relocated safety-related equipment maintains its ability to detect an accidental radiation release. Therefore, the consequences of an accident are not increased. The installation does not affect other equipment necessary for accident mitigation. Therefore, the consequences of evaluated accidents are not increased. A malfunction of the Area Radiation Monitors (ARM) has already been addressed by the FSAR. The same consequences exist for the relocated equipment. The exposure of the ARM equipment will not be increased. Also the sampling ability will not be changed. Therefore, no new possibility of malfunction will be introduced. The equipment neither adds nor creates an accident scenario different than already evaluated in the FSAR. This modification will maintain the margin of safety as defined in the TS bases.

Design Change Package, DCP-3123

DCP-3123, CB Pedestal Crane Replacement (Revision 0)

Description of Change

This change consists of replacing the existing hydraulic powered reactor containment building (CB) pedestal crane with an electric powered crane and modifying the existing pedestal crane seismic storage support to facilitate the new crane in its storage/laydown position.

Reason for Change

The purpose of this change is to improve reliability of the CB pedestal crane.

Safety Evaluation

There is no evaluation in the FSAR for changing or modifying a CB pedestal crane or the pedestal crane storage support. The new crane will be in a storage/laydown position during plant operations which is a seismic category I design. This assures that the crane will not interfere with the operability of any safety-related components during an earthquake. Also, lifting operations necessary to make the crane change-outs are governed by NUREG-0612 guidelines, "Control of Heavy Loads". The crane is not a safety-related component, but its bolt mounting system at both the operating and storage position is designed seismic category I. Thus, the probability or consequences of an accident previously evaluated in the FSAR will not be increased. No new accident possibilities are created.

The probability or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The crane is not a safety-related component and is seismically secured in a storage position during plant operation. There is no equipment being added by this change that has a safety function. Therefore, the possibility of equipment malfunction important to safety will not be affected. The margin of safety will not be reduced for the same reasons.

Design Change Package, DCP-3128

DCP-3128, Pressurizer Relief Valve Drains (Revision 0A)

Description of Change

This change will route piping from the existing pressurizer relief valve body drains which are presently capped, to downstream piping at a lower elevation.

Reason for Change

Relief valves RC-317A and RC-317B are weeping, causing damage to the valve internals and allowing condensate to collect in the valve bodies and discharge piping. This could cause water surge when the valve disk lifts, creating severe impact loads on downstream elbows. The new piping will drain any condensation build-up and prevent water surge or damage to the valves, piping, and pipe supports.

Safety Evaluation

The drain piping is non-safety and will have no impact on pressurizer relief valve function and will not increase the probability or consequences of an accident previously evaluated. The possibility of creating an accident different than already evaluated will not occur because the relief valve drain piping is non-safety and is on the downstream side of the valve. The probability or consequences of malfunction of equipment will not occur because body drain piping is located on the discharge side of the valve and does not impact the valve function. Also, because of the location of the piping the addition of the body drain line will not create the possibility of a malfunction to equipment already evaluated. Since there is no impact on the valve function, there will be no impact to the safety margin of the pressurizer relief valves.

Design Change Package, DCP-3142

DCP-3142, Seismic Qualifications for Valves ACC-126A and ACC-126B (Revision 0)

Description of Change

This design change replaces the existing positioners and air regulators for valves ACC-126A and ACC-126B with components that are qualified to function during and after a seismic event.

Reason for Change

The existing positioners and air regulators are not seismically qualified. The design basis for the valves requires that they function before, during and after a seismic event. The change will allow correction of this nonconformance.

Safety Evaluation

The probability of an accident previously evaluated in the FSAR will not be increased. The Auxiliary Component Cooling Water System (ACCW) is a support system for the Safety Injection System, Containment Spray System, Containment Isolation System, Main Steam Isolation System, Emergency Feedwater System and Containment Cooling System. These systems are required to operate following a loss of coolant accident (LOCA). The design change to valves ACC-126A and ACC-126B ensure that the valves (valve operators) are qualified to function after a seismic event. These valves are required to work post-LOCA to ensure that the wet cooling towers have a minimum of a thirty day post-LOCA water supply for long term cooling. The valves are to be returned to original design (automatic operation - temperature modulating valves) to ensure that the thirty day post-LOCA water supply for the wet cooling towers exists. Valves CC-620, SI-129A, and SI-129B are to be enhanced (have components qualified to operate after a seismic event) to ensure that the valves remain in their failed position after a design basis accident (CC-620 fails closed and SI-129A and B fail open).

The consequences of an accident previously evaluated in the FSAR will not be increased. Replacing the unqualified component with qualified components will allow the valves to function per original design. Valves ACC-126A and ACC-126B will be assured of operating after a seismic event. This will assure that the long-term cooling mechanism will be able to function and bring the plant to a safe shutdown. Valves CC-620, SI-129A and SI-129B will be assured of failing to their correct position and remaining in that position.

The possibility of an accident which is different than previously evaluated in the FSAR will not be created. Valves ACC-126A and ACC-126B are to be returned to their original design functions. Valves CC-620, SI-129A and SI-129B will be assured of failing to their designed failure positions and remaining there.

The probability of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The FSAR states that valves ACC-126A and ACC-126B , and CC-620 are actuated on receipt of a safety injection actuation signal (SIAS). Valves ACC-126A and ACC-126B have their setpoints reset to maintain Component Coolant Water System (CCWS) cold temperature at 115°F. Valve CC-620 is closed on receipt of an SIAS to divert component cooling water (CCW) to the RCPs to ensure the RCP seal reliability. For post-LOCA operation, a loss of air is assumed for flow control valves SI-306 an SI-307 since the air supplies for these valves are not seismically qualified. Under these circumstances, valves SI-306 and SI-307 fail open and cannot be remotely controlled from the main control room. The replacement of the unqualified components with qualified components will ensure that the valves ACC-126A and ACC-126B continue to function as designed (modulate). Valve CC-620 will be assured of closing and remaining closed to divert CCW to the RCPs. Valves SI-129A & B will be assured of remaining open and if the air system is not lost the valves will have a better ability of remaining operable, therefore maintaining control from the main control room.

The consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will no be increased. Replacement of the valve positioner on the valves ACC-126A and ACC-126B with positioners that are qualified to function after a seismic event will allow the valves to function as designed. The replacement of the positioner on valves CC-620, SI-129A and SI-129B will enhance the operation of the valves by ensuring that they remain in their failed position and if the air supply remains allowing the valves to be controlled from the control room.

The possibility of a malfunction of equipment important to safety different than already evaluated in the FSAR will not be created. Valves ACC-126A and ACC-126B will be returned to their original design function by using equipment qualified to function after a seismic event. Valves CC-620, SI-129A and SI-129B will be assured of remaining in their failed position. Also they will be enhanced to allow for remote operation if the air supply remains operable.

The margin of safety as defined in the bases of any TS will not be reduced because the valves are being modified to function as specified in the design basis.

Design Change Package, DCP-3163

DCP-3163, Timing Problem on PRM-IRM-7050 A and B (Revision 0)

Description of Change

This change installs a 60 Hz crystal oscillator assembly in the computers (RM-80) of CCW radiation monitors A and B (PRM-IRE-7050A and B). This assembly provides a 5 volt ac signal to the clock circuit in the radiation monitor's RM-80. This prevents noise on the 60 Hz ac power supplied to the radiation monitor from affecting the timing.

Reason for Change

The CCW radiation monitor A has experienced timing problems caused by noise spikes on the AC power which supply it. The timing problems cause the radiation monitor to count time too fast. This causes the source check test to fail, trend to update too fast, and a non-conservative error in the reading of the radiation monitor.

Safety Evaluation

This design change does not alter the intended function of the CCW radiation monitors. Their function is to monitor the CCW system for radioactive material and alarm when a preset level is reached. This informs operations personnel that a radioactive system is leaking into the CCW system and appropriate actions can be taken. The accidents evaluated in Chapter 15 of the FSAR do not discuss the CCW radiation monitors in the sequence of events. The probability of an accident previously evaluated in the FSAR is not increased because this design change has no effect on the required function of the CCW radiation monitors.

The accidents evaluated in Chapter 15 of the FSAR take no credit for mitigation of any consequences by the CCW radiation monitors. This design change has no affect on the CCW radiation monitor's ability to perform its function. The CCW system is a closed loop and does not directly discharge to the environment. The CCW radiation monitors do not monitor radioactive material discharged to the environment. Therefore, based on this, the consequences of an accident previously evaluated in the FSAR will not be increased.

The possibility of an accident which is different than already evaluated in the FSAR is not created. The CCW radiation monitor function is to monitor the CCW system for radioactive material and alarm when preset levels are exceeded. This design has no effect on this required function. They are not part of the RCS pressure boundary or required for safe shutdown of the reactor. Total failure of these radiation monitors would not cause an accident of any type.

The probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased nor is the possibility of a malfunction of equipment different than previously evaluated in the FSAR created. As currently designed, intermittent noise on the ac power supplied to these radiation monitors affects the timing circuit by causing it to count time too fast. This failure causes the radiation monitors to be out of service because it causes a non-conservative error. This change will prevent this malfunction from occurring by supplying a clean source of 60 Hz ac for the timing circuit. This change affects only the CCW radiation monitors. It is an internal change to the radiation monitor's RM-80. This change will not affect any other equipment. The CCW radiation monitors are safety class 1E and seismic category 1. This change is designed to meet these required qualifications. The CCW radiation monitors inform operations personnel that a radioactive system is leaking into the CCW system in order that appropriate actions can be taken. This change has no affect on this required function. The functions as described in the bases of the TS are not affected; therefore, the margin of safety will not be reduced.

Design Change Package, DCP-3165

DCP-3165, Installation of Electrocube Noise Suppressors (Revision 0)

Description of Change

This design change package installs Electrocube noise suppressors in the circuitry of radiation monitors PRM-IRE-001, 0648, 5107A and B and 6778.

Reason for Change

The suppressors are designed to eliminate false control functions and alarms due to electrical noise. The noise is induced into the electrical system by the normal opening and closing of control circuit relay or switch contacts. The noise causes various false radiation monitoring control functions and alarms to occur. Extensive trouble-shooting was performed and identified the relay or switch contacts causing the false actuation.

Safety Evaluation

The radiation monitors affected by this design change are part of the Effluent Radiological Monitoring System. This system is designed to meet the requirements of 10CFR20, 10CFR50, and follow the recommendation of Regulatory Guide 1.21 (June, 1974) to the extent specified in the TS during normal operations, including AOO. Only principal, normally radioactive or potentially radioactive release paths are monitored by the affected monitors. A complete failure of any one of these monitors during normal operations or AOO could result in an unmonitored release to the environment. The installation of the Electrocube noise suppressors will reduce the probability of an unmonitored release by ensuring that required control functions and alarms will occur when radiation levels exceed specified limits.

The design basis SGTR, Radioactive Waste System Leak or Failure, and the design basis Fuel Handling accidents bound the possible accidental release of radiation through any of the paths monitored by the affected non-safety-related radiation monitors (PRM-IRE-0001, 0648, 5107A&B, and 6778). Failure of these monitors would result in no change in release to the environment. For the accidents addressed, safety-related area monitors are available to aid in mitigating the consequences.

The radiation monitoring system will retain all of the present design features. This design change only installs minor components that do not change the operation of the system. These components suppress noise induced into the system electronics so that normal control functions and alarms can occur. No new failure modes are created by the implementation of this change.

Radiation monitors PRM-IRE-0001, 0648, 5107A&B, and 6778 are non-safety, non-seismic monitors with no electrical ties to equipment important to safety. This change installs noise suppressors within the non-safety-related monitor electronics. Therefore the possibility or consequences of a malfunction of equipment important to safety is not increased.

This change does not add, delete or revise any equipment important to safety. This change will allow the monitors to operate as specified in the FSAR. No new failure modes are created and no new equipment interfaces will be created by the implementation of this design change.

The TS requirements for these radiation monitors is not adversely affected by the implementation of this change because operation and design function as originally intended will not change. This change will provide better assurance that the margin of safety is maintained.

Design Change Package, DCP-3167

DCP-3167, Video Hard Copy Units (Revision 0)

Description of Change

This change installs a Tektronix video hard copy unit to the Emergency Offsite Facility (EOF), Backup EOF, and the Technical Support Center conference room.

Reason for Change

This change will aid in dose assessment and retention requirements.

Safety Evaluation

The copiers will not interact with a safety system required to prevent or mitigate the consequences of an accident or maintain the plant in a safe operational condition. Equipment important to safety previously evaluated is not being removed or altered by this change. The equipment added will assist those in the emergency and technical support centers to assess environmental conditions. The margin of safety will not be reduced.

Design Change Package, DCP-3219

DCP-3219, CECOR Analysis System / Plant Monitoring Computer Interface (Revision 0)

Description of Change

This change provides an on-line reactor CECOR analysis (ORCA) data transfer system. The Waterford 3 ORCA system will consist one ORCA computer and two modems which will be located in the plant computer room. The plant monitoring computer (PMC) will interface with the ORCA computer through a RS-232 communication data link. The PMC is quality related, non-safety, and non-seismic.

Reason for Change

The ORCA system will eliminate the need to transfer data to Power Computing Corporation in Dallas for coding and the need to rent a Combustion Engineering computer during restart for power ascension testing.

Safety Evaluation

The probability or consequences of an accident previously evaluated in the FSAR will not be increased because the ORCA computer will not interact with a safety system required to prevent or mitigate an accident or maintain the plant in a safe condition following an accident. The addition of the ORCA computer and data link will not increase the possibility of an accident since the computer will not be connected to any safety system required to maintain the plant in a safe operational condition. The probability or consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because the ORCA computer will not be connected to equipment important to safety. Also, for the same reason no new possibility of malfunction is created. This evaluation documents a change to FSAR Section 7.5 and figure 7.5.A-2, no margins of safety are affected.

Design Change Package, DCP-3230

DCP-3230, Removal of Decontamination Facility Partition Wall (Revision 0)

Description of Change

This design change removes a four foot high by ten foot long stainless steel partition wall in the decontamination facility.

Reason for Change

The partition wall was intended to serve as a barrier to separated clean from contaminated items. It is no longer needed for this purpose because of the manner in which the materials are handled. Also, the new liquid abrasive decontamination equipment requires removal of the wall to accommodate installation.

Safety Evaluation

The partition wall in the decontamination facility was intended to serve as a barrier to separate clean items from contaminated items. It serves no safety function as analyzed in the FSAR. The partition is non seismic, non- safety-related. It's presence or absence does not affect plant operation, postulated accidents or TSs. The decontamination room has a stainless steel liner plate acting as a protective coating over the concrete. The liner provides a smooth, water-tight surface that is easily decontaminated. The water-tight integrity of the stainless steel liner will not be degraded by this modification.

Design Change Package, DCP-3239

DCP-3239, Main Steam Isolation Valve (MSIV) - Thermal Relief Inlet Valve (Revision 0C)

Description of Change

This modification installs a ¼ inch, globe type, isolation valve upstream of each thermal relief valve to allow for brief isolation for maintenance or replacement of the relief valve. The set pressure of the thermal relief valve on each MSIV will be raised from 3000 psig to 3250 psig. An eight digit non-resettable counter is being added to each of the four hydraulic pump motors to record the number of start/stop cycles.

Reason for Change

The ¼ inch thermal relief valve on the 3/4 inch hydraulic supply to the main steam isolation valve (MSIV) actuator presently cannot be isolated for maintenance or replacement while the plant is operating. The thermal relief valve setpoint is being changed to reduce the probability of unnecessary cycling of the hydraulic system. The purpose of the counter is to document the exact number of pump starts to aid in future troubleshooting.

Safety Evaluation

The probability or consequences of an accident previously evaluated in the FSAR will not be increased because the piping being modified is non-safety-related and is not required for safe shutdown. The portion that is seismic I has been re-analyzed. This modification will not affect the closure capability of the MSIV's.

The possibility of an accident which is different than any already evaluated in the FSAR will not be created because the piping being modified is non-safety-related, and will not affect other safety-related components or systems.

This change does not increase the probability of equipment malfunction because failure of the hydraulic system being modified would not prevent the MSIV from performing its safety function, which is to close. The consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because if the equipment being modified failed, the MSIV would still close due to the nitrogen blanketing. The possibility of malfunction of equipment important to safety different than any already evaluated in the FSAR will not be created because no new equipment important to safety is being added by this design change. The margin of safety will not be reduced because the piping being modified is non-safety-related.

Design Change Package, DCP-3243

DCP-3243, Seismic Qualification of CS-125A and CS-125B Air Regulators (Revision 0)

Description of Change

This modification provides qualified air regulators which will replace the existing regulators to permit manual closing capability from the main control room during and after a seismic event. The modification also provides a test connection for air pressure switch CS-IPS-7122A.

Reason for Change

A nonconformance (CI 262785) identified that the containment spray header isolation valves CS-125A and CS-125B have unqualified air regulators. The same condition identification (CI) identified that the pressure switch was not provided with a test connection per design documentation.

Safety Evaluation

The engineering evaluation for the nonconformance determined that failure of the unqualified regulator would not prevent valves CS-125A and CS-125B from performing their intended function. Both valves are designed to "fail open" on loss of power or loss of air, therefore they are assured of maintaining the Engineered Safety Features Actuation System (ESFAS) actuated position. Thus, the consequences of the regulators failing would not affect the requirements of the containment spray system to aid in cooling the plant and decreasing containment internal pressure post accident. The changes increase assurance that accumulator pressure will be available after the accident mitigation phase for remote manual closure of CS-125A and CS-125B. Closure of CS-125A and CS-125B would enable shutdown cooling (SDC) to proceed while manual valves CS-117A and CS-117B are being closed. The change also increases assurance that CS-125A and CS-125B remain closed after a seismic event and that the sixty-five psig regulator setpoint is maintained after a seismic event, whether the containment spray actuation system (CSAS) is initiated or not.

PROCEDURE CHANGES

Plant Procedure, EP-002-050

EP-002-050, Offsite Dose Assessment (Manual) (Revision 10)

Description of Change

This revision involves changes to the Emergency Plan, and Section 2.3.3.2.2e and Appendix 7.5 A of the FSAR to replace the Computerized Emergency Planning and Data Acquisition System (CEPADAS) with the Safety Parameter Display System (SPDS) displays MARMOND 1 AND MARMOND 2 and the Microcomputer Dose Assessment Program (DOSECODE) as the primary method of performing dose projection during an emergency.

Reason for Change

This revision updates procedures to reflect the replacement of the CEPADAS program with the SPDS MARMOND 1 AND 2 displays and microcomputer dose assessment program as the primary dose assessment method.

The MARMOND 1 display provides fifteen minute average meteorological information. MARMOND 2 provides radiological effluent monitor readings and flow rates. The DOSECODE program is a computerized calculational program, loaded on IBM compatible Compaq microcomputers located in the emergency response facilities. This system will provide a method to rapidly assess the offsite impact of radiological releases using the data from the SPDS displays which is input by the operator.

Safety Evaluation

The probability of occurrence or the consequences of an accident previously evaluated in the FSAR will not be increased because no changes are being made to any safety-related system. All protective functions are provided by other systems. Similarly, the possibility for an accident of a different type than any evaluated previously in the FSAR will not be created.

The probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The reliability of gathering meteorological data will be increased due to enhanced software and equipment. Also, the loss of meteorological data gathering capability from the PMC is presently evaluated in the FSAR.

The margin of safety as defined in the basis for any TS is not reduced because no safety-related function will be affected by this change. The PMC performs no safety-related functions. The change to this procedure results in a change to the Emergency Plan but does not decrease the effectiveness of the Emergency Plan.

Plant Procedure, EP-002-051

EP-002-051, Offsite Dose Assessment (Computerized) (Revision 2)

Description of Change

This revision involves changes to the Emergency Plan, and Section 2.3.3.2.2e and Appendix 7.5A of the FSAR to replace the Computerized Emergency Planning and Data Acquisition System (CEPADAS) with the Safety Parameter Display System (SPDS) displays MARMOND 1 AND MARMOND 2 and the Microcomputer Dose Assessment Program (DOSECODE), as the primary method of performing dose projection during an emergency.

Reason for Change

This revision updates procedures to reflect the replacement of the CEPADAS program with the SPDS MARMOND 1 AND 2 displays and microcomputer dose assessment program as the primary dose assessment method.

The MARMOND 1 display provides fifteen minute-average meteorological information. MARMOND 2 provides radiological effluent monitor readings and flow rates. The DOSECODE program is a computerized calculational program, loaded on IBM compatible Compaq microcomputers located in the emergency response facilities. This system will provide a method to rapidly assess the offsite impact of radiological releases using the data from the SPDS displays which is input by the operator.

Safety Evaluation

The probability of occurrence or the consequences of an accident previously evaluated in the FSAR will not be increased because no changes are being made to any safety-related system. All protective functions are provided by other systems. Similarly the possibility for an accident of a different type than any evaluated previously in the FSAR will not be created.

The probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The reliability of gathering meteorological data will be increased due to enhanced software and equipment. Also, the loss of meteorological data gathering capability from the PMC is presently evaluated in the FSAR.

The margin of safety as defined in the basis for any TS is not reduced because no safety-related function will be affected by this change. The PMC performs no safety-related functions. The change to this procedure results in a change to the Emergency Plan but does not decrease the effectiveness of the Emergency Plan.

Plant Procedure, NOECP-254

NOECP-254, Control Of Erosion / Corrosion (Revision 0)

Description of Change

NOECP-254 replaces procedures PE-5-034 and PE-5-036. All of the examination requirements are contained in the new procedure, including additional examination requirements which were determined to be necessary.

Reason for Change

This change was made to consolidate the program into one procedure and to add additional improvements from reviews and evaluations of an NRC information notice, an NRC bulletin some NUREG documents and a NUMARC paper.

Safety Evaluation

This procedure is replacing PE-5-034 an PE-5-036. It contains all of the examination requirements of these two procedures, including additional examination requirements which were determined necessary.

Plant Procedure, OP-001-003

OP-001-003, RCS Drain Down (Revision 9)

Description of Change

The changes to the procedures were the result of a calculation revision. Calculation EC-M88-012 provides the basis for limitations on when all cold leg nozzle dams can be installed in the RCS to ensure reactor safety in the event of a loss of SDC. The limitations were revised based on lower relative decay heats compared to the previous decay heat model. The calculation establishes that core uncovery will occur two hours after a loss of SDC for the assumed RCS configuration if the reactor has been shutdown for fifteen days. The prior calculation required twenty-four days after shutdown for the same two hour core uncovery time. The basis for two hours is the conservative assumption that containment can be closed in less than two hours (unchanged from previous revisions). The requirement to keep containment closed for at least four days after reactor shutdown is maintained by reducing the time assumed to be required for containment closure from two hours to less than or equal to 1.75 hours. This time interval is conservative since it has been established that containment can be closed in less than 1.5 hours.

Reason for Change

The procedure changes resulted from revisions to a calculation which refined and improved the decay heat curve. This refinement allowed for changes to the specific limitations referenced in the procedures.

Safety Evaluation

Case 1: Four Day Criteria after Reactor Shutdown

The change in the assumed time required to close reactor containment from 2.0 hours to 1.75 hours will not affect the probability of any accidents. This will have no effect upon the probability of losing SDC or of any other accident.

The assumed time for closing reactor containment is being reduced from 2.0 hours to 1.75 hours. This is acceptable since the actual time to close containment is estimated to be 1.0 to 1.5 hours after a loss of SDC. Thus, conservatism is being reduced in the time assumed to complete this task. This will not increase the consequences of losing SDC, as a closure time of less than or equal to 1.75 hours will ensure containment closure within the required time. Containment can be closed in the allowed time period and there is sufficient time available for operators to respond to a loss of SDC (for example, by establishing high pressure safety injection [HPSI] flow to the RCS). This change cannot affect the consequences or probability of any other event.

The change in the assumed time for closing reactor containment will not create the possibility of an accident different than any already evaluated. The containment will be closed within the required 1.75 hour time after a loss of SDC, thus, ensuring that containment will be closed prior to the time when core uncovery could occur assuming no makeup is added to the reactor. There is no physical change to the plant, and no change to how it will be operated in shutdown conditions. The time required to close containment in response to a loss of SDC cannot affect the possibility of occurrence of any other event.

The change in the assumed time for closing reactor containment will not affect the probabilities of equipment malfunction. This change will not affect the method of equipment operation in response to a loss of SDC, thus, it can have no affect upon equipment malfunction probabilities. Because these changes will ensure that containment is closed prior to the time when core uncovery would occur after a loss of SDC, the consequences of malfunction of equipment important to safety will be unaffected and the possibility of malfunction is not created.

These changes will ensure that containment is closed within 1.75 hours after a loss of SDC, which is a sufficient and conservative period of time to complete containment closure. This is a reduction in the conservatism assumed in the time required to close containment. This is acceptable since the actual time to close containment is estimated to be between 1.0 to 1.5 hours after losing SDC. The requirement to close containment in less than or equal to 1.75 hours will ensure containment closure within the required time duration. Thus, this change does not affect the margin of safety since containment will be closed prior to core uncovery in the event of prolonged loss of SDC event of a prolonged loss of SDC.

Case 2: Installation of all Cold Leg Dams

The procedure changes associated with EC-M88-012-R02 concerning limitations when all cold leg nozzle dams are installed do not affect accident probabilities. The calculated time to core uncovery after a prolonged loss of SDC has no effect upon the probability of losing SDC. The time interval to take action to ensure containment closure (two hours) is unchanged. Accident consequences are not affected because, due to the limitations and required actions, containment will have been closed prior to core uncovery after a loss of SDC.

There are no changes to the plant as a result of the revised calculation or procedure changes. The changes concern the time required to respond to a previously evaluated accident (loss of SDC) and does not create a possibility of a different type accident than previously evaluated. The changes are based on the calculation of the time window in which recovery actions must be performed and reactor containment must be closed after a malfunction has caused a loss of SDC. There is no change in equipment malfunction probabilities associated with the calculation. Because the changes incorporate the conservative assumption that it takes two hours to close reactor containment, there is no change to the consequences of equipment malfunction and no possibility of a different malfunction than already evaluated. Reactor safety is maintained and the margin of safety is unchanged since containment will be closed prior to core uncovery.

Plant Procedure, OP-001-003

OP-001-003, RCS Drain Down (Revision 10, Change 1)

Description of Change

The procedure change provides closure inhibit of the SDC valves, described in FSAR Sections 9.3.6.1.2.b, 9.3.6.2.1, 9.3.6.2.2, 7.1.1.5, 7.6.1.1.1, 7.6.1.1.2 and Figure 7.4-1. The reactor coolant (RC) loops 1 and 2 SDC upstream suction isolation valves (SI-401A,B) interlocks are inhibited and the RC loops 1 and 2 SDC suction header isolation valves (SI-405A, B) are blocked (gagged) open.

Reason for Change

The purpose of this procedure change is to improve the reliability of the SDC in accordance with recommendations of the Shutdown Cooling Task Force.

Safety Evaluation

Although the interlocks for auto closure are inhibited, SI-401A and SI-401B and the RC loop 1 and 2 SDC suction header isolation valves (SI-407A,B) can be closed by the operator using keyswitches on control panel CP-8, preventing over pressurization of the affected piping equipment. Also, the RC loops 1 and 2 SDC header relief valves (SI-406A and B) will relieve at 430 psig. An alarm will alert the operator for appropriate action if pressure increases to 392 psia with its associated valve not closed. Defeat of the interlock function reduces the possibility of failure and subsequent loss of SDC. The consequences of an overpressurization event are not changed and no new system interactions or connections are created. The boundaries, margins of safety and accident response are not adversely affected.

Plant Procedure, OP-005-007

OP-005-007, Main Turbine Generator (Revision 5 Change A)

Description of Change

The deviation from the existing procedure will allow start-up of the turbine generator with a minimum of hydrogen pressure in the generator. The deviation requires that both stator coil water pumps be secured and hydrogen pressure in the generator be between two and three psig.

Reason for Change

Vibration on the turbine-generator #11 bearing has been a source of concern. One possible cause of this vibration is a hydrogen seal rub. It is theorized that the hydrogen pressure is binding the seal ring against the seal bracket. When the generator rotor comes up to speed it rises slightly reducing its clearance to the seal ring. Starting the turbine generator with a minimum of hydrogen pressure will allow the seal ring to rise with the rotor. This should help maintain seal ring to rotor clearance. Results of this deviation will supply valuable information to help evaluate the vibration on bearing #11.

Safety Evaluation

The procedure deviation will not increase the probability of occurrence of the following FSAR Chapter 15 accidents: increase in main steam flow, turbine trip, increase in main steam flow with a LOOP, turbine trip with single active failure, or FSAR Section 3.5, missile protection. The procedure deviation will allow the generator to run, unloaded, with a minimum of hydrogen pressure and no stator coil water flow. As long as the generator is not loaded (off-grid) there is no need for the cooling provided by these system. Prior to loading, the generator hydrogen pressure and stator coil water flow will be placed in normal operating ranges. The consequences of the accidents listed will not be altered in any way by this proposed deviation. The specific pressure of hydrogen in the generator and having stator coil water in service is not a condition affecting any of these accidents.

The procedure deviation does not affect any equipment important to safety. Directly affected systems are the generator, stator coil water, and generator gas. These systems do not contain components important to safety. The procedure deviation will not create the possibility of a different type of accident. The deviation alters the time frame in which the generator is pressurized with hydrogen and the stator coil water system is placed in service. The accidents evaluated in the SAR are applicable as analyzed.

The procedure deviation does not affect any equipment important to safety. Reducing hydrogen pressure and having no stator coil water flow during generator start-up will not alter the possibility of a malfunction of any equipment. These systems are utilized to provide cooling of the generator to maximize output power. Both systems will be returned to normal operating pressure and flow prior to synchronization with the grid. The purpose of this deviation is to attempt to decrease bearing vibration and thus decrease the possibility of malfunction of the generator. The procedure deviation does not affect any protective boundaries or margins of safety.

Plant Procedure, OP-010-001

OP-010-001, General Plant Operations (Revision 12 Change C)

Description of Change

Procedure OP-010-001 was temporarily revised to allow the nominal RCS pressure to be reduced to less than 2250 psia. The change widened the allowable pressurizer pressure bands from 2235-2265 psia to 2075-2265 psia.

Reason For Change

The purpose of the temporary change was to reduce pressurizer relief valve leakage to the quench tank and allow the valve to reseat.

Safety Evaluation

The change does not increase the probability of occurrence of any accident previously evaluated in the FSAR. RCS performance will be negligibly affected by this change. The RCS pressure will remain within the analyzed bounds of 2000 to 2300 psia and the frequency of occurrence of accident initiators will not be affected.

There is no increase in the probability of any analyzed accident. RCS pressure will still be maintained in a ± 50 psia control band about the nominal value by the Pressurizer Pressure Control System. The change has no impact with respect to the TS governing reactor vessel pressure and temperature limits.

The change will not increase the consequences of any accident previously analyzed in the FSAR. With a pressurizer steady-state pressure below 2150 psia at 100% power, a SIAS or containment isolation actuation signal (CIAS) may occur during a reactor trip. A caution is added to procedures informing the operators of this possibility to eliminate any unexpected responses.

There will be no increases in the consequences of a reactor trip. The small decrease in pressurizer temperature associated with the change has no measurable negative impact on reactor safety and the small perturbations in the dynamics of pressurizer response due to the lower temperatures and pressure has already been accounted for in the FSAR. Thus, there is no increase in the probability of occurrence of a malfunction of equipment important to safety.

The change does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. There are no new system interactions or connections created as a result of this change and there is no new methodology with which the RCS is being operated. Thus, there is no possibility of the creation of a new and different type of accident than any previously evaluated in the FSAR.

There are no new failure mechanisms that could be created by the proposed change. There will be no change to how equipment is operated, only changes to the setpoint for nominal RCS pressure, which will not create the possibility of malfunction of equipment important to safety of a different type than previously evaluated in the FSAR. The steady state RCS pressure is maintained between 2025 psia and 2250 psia. This steady state range is bounded by initial conditions assumed in the accident analysis and is therefore acceptable. Additionally, RCS pressure is maintained within the steady state design pressure of 2500 psia.

There is no degradation in the performance of the pressurizer safety valves associated with the subject temporary reduction in RCS pressure, thus there will be no reduction in any margin of safety based upon RCS pressure response for any event analyzed in the FSAR.

Plant Procedure, OP-100-007

OP-100-007, Shift Turnover (Revision 6, Change B)

Description of Change

Procedure OP-100-007 was temporarily revised to allow the nominal RCS pressure to be reduced to less than 2250 psia. The change widened the allowable pressurizer pressure bands from 2235-2265 psia to 2075-2265 psia.

Reason For Change

The purpose of the temporary change was to reduce pressurizer relief valve leakage to the quench tank and allow the valve to reseat.

Safety Evaluation

The change does not increase the probability of occurrence of any accident previously evaluated in the FSAR. RCS performance will be negligibly affected by this change. The RCS pressure will remain within the analyzed bounds of 2000 to 2300 psia and the frequency of occurrence of accident initiators will not be affected.

There is no increase in the probability of any analyzed accident. RCS pressure will still be maintained in at 50 psia control band about the nominal value by the Pressurizer Pressure Control System. The change has no impact with respect to the TS governing reactor vessel pressure and temperature limits.

The change will not increase the consequences of any accident previously analyzed in the FSAR. With a pressurizer steady-state pressure below 2150 psia at 100% power, a SIAS or containment isolation actuation signal (CIAS) may occur during a reactor trip. A caution is added to procedures informing the operators of this possibility to eliminate any unexpected responses.

There will be no increases in the consequences of a reactor trip. The small decrease in pressurizer temperature associated with the change has no measurable negative impact on reactor safety and the small perturbations in the dynamics of pressurizer response due to the lower temperatures and pressure has already been accounted for in the FSAR. Thus, there is no increase in the probability of occurrence of a malfunction of equipment important to safety.

The change does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. There are no new system interactions or connections created as a result of this change and there is no new methodology with which the RCS is being operated. Thus, there is no possibility of the creation of a new and different type of accident than any previously evaluated in the FSAR.

There are no new failure mechanisms that could be created by the proposed change. There will be no change to how equipment is oper d, only changes to the setpoint for nominal RCS pressure, whi ill not create the possibility of malfunction of equipment imp ant to safety of a different type than previously evaluated in the SAR. The steady state RCS pressure is maintained between 2025 psia and 2250 psia. This steady state range is bounded by initial conditions assumed in the accident analysis and is therefore acceptable. Additionally, RCS pressure is maintained within the steady state design pressure of 2500 psia.

There is no degradation in the performance of the pressurizer safety valves associated with the subject temporary reduction in RCS pressure, thus there will be no reduction in any margin of safety based upon RCS pressure response for any event analyzed in the FSAR.

Plant Procedure, OP-901-046

OP-901-046, Shutdown Cooling Malfunction (Revision 6)

Description of Change

The changes to the procedures were the result of a calculation revision. Calculation EC-M88-012 provides the basis for limitations on when all cold leg nozzle dams can be installed in the RCS to ensure reactor safety in the event of a loss of SDC. The limitations were revised based on lower relative decay heats compared to the previous decay heat model. The calculation establishes that core uncovery will occur two hours after a loss of SDC for the assumed RCS configuration if the reactor has been shutdown for fifteen days. The prior calculation required twenty-four days after shutdown for the same two hour core uncovery time. The basis for two hours is the conservative assumption that containment can be closed in less than two hours (unchanged from previous revisions). The requirement to keep containment closed for at least four days after reactor shutdown is maintained by reducing the time assumed to be required for containment closure from two hours to less than or equal to 1.75 hours. This time interval is conservative since it has been established that containment can be closed in less than 1.5 hours.

Reason for Change

The procedure changes resulted from revisions to a calculation which refined and improved the decay heat curve. This refinement allowed for changes to the specific limitations referenced in the procedures.

Safety Evaluation

Case 1: Four Day Criteria after Reactor Shutdown

The change in the assumed time required to close reactor containment from 2.0 hours to 1.75 hours will not affect the probability of any accidents. This will have no effect upon the probability of losing SDC or of any other accident.

The assumed time for closing reactor containment is being reduced from 2.0 hours to 1.75 hours. This is acceptable since the actual time to close containment is estimated to be 1.0 to 1.5 hours after a loss of SDC. Thus, conservatism is being reduced in the time assumed to complete this task. This will not increase the consequences of losing SDC, as a closure time of less than or equal to 1.75 hours will ensure containment closure within the required time. Containment can be closed in the allowed time period and there is sufficient time available for operators to respond to a loss of SDC (for example, by establishing HPSI flow to the RCS). This change cannot affect the consequences or probability of any other event.

The change in the assumed time for closing reactor containment will not create the possibility of an accident different than any already evaluated. The containment will be closed within the required 1.75 hour time after a loss of SDC, thus, ensuring that containment will be closed prior to the time when core uncovery could occur assuming no makeup is added to the reactor. There is no physical change to the plant, and no change to how it will be operated in shutdown conditions. The time required to close containment in response to a loss of SDC cannot affect the possibility of occurrence of any other event.

The change in the assumed time for closing reactor containment will not affect the probabilities of equipment malfunction. This change will not affect the method of equipment operation in response to a loss of SDC, thus, it can have no affect upon equipment malfunction probabilities. Because these changes will ensure that containment is closed prior to the time when core uncovery would occur after a loss of SDC, the consequences of malfunction of equipment important to safety will be unaffected and the possibility of malfunction is not created.

These changes will ensure that containment is closed within 1.75 hours after a loss of SDC, which is a sufficient and conservative period of time to complete containment closure. This is a reduction in the conservatism assumed in the time required to close containment. This is acceptable since the actual time to close containment is estimated to be between 1.0 to 1.5 hours after losing SDC. The requirement to close containment in less than or equal to 1.75 hours will ensure containment closure within the required time duration. Thus, this change does not affect the margin of safety since containment will be closed prior to core uncovery in the event of prolonged loss of SDC event of a prolonged loss of SDC.

Case 2: Installation of all Cold Leg Dams

The procedure changes associated with EC-M88-012-R02 concerning limitations when all cold leg nozzle dams are installed do not affect accident probabilities. The calculated time to core uncovery after a prolonged loss of SDC has no effect upon the probability of losing SDC. The time interval to take action to ensure containment closure (two hours) is unchanged. Accident consequences are not affected because, due to the limitations and required actions, containment will have been closed prior to core uncovery after a loss of SDC.

There are no changes to the plant as a result of the revised calculation or procedure changes. The changes concern the time required to respond to a previously evaluated accident (loss of SDC) and does not create a possibility of a different type accident than previously evaluated. The changes are based on the calculation of the time window in which recovery actions must be performed and reactor containment must be closed after a malfunction has caused a loss of SDC. There is no change in equipment malfunction probabilities associated with the calculation. Because the changes incorporate the conservative assumption that it takes two hours to close reactor containment, there is no change to the consequences of equipment malfunction and no possibility of a malfunction different than already evaluated. Reactor safety is maintained and the margin of safety is unchanged since containment will be closed prior to core uncovery.

Plant Procedure, OP-901-054

OP-901-054, Loss of Vital Instrument Bus (Revision 4 Change 5)

Description of Change

OP-901-054 is reformatted for human factor concerns in accordance with OP-100-013, Writer's Guide. This includes descriptions of other plant components affected by the loss of a vital instrument bus, and steps to be taken in response to these effects.

Reason for Change

The revision is for clarification of operator actions during an unusual occurrence.

Safety Evaluation

Upon a loss of instrument bus 3MA-S or 3MB-S during a loss of normal AC power / station blackout, the associated Train CC, ACC, and ultimate heat sink must be considered for operability status. The result would be a loss of component cooling header temperature indication, loss of ACC flow control (ACC-126A or B will fail open), and loss of dry cooling tower (DCT) and wet cooling tower (WCT) fan sequencing. The changes made to this procedure are to maintain this equipment "functional", not to return it to "operable" status (CCW, itself, may still be considered operable because header temperature indication is not vital to its function -- although DCT fan sequencing is affected).

Since auto control of DCT and WCT fans and ACC-126A(B) is lost, steps were added to control these systems in manual. It must be considered that this equipment will not operate properly even if these steps were not taken. These steps were added to minimize the plant effects once one of these instrument busses was lost.

Upon a loss of 3MC-S or 3MD-S buss the associated train (A for 3MC-S, B for 3MD-S) chilled water (CHI), control room HVAC (HVC), controlled ventilation area system (CVAS), shield building ventilation (SBV) and fuel handling building HVAC (HVF) will become inoperable due to the loss of various control features. Steps were added to secure this equipment to prevent any physical damage from occurring.

The associated steam generator (SG) atmospheric dump valve (ADV) will fail closed, and control room indications for the associated EDG will be lost. Therefore, steps were added to operate the ADV locally, if necessary, so that guidance will be readily available if the instrument bus were lost while this valve was in operation. Steps were added to take local control of the EDG, if running, so that its control and monitoring location will be the same, therefore, reducing the likelihood of EDG damage. The emergency start feature of the EDG will remain unaffected.

With all the above considered, the changes made to OP-901-054 will not increase the likelihood of either a loss of normal AC power or a station blackout.

Upon the loss of instrument bus 3MC-S or 3MD-S during an asymmetric steam generator transient (MSIV closure), the associated ADV will fail closed. The ADV will not open if the associated MSIV were to go closed; therefore, the steps added to place the ADV in local control will have no effect in this condition.

During a LOCA, the ACC pumps and the DCT fans are designed to automatically operate to maintain CCW temperature at 115°F. With ACC pump A(B) dish valve in manual and the DCT fans in manual this will not occur without operator interaction. Had this equipment been left in auto control, this would not occur either. The steps added to this procedure concerning this temporarily maintain the plant stable until the instrument bus is returned to service. These steps will not increase the likelihood of a LOCA.

During a LOCA, the CHW, control room emrgency filtration unit, CVAS, and SBV systems are designed for auto start. When 3MC-S or 3MD-S instrument bus is lost, one train will not operate properly. The heaters will not energize and the units will not trip on low filter differential temperature. Steps were added to this procedure to secure this equipment and to review its associated TS. This will not prevent the equipment from auto starting on an SIAS, so the main purpose to the procedure steps is to secure the equipment to prevent damage and to make the operator aware that this equipment is out of service. The steps in this procedure will not increase the radiation release consequences of a LOCA; but, in fact, will reduce the problems that could incur if a LOCA were to happen while an instrument bus was down. They make the operator aware of safety equipment that will not operate properly during an SIAS. When both trains start up on the SIAS, it is permitted to secure one train of control room emergency filtration units, CVAS, and SBV.

Steps were added to the fuel handling accident (FHA) procedure to secure the fuel handling building (FHB) emergency filtration units affected by the loss of an instrument bus and to review its associated TS. This is to prevent damage to the FHB emergency filtration units due to its loss of heaters and trip functions. This makes the operator aware that the unit will not operate properly during an FHA. When both trains start on the FHA, it is permitted to secure one train to avoid problems that may occur while an instrument bus is down.

Steps were added to this procedure for local operation of ADV's, if ADV operation is required. The ADV will fail closed on a loss of its associated instrument bus, and if operated locally, an operator will be at the local station, and thereby available to close the ADV in the event of a SGTR. This will not increase the consequences of a SGTR because the ADV is not designed to auto close in this event and requires operator action.

CCW, ACCW, ultimate heat sink, control room emergency filtration units, CVAS, SBV and FHB emergency filtration units are effected by the loss of an instrument bus. Steps were added to this procedure to make the operator aware of this and to guide him on what actions to take and what TSs to review. The steps added, therefore, do not increase the probability of a malfunction; but, instead inform the operator that the malfunction has occurred. The operator then takes action to prevent physical damage to the equipment so that it will be available once the instrument bus is restored. Therefore, the effect on the safety equipment is in preserving it for use once the off-normal is exited. During a LOCA, SGTR, and FHA, this equipment (except for CCW) is already out of service due to the loss of a vital instrument bus. Some of the equipment remains functional (ACCW and ultimate heat sink) as long as it is controlled in manual. this will not increase the consequences of component malfunction. However, it will preserve the equipment so that it may be useful once the instrument bus is restored to service. All steps taken are within the guidance of TSs.

There are no new system interactions created within this procedure revision. The equipment operated in manual is designed to do so, and is already out of service as per TSs. Therefore, no new type of accident could be created from this.

The only new connections made are the I&C test instruments, which will be connected to the already de-energized PVC cabinets. this will be a temporary modification, lasting only until the process analog control cabinet is re-energized. This will create no new type accident. The instrumentation is the same as is used during normal testing of the equipment.

The equipment that is to be monitored or operated in manual have lost their instrument inputs due to the loss of the instrument bus. The steps added to this procedure could not create a new type malfunction of the equipment, because they are designed for manual operation in the event that auto operation were to fail.

Plant Procedure, OP-902-002

OP-902-002, Loss of Coolant Accident Recovery Procedures (Revision 3)

Description of Change

This revision incorporates the latest changes to CEN-152, Rev.3, NRC Inspection Audit, and NUMARC 87-00, Section 3, Guidelines in Compliance With Station Blackout Rule of 10CFR50.63. The change to the procedure as described in FSAR Section 6.3.3.4 states that "the operator terminates charging pump operation between one half and two hours following the event." This revision to OP-902-002 allows continued operation of the charging pumps.

Reason for Change

To update and improve the guidelines as specified in the above references.

Safety Evaluation

The maximum boric acid concentration in the BAMT has been reduced from 12 wt% to 3.5 wt%. If the total contents of both BAMTs at the maximum boric acid concentration were injected into the RCS, the amount of boron added to the RCS would still be less than half of that assumed in the long term cooling analysis. Furthermore, it would take an additional twelve hours for the charging pumps to inject enough boron from the refueling water storage pool (RWSP) to equal the amount assumed in the long term cooling analysis. Since this is well beyond the time that a flushing flow in the reactor vessel is established by simultaneous hot and cold side injection (two to four hours after SIAS), boron precipitation will not occur.

Since the entire contents of both BAMTs can be injected without a boron precipitation concern, the time restriction on switching suction can be eliminated. This gives the operator the greatest flexibility since he can use any portion or all of the BAMT inventory depending on conditions (such as required shutdown margin).

Plant Procedure, OP-902-007

OP-902-007, Steam Generator Tube Rupture Recovery Procedure (Revision 3)

Description of Change

This revision incorporates the latest changes to CEN-152, Rev. 3, NRC Inspection Audit conducted in July, 1988, and NUMARC 87-00, Section 3 guidelines in compliance with Station Blackout rule of 10CFR50.63.

Reason for Change

To update and improve the guidelines as specified in the above references.

Safety Evaluation

FSAR Section 6.3.3.4 states that "the operator terminates changing pump operation between one and a half and two hours following the event." OP-902-007 Revision 3, Steps 41 and 43, allow continued operation of the charging pumps.

The following justifies switching the charging pump suction from the BAMT to the RWSP after thirty minutes to one hour from a SIAS in the emergency operation procedures (EOPs). This is different from the long term cooling (boron precipitation) analysis in the FSAR which assumed that the charging pumps inject water from the BAMT for two hours after SIAS occurs and then are stopped. The concern addressed here is that continued operation of the charging pumps after a large break LOCA (the most limiting event) will cause more boron to be injected to the reactor vessel than assumed in the safety analysis.

For Cycle 2, the maximum boric acid concentration in the BAMT's has been reduced from 12 wt.% to 3.5 wt.%. If the total contents of both BAMTs at the maximum boric acid concentration were injected into the RCS, the amount of boron added to the RCS would still be less than half of that assumed in the long term cooling analysis. Furthermore, it would take an additional twelve hours for the charging pumps to inject enough boron from the RWSP to equal the amount assumed in the long term cooling analysis. Since this is well beyond the time that a flushing slow in the reactor vessel is established by simultaneous hot and cold side injection (two to four hours after SIAS), boron precipitation will not occur.

Therefore, switching the charging pump suction from the BAMTs to the RWSP at thirty minutes to one hour after SIAS occurs is bounded by the long term cooling safety analysis in the FSAR. Furthermore, since the entire contents of both BAMTs can be injected without a boron precipitation concern, the time restriction on switching suction can be eliminated. This gives the operator the greatest flexibility since he can use any portion or all of the BAMT inventory depending on conditions (such as required shutdown margin). However, a step should be included in the EOPs to remind the operator to switch suction to the RWSP before the BAMTs are empty to preclude cavitation and gas binding of the charging pumps.

It should be noted that because the BAMT boron concentration is much lower for Cycle 2, the minimum time required to inject enough boron to achieve the required shutdown margin during a cooldown is much longer. For a cooldown to 200°F and one changing pump available, the time to reach a shutdown margin of 5.15% has been conservatively calculated to be two to three hours after the start of emergency boration. To be shutdown by 2% (shutdown margin of 1%) would require one to one and a half hours of emergency boration. This may be too long to be practical as a general emergency boration termination criteria for all situations. An alternative is to eliminate the time criteria and allow the operator to switch suction to the RWSP at any time (prior to emptying the BAMTs) based on the particular event and shutdown margin calculation was completed and could be terminated as needed.

A second request was to verify that the two to four hours post-LOCA time to establish simultaneous hot and cold side safety injection is still applicable to Cycle 2. There have been no changes to the long term cooling safety analysis or plant design that would cause this time to change. Therefore, this step in the LOCA EOP should remain the same.

Based on this information, the proposed change does not involve an unreviewed safety question.

Plant Procedure, OP-903-003

OP-903-003, Charging Pump Operability Check (Revision 7, Change 3)
OP-903-004, BAM Pump Operability Test (Revision 7, Change 5)
OP-903-035, Containment Spray Pump Operability Check (Revision 6 Change 6)
OP-903-046, Emergency Feedwater Pump Operability Check (Revision 7, Change 7)
OP-903-050, CCW/ACCW Pump Operability Test (Revision 7, Change 5)
OP-903-063, Chilled Water Pump Operability Verification (Revision 6, Change 5)

Description of Change

The above revisions allowed for the use of a Fluke model 51 k/j thermometer on 0 - 500°F range to record bearing temperature in lieu of a portable RTD Temperature Monitor with a range of 0 - 250°F and an accuracy of ± 5%.

Reason for Change

The required M&TE was not available on site. The Fluke model 51 k/j was available and satisfies the accuracy requirement.

Safety Evaluation

The change does not affect the ability of the involved systems to perform their safety functions. The accuracy of the proposed temperature monitoring instrument ± 10°F is within the required accuracy for the range of instruments as required by ASME Boiler and Pressure Vessel Code Section XI, IWP 4000. The required accuracy is ± 5% of full range with full range being three times the reference value. For all of the pumps involved, the required accuracy will be greater than ± 10°F. Therefore, the ability to detect imminent failure is not diminished and there is no increase in the probability of occurrence or consequences of a malfunction of equipment important to safety. No changes to the facility are being made and the pump bearing temperatures will be read within accuracy requirements, enabling evaluation of pump operating characteristics. Thus, the possibility of an accident or malfunction of equipment important to safety of a different type than previously evaluated in the SAR is not created. Equipment operability will not be affected by this change, also the inservice test requirements will be met. The margin of safety as defined in the bases of the TSs is not reduced.

Plant Procedure OP-903-021

OP-903-021, Radioactive Gas Effluent Monitoring System Source Check (Revision 2)

Description of Change

This change deletes steps in OP-903-021 referring to PRM-IRE-0001 and adds steps referring to PRM-IRE-0002. The Condenser Off-gas Radiation Monitoring System has been modified by deleting system redundancies and providing a permanent sample cooling and conditioning skid. Radiation monitor PRM-IRE-0001 is being deleted. The function of this monitor will be relocated to monitor PRM-IRE-0002. Sample cooling will be provided by a self contained package chiller.

Reason for Change

The procedure change reflects a change in the plant design. The design change is a system improvement that will improve reliability and replace the need for potable water currently being used as a cooling medium.

Safety Evaluation

The design basis SGTR bounds the possible accidental release of radiation through the turbine building vent. Other smaller primary to secondary leaks result in far less radioactive gas inventory in the steam system. SGTR may last fifteen minutes before a low pressurizer pressure causes an automatic reactor trip and subsequent main steam line isolation upstream of the condenser. A less serious SGTR may not result in a reactor trip, and operator action to isolate the leaking steam generator may occur thirty minutes after the SGTR. The wide range monitor modification takes all of the functions of the narrow range monitor without sacrificing sensitivity or performance. No greater or lesser amount of radioactive gases exit the turbine building vent after a SGTR. The wide range monitor change does not eliminate a backup narrow range monitor control function because one never existed, nor was there a requirement for a backup.

The new design retains the previous conditions where monitor power is lost in a LOOP and diversion valves fail as is. Although the feature does exist to manually connect the monitor to the EDGs in a LOOP, the condenser vacuum pumps will not operate, therefore, monitoring is not required. Under the worst conditions, SGTR and LOOP scenarios, if the monitor becomes manually loaded on to an emergency power supply, the non-safety-related solenoid sample isolation valves would remain closed and the motorized diversion valves would not automatically reposition. This occurrence would leave the monitor in a useless mode. In the worst case SGTR scenario, TS controls on secondary chemistry provides more general public dose protection than the off-gas diversion system. The bulk of the effluent leaves through other pathways not measured by this monitor.

The order of events during an accident prohibits any but the smallest quantities of radioactive gas to enter the off-gas system because of main steam isolation features. During the early phases of SGTR, the reactor control system would attempt to replace the lost RCS inventory. The bulk of the plant systems, including power production, would continue so normal plant conditions and responses to radioactivity in the off-gas is expected. A concurrent LOOP causes main steam isolation and the end to the source of radioactive off-gas. The new design does not re-arrange any safety-related system or delete functions which could create a new accident.

The wide range monitor skid only changes to adopt the narrow range monitor output function (diversion valve control). A narrow range monitor is part of the wide range monitor skid. Further, the new sample conditioning system and revised sample line routing increases the reliability of the wide range monitor by reducing the probability of moisture intrusion into the monitor.

A wide range off-gas monitor failure results in consequences no different than a power failure to the non-safety-related, "fail as-is" diversion valves. A monitor failure causes a loss of automatic control (diversion) and effluent measurements no different than the current design.

This design change does not add, delete, or revise any safety-related components. All equipment modified by this change is located in the turbine building. Therefore, the possibility of a malfunction of equipment important to safety is not created by the implementation of this change.

The margin of safety comes from adherence to regulations requiring condenser off-gas monitoring. Because radiation monitor PRM-IRE-0002 can perform all functions presently performed by radiation monitor PRE-IRE-0001, the present regulatory requirements are not affected by this change. The TSs list the important radiation monitoring functions in the plant. The only listing relevant to the Main Condenser Evacuation System (MCES) falls under the Effluent Accident Monitor heading. The wide range monitor PRM-IRE-0002 can accommodate that without PRM-IRE-0001. The TS lists lower limits of detection (LLD) for MCES monitoring. Monitor PRM-IRE-002 contains a low range detector which will monitor the LLD.

Plant Procedure OP-903-030

OP-903-030, Safety Injection Pump Operability Verification (Revision 6, Change A)

Description of Change

Revision 6 to OP-903-030, "Safety Injection Pump Operability Verification", temporarily changed the safety injection pump recirculation flow from 28.7 to 27.6 gallons per minute (gpm) by throttling stop check valve SI-205B.

Reason for Change

The temporary change was made to allow plant staff to collect data at baseline readings to comply with ASME Boiler and Pressure Vessel Code Section XI requirements.

Safety Evaluation

Throttling of stop/check valve SI-205B does not affect the actual safety function of the pump because the deviation in recirculation flow will not go below the minimum recirculation flow of twenty-five gpm. The Safety Injection System HPSI pump "B" is a mitigating safety system and will not add to or increase the likelihood of an accident. Since the safety function is unaffected there is no increase in the consequences of an accident previously evaluated.

Throttling of valve SI-205B has no effect on the likelihood of system or component malfunction since the pump will be used for testing only and the minimum recirculation flow requirements will be met. The component involved is a pre-existing component and no other equipment or system interface is created or affected by the change. The testing is being performed to meet inservice test requirements and will assure that the system has not degraded with time and use. The margin of safety is thus maintained. There is no increase in probabilities for equipment malfunction or accidents which have been evaluated in the FSAR.

Plant Procedure, OP-903-072

OP-903-072, Containment Building Penetration Check (Revision 5)

Description of Change

The changes to the procedures were the result of a calculation revision. Calculation EC-M88-012 provides the basis for limitations on when all cold leg nozzle dams can be installed in the RCS to ensure reactor safety in the event of a loss of SDC. The limitations were revised based on lower relative decay heats compared to the previous decay heat model. The calculation establishes that core uncovery will occur two hours after a loss of SDC for the assumed RCS configuration if the reactor has been shutdown for fifteen days. The prior calculation required twenty-four days after shutdown for the same two hour core uncovery time. The basis for two hours is the conservative assumption that containment can be closed in less than two hours (unchanged from previous revisions). The requirement to keep containment closed for at least four days after reactor shutdown is maintained by reducing the time assumed to be required for containment closure from two hours to less than or equal to 1.75 hours. This time interval is conservative since it has been established that containment can be closed in less than 1.5 hours.

Reason for Change

The procedure changes resulted from revisions to a calculation which refined and improved the decay heat curve. This refinement allowed for changes to the specific limitations referenced in the procedures.

Safety Evaluation

Case 1: Four Day Criteria After Reactor Shutdown

The change in the assumed time required to close reactor containment from 2.0 hours to 1.75 hours will not affect the probability of any accidents. This will have no effect upon the probability of losing SDC or of any other accident.

The assumed time for closing reactor containment is being reduced from 2.0 hours to 1.75 hours. This is acceptable since the actual time to close containment is estimated to be 1.0 to 1.5 hours after a loss of SDC. Thus, conservatism is being reduced in the time assumed to complete this task. This will not increase the consequences of losing SDC, as a closure time of less than or equal to 1.75 hours will ensure containment closure within the required time. Containment can be closed in the allowed time period and there is sufficient time available for operators to respond to a loss of SDC (for example, by establishing HPSI flow to the RCS). This change cannot affect the consequences or probability of any other event.

The change in the assumed time for closing reactor containment will not create the possibility of an accident different than any already evaluated. The containment will be closed within the required 1.75 hour time after a loss of SDC, thus, ensuring that containment will be closed prior to the time when core uncovery could occur assuming no makeup is added to the reactor. There is no physical change to the plant, and no change to how it will be operated in shutdown conditions. The time required to close containment in response to a loss of SDC cannot affect the possibility of occurrence of any other event.

The change in the assumed time for closing reactor containment will not affect the probabilities of equipment malfunction. This change will not affect the method of equipment operation in response to a loss of SDC, thus, it can have no affect upon equipment malfunction probabilities. Because these changes will ensure that containment is closed prior to the time when core uncovery would occur after a loss of SDC, the consequences of malfunction of equipment important to safety will be unaffected and the possibility of malfunction is not created.

These changes will ensure that containment is closed within 1.75 hours after a loss of SDC, which is a sufficient and conservative period of time to complete containment closure. This is a reduction in the conservatism assumed in the time required to close containment. This is acceptable since the actual time to close containment is estimated to be between 1.0 to 1.5 hours after losing SDC. The requirement to close containment in less than or equal to 1.75 hours will ensure containment closure within the required time duration. Thus, this change does not affect the margin of safety since containment will be closed prior to core uncovery in the event of prolonged loss of SDC event of a prolonged loss of SDC.

Case 2: Installation of all Cold Leg Dams

The procedure changes associated with EC-M88-012-R02 concerning limitations when all cold leg nozzle dams are installed do not affect accident probabilities. The calculated time to core uncovery after a prolonged loss of SDC has no effect upon the probability of losing SDC. The time interval to take action to ensure containment closure (two hours) is unchanged. Accident consequences are not affected because, due to the limitations and required actions, containment will have been closed prior to core uncovery after a loss of SDC.

There are no changes to the plant as a result of the revised calculation or procedure changes. The changes concern the time required to respond to a previously evaluated accident (loss of SDC) and does not create a possibility of a different type accident than previously evaluated. The changes are based on the calculation of the time window in which recovery actions must be performed and reactor containment must be closed after a malfunction has caused a loss of

SDC. There is no change in equipment malfunction probabilities associated with the calculation. Because the changes incorporate the conservative assumption that it takes two hours to close reactor containment, there is no change to the consequences of equipment malfunction and no possibility of a malfunction different than already evaluated. Reactor safety is maintained and the margin of safety is unchanged since containment will be closed prior to core uncovery.

Plant Procedure, OP-903-102

OP-903-102, Safety Channel ENI Functional Test (Revision 4)

Description of Change

This changes Attachment 10.1, Section 8.2 and 8.3, Local Log Safety Channel Meter Reading from "7.9×10^{-5} to 1.3×10^{-4}" to "0.8 to 1.3".

Reason for Change

This will allow OP-903-102 to be performed after "10^{-4}" bistable is adjusted to 1% during low power physics testing.

Safety Evaluation

Raising the trip set-point will not increase the probability that a reactor transient will occur. If transient occurs, the reactor trip (from CPC) will be enabled at 1% power. The analog trips will operate normally. The raising of the setpoint will not increase the possibility of any kind of equipment malfunction. The analog reactor trips will operate normally. The CPC trips will be enabled at 1%. Raising this setpoint will not cause a new type of accident or malfunction. Special Test Exception 3.10.3 allows such an adjustment. Therefore, no unreviewed safety question exists.

Plant Procedure, OP-903-114

OP-903-114, Local Leak Rate Test (Revision 2 Change E)

Description of Change

This is a change to the local leak rate test (LLRT) instructions for penetration No. 26 to allow testing valve CVC-103 without also doing a LLRT on valve CVC-109.

Reason for Change

The reason for the change was to minimize the number of valve manipulations, thereby minimizing personnel exposure and man hours. The LLRT is being performed as a post maintenance retest for work performed on CVC-103 and is not being performed as a scheduled surveillance which would require testing of valve CVC-109.

Safety Evaluation

The LLRT will be performed in Mode 5 when containment integrity is not required. CVC-101 and CVC-103 are provided to minimize the effect of a letdown line break. CVC-103 and CVC-109 are provided to ensure containment integrity. The test will be performed with CVC-101 and CVC-109 closed. This test does not increase the potential for losing shutdown cooling. CVC-101 and CVC-103 are the normal test boundary for an LLRT of penetration 26 (isolated by CVC-103 and CVC-109). The test boundary for this LLRT of CVC-103 is CVC-101 and CVC-109 which are bounded by the normal test boundary. The same test pressure, test fluid (gas), and acceptance criteria are used per OP-903-114. The only difference is that CVC-109 is not being tested. There are no new system interactions to consider which would increase the possibility of an accident of a different type.

No new methods of failure are created. The change to OP-903-114 alters the test boundary from CVC-111B to CVC-109 and does not test CVC-109. The test boundary ensures the letdown line is isolated.

Plant Procedure, RF-003-002

RF-003-002, Steam Generator Primary Side Services (Revision 1 Change 1)

Description of Change

This change expanded nozzle dam restrictions and added steps to ensure Steam Generator manway nuts are secured. The changes to the procedures were the result of a calculation revision. Calculation EC-M88-012 provides the basis for limitations on when all cold leg nozzle dams can be installed in the RCS to ensure reactor safety in the event of a loss of SDC. The limitations were revised based on lower relative decay heats compared to the previous decay heat model. The calculation establishes that core uncovery will occur two hours after a loss of SDC for the assumed RCS configuration if the reactor has been shutdown for fifteen days. The prior calculation required twenty-four days after shutdown for the same two hour core uncovery time. The basis for two hours is the conservative assumption that containment can be closed in less than two hours (unchanged from previous revisions). The requirement to keep containment closed for at least four days after reactor shutdown is maintained by reducing the time assumed to be required for containment closure from two hours to less than or equal to 1.75 hours.

Reason for Change

The procedure changes resulted from revisions to a calculation which refined and improved the decay heat curve. This refinement allowed for changes to the specific limitations referenced in the procedures.

Safety Evaluation

Case 1: Four Day Criteria After Reactor Shutdown

The change in the assumed time required to close reactor containment from 2.0 hours to 1.75 hours will not affect the probability of any accidents. This will have no effect upon the probability of losing SDC or of any other accident.

The assumed time for closing reactor containment is being reduced from 2.0 hours to 1.75 hours. This is acceptable since the actual time to close containment is estimated to be 1.0 to 1.5 hours after a loss of SDC. Thus, conservatism is being reduced in the time assumed to complete this task. This will not increase the consequences of losing SDC, as a closure time of less than or equal to 1.75 hours will ensure containment closure within the required time. Containment can be closed in the allowed time period and there is sufficient time available for operators to respond to a loss of SDC (for example, by establishing HPSI flow to the RCS). This change cannot affect the consequences or probability of any other event.

The change in the assumed time for closing reactor containment will not create the possibility of an accident different than any already evaluated. The containment will be closed within the required 1.75 hour time after a loss of SDC, thus, ensuring that containment will be closed prior to when core uncovery could occur assuming no makeup is added to the reactor. There is no physical change to the plant, and no change to how it will be operated in shutdown conditions. The time required to close containment in response to a loss of SDC cannot affect the possibility of occurrence of any other event.

The change in the assumed time for closing reactor containment will not affect the probabilities of equipment malfunction. This change will not affect the method of equipment operation in response to a loss of SDC, thus, can have no affect upon equipment malfunction probabilities.

Because these changes will ensure that containment is closed prior to when core uncovery would occur after a loss of SDC, the consequences of malfunction of equipment important to safety will be unaffected and the possibility of malfunction is not created.

These changes will ensure that containment is closed within 1.75 hours after a loss of SDC, which is a sufficient and conservative period of time to complete containment closure. This is a reduction in the conservatism assumed in the time required to close containment. This is acceptable since the actual time to close containment is estimated to be between 1.0 to 1.5 hours after losing SDC. The requirement to close containment in less than or equal to 1.75 hours will ensure containment closure within the required time duration. Thus, this change does not affect the margin of safety since containment will be closed prior to core uncovery in the event of prolonged loss of SDC event of a prolonged loss of SDC.

Case 2: Installation of all Cold Leg Dams

The procedure changes associated with EC-M88-012-R02 concerning limitations when all cold leg nozzle dams are installed do not affect accident probabilities. The calculated time to core uncovery after a prolonged loss of SDC has no effect upon the probability of losing SDC. The time interval to take action to ensure containment closure (two hours) is unchanged. Accident consequences are not affected because, due to the limitations and required actions, containment will have been closed prior to core uncovery after a loss of SDC.

There are no changes to the plant as a result of the revised calculation or procedure changes. The changes concern the time required to respond to a previously evaluated accident (loss of SDC) and does not create a possibility of a different type accident than previously evaluated. The changes are based on the calculation of the time window in which recovery actions must be performed and reactor containment must be closed after a malfunction has caused a loss of

SDC. There is no change in equipment malfunction probabilities associated with the calculation. Because the changes incorporate the conservative assumption that it takes two hours to close reactor containment, there is no change to the consequences of equipment malfunction and no possibility of a malfunction different than already evaluated. Reactor safety is maintained and the margin of safety is unchanged since containment will be closed prior to core uncovery.

Plant Procedure, UNT-005-013

UNT-005-013, Fire Protection Program (Revision 1 Change 1)

Description of Change

This change to UNT-005-013 allows the performance of surveillances other than operations monthly surveillance without performing an unnecessary tour of the annulus.

Reason for Change

The purpose for this change is to eliminate unnecessary tours of the annulus.

Safety Evaluation

This change to expand the circumstances addressed in the compensatory action required for the annulus detection system is recognized to be an operational clarification and facilitates the operation and testing of another safety system (Shield Building Ventilation System). This will have no effect on the probability of occurrence of an accident as previously calculated in the FSAR.

Because the annulus detection system compensatory action requirements presently address an approved exception during monthly Shield Building Ventilation System testing required by TSs, this change is recognized to provide additional clarification for similar testing that is or may be conducted under established plant administrative and test control programs. As such it does not increase the consequences of an accident previously evaluated in the FSAR.

The detection system in question provides early warning fire detection for the annulus which is essentially void of combustibles. Due to the lack of combustibles and the general inaccessibility of this area, thereby, limiting transient combustibles, there exists no significant probability or increase in the probability of a malfunction of equipment important to safety as previously evaluated in the FSAR.

This change is recognized to be of a clarifying nature and expands on activities previously endorsed under the approved Fire Protection Program. It involves no physical impact or interface with installed systems important to safety and results in no increase to the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

This fire detection equipment is intended for the purpose of detecting a fire in the incipient stage to provide sufficient time to effect manual fire fighting efforts and does not add, delete, or impact the operation of other systems important to safety. Thus, the change described in the compensatory measure does not create the possibility for an accident of a different type than any previously evaluated in the FSAR.

Since this change is of a clarifying nature and does not physically affect equipment outside the Fire Protection System, the expanded compensatory action limitations designed to encompass all operational activities does not create the possibility for a malfunction of a different type than any previously evaluated in the FSAR.

Fire protection components were deleted from the plant TSs as part of Amendment 50 to the facility operating license. As such, this change to the administrative procedure does not reduce safety margins as defined in the basis for any TS.

Plant Procedure, UNT-005-013

UNT-005-013, Fire Protection Program Change -Incorporate +10% Tolerance for Pump Performance (Revision 1, Change 2)

Description of Change

This is a change in the Fire Protection Program which expands the acceptance tolerance for pump discharge head and flow performance to that currently endorsed in ASME, OM-6. The range is expanded from +3% / -10% to ± 10%.

Reason For Change

The change is intended to update the pump testing tolerance to current ASME, OM-6 standard.

Safety Evaluation

A fire event would not be impacted by this change. Fire pump performance, and ultimately water system performance would not be adversely affected. A fire event would be mitigated in a manner consistent to that previously assumed in the FSAR. This tolerance change poses no reduction in the required minimum flow requirements of the pumps. The plant fire pumps are maintained in accordance with accepted standards to ensure their operation during a fire. The tolerance difference does not alter any of the required maintenance. During a fire event, pump performance would remain at or above the minimum levels established in the FSAR. This assures adequate and consistent mitigating effects as described in the FSAR. This is a change to the fire pump test criteria, the consequences of a fire pump failure would not be affected. There are no new equipment interactions introduced. This change merely affects the operating criteria and provides for an increase of limit in a conservative direction. There are no physical changes and no new malfunction possibilities. The margin of safety is not changed or reduced.

Plant Procedure, UNT-006-013

UNT-006-013, Emergency Core Cooling System (ECCS) Outage Report Deletion (Revision 2)

Description of Change

This revision is to UNT-006-013 deleted the procedure.

Reason for Change

This procedure is no longer required to fulfill TMI Action Item II.K.3.17.

Safety Evaluation

This is an administrative change to the reportability requirements as stated in FSAR Section 1.9. The NRC indicated by letter dated May 5, 1989 that the requirements of 10CFR50.72 and 50.73 and the industries efforts to report on the nuclear plant reliability data system (NPRDS) were adequate for reporting ECCS outages. This administrative change would have no affect on accident or equipment malfunction considerations or the margin of safety. This evaluation documents, in accordance with procedural requirements, a 10CFR50.59 evaluation for a change in procedures described in the FSAR.

SPECIAL EVALUATIONS

Condition Identification 108918

CI 108918, Erecting Scaffolding by Valve CC-822A

Description of Change

CI / Work Authorization CI 018918 / WA 01039430 erects scaffolding near CCW valve CC-822A.

Reason for Change

The work authorization erected scaffolding by CCW valve CC-822A.

Safety Evaluation

Erecting scaffolding by vaive CC-822A does not increase the probability of an accident. The scaffolding would only damage the actuator which is fail-as-is in the open position. This is the safety position for the valve. This valve provides cooling water to containment fan cooler 3A-SA. Only three of the four containment cooling fans are required. If the scaffolding resulted in damage to the actuator and a stem leak developed, this would be contained on the -4' el. RAB. The valve could be isolated downstream. The CCW effluent is not highly contaminated since it is a closed loop system.

The possibility of an accident which is different than any already evaluated in the FSAR will not be created. For post LOCA conditions only one containment fan is required to be operable for each train. Since only the actuator would be damaged, the CCW to the fan cooler would still be operable. The fail-as-is condition for the valve would assure flow through the normally open valve.

The probability or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because the valve will fail-as-is in the open position, thereby assuring cooling water to the containment fan. This is the safe position for the valve which also assures that no new possibility of malfunction is created.

The margin of safety as defined in the basis to any TS will not be reduced because only one containment fan is required to be operable per train.

Condition Identification 261175

CI 261175, Pressurizer Spray Valve (RC-301A) Repair

Description of Change

The intent of this CI is to repair the reactor coolant pressurizer spray va'.e, RC-301A, by building up the area on the valve body that machine threads into the weld. This will allow the use of a replacement seat and is reflective of the current design.

Reason for Change

The pressurizer spray valve, RC-301A, was inspected during refueling outage 3 following fuel cycle leakage problems. The valve was found to have threads stripped on both the body and seat. The cause was concluded to be a lack of a seat ring locking mechanism.

Safety Evaluation

The accident probability has been reduced because the repai. will be as strong as the original and design improvements will be ins. lled. The only area of concern is the weld area (threads) which, even if it disappeared, would not affect the attachment of the seat to the body. The new extended liner design would retain the seat even without the body threads. Only minor seat leakage would result in this eventuality. The incorporation of an acceptable weld repair in combination with the extended liner will enhance the previous configuration. Hence, there are no accident possibilities created other than those already evaluated.

The repair and design enhancements have reduced the probability of malfunction of the valve. In the worse case (missing weld) the liner would retain the seat but minor leakage may occur. The valve leakage has been previously evaluated. No different malfunction will be created by an acceptable weld repair. The safety margin is not affected because the valve will be repaired per the code to the original design and with incorporation of design enhancements of DCN-ME-113, the valve will be more reliable.

Condition Identification 263002

CI 263002, Decontamination Ultrasonic Control Panel

Description of Change

CI 263002 documents removal of an ultrasonic generator which was part of the original plant design. This component and associated equipment was originally intended to decontaminate items immediately after exiting the containment. The unit consisted of the ultrasonic generator, one ultrasonic tank and two rinse tanks. The three tanks were removed prior to startup and are not currently installed as shown on plant drawings. The ultrasonic generator is installed as shown on the drawings.

Reason For Change

The ultrasonic decontamination system is not in use and will not be placed in service due to the introduction of new liquid abrasive decontamination equipment.

Safety Evaluation

The decontamination equipment performs no safety-related function. Removal in accordance with the plant modification program assures that the plant configuration will not be adversely affected. Thus, the probability of an accident will not be increased and the possibility of an accident different than evaluated is not created. The equipment was intended to provide a means of minimizing levels of exposures to personnel during outage activities. This objective is met by an alternative design. The equipment provides no preventive or mitigative capability for design basis events, therefore, the consequences of any accident condition will not be increased. Since the equipment is being properly removed and no new system interfaces are created, equipment malfunctions considerations are not impacted. There are no TSs or bases related to this equipment which could affect the margin of safety.

Condition Identification 264455

CI 264455, Steam Generator Moisture Carryover and Feedwater Flow Special Test

Description of Change

This is a special test to measure the moisture carryover exiting the steam generators and the feedwater flow rates. A lithium-hydroxide tracer solution is injected into the main feedwater line and sampled at steam generator blowdown.

Reason for Change

This is a special test to measure the moisture carryover exiting the steam generators and the feedwater flow rates.

Safety Evaluation

All equipment utilized in conducting the special test are part of the Steam and Power Conversion System and are not safety-related. Failure of any equipment utilized during the test would affect no safety-related systems. The lithium-hydroxide material was verified to be safe for the intended use. The probability or consequences of an accident previously evaluated in the FSAR will not be increased. The worst case scenario would be failure of the sample point lines. The volume of feedwater discharge would be minimal and power levels would remain conservative. No safety-related equipment would be affected. Therefore, the probability or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The margin of safety is unaffected by the test.

Condition Identification 267931

CI 267931, Control Element Drive Mechanism (CEDM) Cooling Unit E-16(3D) Motor

Description of Change

CI 267931 documents replacement of control element drive mechanism (CEDM) cooling unit E-16(3D) motor with a GE motor instead of a Westinghouse motor.

Reason For Change

The motor is being replaced with a GE motor because an equivalent Westinghouse motor cannot be obtained in a timely manner. The new GE motor performs the intended function of the original motor with some differences in performance parameters.

Safety Evaluation

The motor replacement is associated with a non-safety-related system which is not required for safe shutdown nor for any support functions for accident scenarios analyzed in the FSAR. Hence, there is no impact on accidents evaluated in the FSAR. The replacement motor is required to retain structural integrity during and after a seismic event but does not have to retain operability. Analysis concluded that the motor will retain its structural integrity, and will not become a missile and damage other equipment required for radiological release control. The replacement motor has minor deficiencies (e.g., lack of space heaters) and variations (e.g., insulation class). These differences have been evaluated and have the potential to impact reliability but will not impact any equipment important to safety. The CEDM cooling system does not support any safety-related equipment and, therefore is not considered important to safety. There are no new interactions or connections associated with this replacement that would impact a protective boundary. The failure modes of the new motor are similar to the original motor. There is no impact on any margin of safety.

Condition Identification 268414

CI 268414, H_2 Recombiner 'A' Active Filter Replacement

Description of Change

CI 268414 documents replacement of H_2 Recombiner 'A' Active Filter which requires an increase in the mounting bolt size. The change also requires that the mounting holes be drilled and tapped to a larger size.

Reason For Change

The active filter on the power supply panel for Hydrogen Recombiner 'A' was originally attached to the panel with bolts. Due to maintenance problems this filter was removed. Because of the configuration of the power supply panel, there is no convenient access to the back of the panel. Therefore, the bolts cannot be replaced. The bolts were replaced with screws instead of bolts.

Safety Evaluation

The change involves modifying the structural restraints of a component in the H_2 Recombiner 'A' active filter system. This system mitigates the consequences of a LOCA. Its malfunction does not increase the probability of occurrence of this or any other accidents previously evaluated in the FSAR. This change as documented will not compromise the structural integrity of the system. The radiological release consequences of any evaluated accidents will not be affected. The proposed change does not introduce any new system interactions or connections. Since the structural integrity is not compromised, this change will not adversely affect any equipment important to safety or increase the consequences of equipment malfunctions. The possibility of a malfunction of a different type is not created. Additionally, no new accident possibilities are created. The equipment will function as designed with no reduction in capabilities, therefore, the margin of safety is not reduced.

Special Issue

Cycle 4 Reload

Description of Change

This Cycle 4 reload consists of the replacement of eighty-four fuel assemblies, as described in the Reload Analysis Report submitted by Combustion Engineering. The Cycle 4 core will be composed of 84 fresh batch F assemblies, eighty-four batch E assemblies initially inserted in Cycle 3, forty-eight batch B assemblies initially inserted in Cycle 2 and 1 batch C assembly which was discharged at the end of Cycle 2 after being burned for two cycles. Forty-one Batch C assemblies which have been used for Three cycles and Forty-four batch D assemblies which have been used for two cycles will be discharged to the spent fuel pool. The reload batch will consist of sixteen type F0 assemblies (no poison rods), twenty type F1 assemblies with 8 burnable poison rods per assembly, and 48 type F2 assemblies with sixteen burnable poison rods per assembly. The core will be loaded with quarter core rotational symmetry.

Specific changes from the reference cycle include:

Burnup for some fuel rods may exceed the 52,000 MWD/T peak rod burnup value discussed in the CE High Burnup Topical Report approved by the NRC.

Fuel Assembly Mechanical Design Changes - The batch F poison rod overall length has been increased by 0.25 inches and the locking discs that are used in the lower end fitting connection to the guide tube as an antirotation device have been redesigned.

Higher fuel enrichment - batch F assemblies will contain fuel rods with a maximum nominal 4.05 weight % U235 enrichment.

Shoulder Gap - The maximum fuel rod fluence affects the rod and guide tube growth rate.

Flatter power distribution - Cycle 4 will have a flatter power distribution (pin census).

A new CEA drop time curve, based on measured plant data, was used.

Reason for Change

This safety evaluation was performed for the Cycle 4 fuel reload.

Safety Evaluation

There are no accidents evaluated in the FSAR that are initiated by the reactor core. Accidents are initiated by equipment malfunctions that then affect the fuel. The fuel itself has no effect on the likelihood of occurrence of an accident.

As documented in the Cycle 4 Reload Analysis Report, the consequences for all previously evaluated accidents remain bounded by the reference cycle analyses and within NRC acceptance limits. This change does not increase the consequence of any accident previously evaluated in the FSAR. CE reviewed all accident analyses as part of the Cycle 4 reload analysis. In most cases, comparison of key input parameters between Cycle 4 and the reference cycle determined that the Cycle 4 input was bounded by the reference cycle input and no reanalysis was required. The analysis input accounts for each of the specific changes identified earlier. Reanalyses of the following events were required to determine if, for Cycle 4, they would still be bounded by the reference analyses:

1. Pre-Trip Steam Line Break
2. CEA Ejection and PDIL Verification
3. Excess Load with Loss of AC power
4. Loss of Flow Accident
5. Sheared Shaft / Seized Rotor
6. Subcritical and Low Power CEA Withdrawal

There are no new system interactions or connections associated with core reload. The minor mechanical design changes do not affect the performance of the fuel assemblies. The growth of the longer poison rod is bounded by fuel growth. The redesigned locking discs meet all existing design and interface requirements for fuel assemblies. Therefore, operation of Waterford 3 with the Cycle 4 reload core will not cause an accident of a different type than any previously evaluated in the FSAR.

All equipment important to safety will function in the same manner with the reload core as with the previous core. There is no characteristic of the Cycle 4 different from the cores of previous cycles which would tend to increase the probability of a malfunction of equipment important to safety. The mechanical design changes do not affect the performance of the fuel assembly as discussed above. Therefore, the consequences of equipment malfunction are not increased. All core assemblies were reviewed for shoulder gap clearance, with the result that there is sufficient shoulder gap margin for growth of all fuel rods at the anticipated end of cycle fluence.

All equipment important to safety will function in the same manner with the reload core as with the previous core. Therefore, the consequences of equipment malfunction are not increased. Installation of a reload core cannot cause the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR. Equipment important to safety will function in the same manner with a reload core as with the previous core. the change in core characteristics does not change any parameter that would affect the function of equipment important to safety. There are no new methods of failure associated with any of the changes identified previously for the Cycle 4 reload.

The new fuel assemblies use an identical design to the existing assemblies with the minor exceptions noted previously. These changes do not affect the fuel boundary. The peak fuel burnup for a small number of fuel rods may exceed the 52,000 MWD/T burnup value discussed in the CE High Burnup Topical Report, CENPD-269. The mechanical design analysis shows that the fuel performance parameters for these rods are within the appropriate design criteria. The physics data input to the Cycle 4 safety analysis, which treat fuel exposure explicitly, show that the power level of these high burnup fuel rods is low, and therefore not limiting. These fuel rods are in assemblies from batch D, which has a maximum batch average burnup of 43,760 MWD/T (including uncertainties). This is below the 45,000 MWD/T batch average limit for which the analysis methodology has been approved by the NRC.

All accidents have been shown to have consequences bounded by the reference cycle and below the appropriate NRC acceptance limits. Therefore, there is no reduction in any margin of safety.

Special Issue

Devitalization of Emergency Diesel Generator "B" Room

Description of Change

Access points to EDG "B" room, Door-23 and Door-24, were devitalized while in modes 5 and 6.

Reason for Change

The access points were devitalized to facilitate testing and maintenance activities during plant outage.

Safety Evaluation

Only one EDG is required in modes 5 and 6. EDG "A" will be operable along with its associated electrical train. Therefore, devitalizing EDG "B" will not increase the probability or consequences of an accident nor will this action create the possibility of a different type accident. Equipment malfunction considerations are similarly unaffected. The margin of safety as defined in the basis for TSs is not reduced because only one electrical train and its associated EDG are required to be operable.

Special Issue

Devitalization of MSIV "A" and "B" Wing Areas

Description of Change

The MSIV "A and B" wing areas were devitalized.

Reason for Change

These areas were devitalized to facilitate testing and maintenance activities during plant outage.

Safety Evaluation

In mode 5 and 6 with temperature less than 200°F, the main steam system is not required to maintain reactor integrity. The flow elements, safety relief valves, atmospheric dump valves (ADVs), isolation valves for steam supply to emergency feedwater pump turbine and MSIV are not applicable. TS 3.4.4c mentions penetration providing direct access from containment atmosphere to outside atmosphere be closed. However, if MSIVs are taken apart, it is still a closed loop. If the steam generator is opened, a guard should be posted in containment or wing areas. Therefore, the probability of an accident previously evaluated in the FSAR will not be increased.

This involves a secondary system. Since no reactor fluid is flowing in modes 5 and 6, there is no chance for a primary to secondary leak. Therefore, the consequences of an accident previously evaluated in the FSAR will not be increased.

In modes 5 and 6, secondary main steam system is not required to be operable. The main steam system in the wing areas does not create the possibility of an account which is different. Therefore, the possibility of an accident which is different than any already evaluated in the FSAR will not be created.

The main steam equipment in this area consisting of flow elements safety relief valves, ADVs, isolation valves for steam to emergency feedwater pump turbine and MSIV are not required to provide any safety function in modes 5 and 6. As such, the probability of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

In modes 5 and 6, no reactor fluid is flowing. Consequently, there in essentially no concern for a primary to secondary leak occurring and the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

In modes 5 and 6, the main steam equipment consisting of flow elements, safety relief valves, ADVs, and isolation valves for steam to emergency feedwater pump turbine and MSIVs have no function in the shut down of the plant or maintaining the safe shutdown of the plant. Therefore, the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR will not be created.

Section 3/4.a.4 of the TSs was reviewed and even though the MSIVs may be removed, a closed system still exists to steam generators. If the steam generators are opened, a guard should be posted in the containment wing area. As such, the margin of safety as defined in the basis of any TS will not be reduced.

Special Issue

Devitalization of Turbine Generator Building Roof (+67' elevation)

Description of Change

The turbine generator building roof area (+67' elevation, access points D-180 and D-181) was removed from vital area requirements.

Reason for Change

These areas were devitalized to facilitate maintenance and modification activities during plant outage.

Safety Evaluation

The accident, equipment malfunction and margin of safety considerations will not be affected because access will be controlled to the RAB roof at the fence.

Special Issue

LP&L-400-001, UGS Stand Installation and Removal Procedure, Waterford 3 (Revision 0)

LP&L-400-006, FAP Modification Equipment Dry Setup and Checkout Procedure, Waterford 3 (Revision 0)

LP&L-400-007, FAP Modification Equipment Wet Setup and Checkout Procedure, Waterford 3 (Revision 0)

STD-NSS-090, Procedure for the Electrical Discharge Machining of the Fuel Alignment Plate (Revision 4)

STD-NSS-091, X-Y Table, TDS and Elevator Operation Procedure for Fuel Alignment Plate Modification (Revision 4)

STD-NSS-093, Fuel Alignment Plate Flow Restrictor Plug Insertion and Checkout Procedure (Revision 4)

STD-NSS-095, Fuel Alignment Plate Nozzle Hole Gauging Operation and Tool Check-out Procedure (Revision 4)

Description of Change

The above listed procedures were utilized to control activities for the fuel alignment plate modifications under Design Control Package 3097. This activity was evaluated and included in this report under the design control package section. The checkout and operational phases of these procedures could cause the addition of a non-borated solution to the refueling cavity during refueling operations.

Reason for Change

This safety evaluation is to justify that the addition of a non-borated solution will not affect refueling cavity boron concentration.

Safety Evaluation

The equipment described in these procedures require the use of approximately thirty-five gallons of hydraulic fluid which could leak into the refueling cavity during refueling. The fluids are basically non-hazardous and are 100% soluble in water. The addition of this small amount of non-borated solution will not affect a boron dilution accident. The procedures will only be used during modes 5 and 6 and will not create any new accident possibilities. Work will be on the upper guide structure that is removed from the reactor vessel. The modifications have been previously evaluated by DCP-3097.

The movement of the equipment associated with these procedures will not affect safety-related equipment since the procedure for control of heavy loads will be used. There are no changes to the FSAR or TSs required by these procedures. The non-metallic materials (e.g.hydraulic fluid) have been approved by CE (NSSS vendor) for use in contact with NSSS surfaces.

Special Issue

Operation of the DWST and ACCW System with Xenon Contamination

Description of Change

This activity will allow operation of the Demineralized Water Storage Tank (DWST) and the ACCW system with xenon contamination and allow the xenon to vent to the atmosphere. The radioactive gas will be released through the DWST relief valve and the wet cooling tower which is not a normal radioactive release pathway.

Reason for Change

Plant operations require the use of the ACCW system and the DWST. The water in these systems is contaminated with low levels of radioactive Xenon-133 gas.

Safety Evaluation

The DWST and ACCW system will continue to function as designed; therefore this activity does not change the probability of any accident from occurring. This activity does not increase the consequences of any accident. Table 11.3-8 of the FSAR lists the average annual airborne concentration of Xenon-133 as 2.41 E-10 microcurie per cubic centimeter (cc). The maximum concentration of Xe-133 from this activity will be 4.5 E-12 microcurie per cc. This level of activity is insignificant when compared to the average Xe-133 activity. No equipment important to safety is affected because the Xe-133 will not change the operating characteristics of the system or affect any equipment important to safety. This activity does not create any new system interactions or connections; therefore, the possibility of a new accident of a different type than those previously evaluated is not created. This activity will not have any affect on the equipment. The ACCW system and the DWST will continue to operate and function as previously described. This activity does not reduce the margin of safety. The radioactivity discharged is well below the average radioactivity identified in the FSAR and the ACCW systems and DWST will continue to operate and function as previously designed.

The dose commitment from the release has been calculated to be 3.79E-6 mRad gamma dose and 1.13E-5 mRad beta dose. The quarterly gamma dose limit is 5 mRad and the beta dose limit is 10 mRad. This release is 7.58 E-5% of the gamma dose limit and 1.13 E-4% of the beta dose limit. From all release for the current quarter Waterford 3 is at 5.2% of the beta dose limit and 3.8% of the gamma dose limit, so this release will be a small fraction of normal effluents.

Special Issue

LP&L Organization Changes

Description of Change

This change involves a reorganization of the Waterford 3 upper management.

Reason for Change

Louisiana Power and Light (LP&L) is a wholly-owned subsidiary of Entergy Corporation. On August 15, 1989, LP&L submitted to the NRC proposed license amendments to designate Entergy Operations, Inc., a new subsidiary of Entergy, as the licensed operator for Waterford 3. This organizational change is a result of efforts by Entergy to have the management of Waterford 3 consolidated with that of the other Entergy-owned nuclear plants.

Safety Evaluation

The probability, consequences, and possibility of a malfunction different than previously evaluated cannot be increased by the proposed change because the change would not affect the function or operation of any structure, system or component at Waterford 3. Specifically, as a result of the proposed change, there will be no physical changes to the facility, and all limiting conditions for operation, limiting safety system settings, and safety limits specified in the plant's TSs will remain unchanged. Plant operating and emergency procedures will also not be changed in any substantive way. The only changes made to any internal documents and procedures would be, if needed, administrative changes to reflect the revised organizational management responsibilities.

The proposed organizational changes will not diminish management effectiveness. The organization continues to be based on clear lines of authority and responsibility, and continues to be based on clear lines of authority and responsibility, and continues to meet the general guidelines as established in Waterford 3 TS 6.2.1. Moreover, the proposed change does not affect the technical qualifications of the onsite operating organization. Finally, by adding experience and increased management attention, and by achieving some of the benefits of system-wide nuclear consolidation, the proposed change should actually enhance the qualifications of the offsite organization.

The proposed change also will not increase the consequences of any accident or malfunction previously evaluated. Because plant structures, systems, and components are unchanged, there can be no change to plant response to analyzed events. Further, because there are no substantive changes to operating or emergency procedures, there can be no increase in consequences of any accident or malfunction.

As indicated above, all limiting conditions for operation, limiting safety system settings, and the safety limits will remain unchanged following implementation of the proposed organizational changes. In addition, plant operating and EOPs will not be affected in any substantive way. As such, the plant conditions for which the design basis accident analyses have been performed will remain valid. The design and design bases of Waterford 3 will remain the same. Therefore, the current plant safety analyses remain complete and accurate in addressing the relevant licensing basis events and in analyzing plant response and consequences. The proposed change does not create the possibility of a new or different kind of accident or malfunction from those previously evaluated.

Plant TSs ensure that the plant operates in a manner that will ensure acceptable levels of protection of public health and safety. The margins of safety that provide the basis for TSs are based upon the licensee's safety analysis report, the NRC's safety evaluation report, and other licensing basis documentation. These margins relate to NRC acceptance criteria for physical parameters that define the performance of the fission product barriers (i.e., fuel cladding, RCS boundary and containment).

Because the proposed organizational changes involve no changes to the physical design or to the operation of the plant, there will be no change to any of the relevant margins of safety. The proposed change, therefore, cannot involve a reduction on a margin of safety previously established.

Special Issue

Pump and Valve Inservice Test Plan (Revision 6 Change 1)

Description of Change

This change incorporates the same changes contained in Revision 5 Change 1. The safety evaluation for Revision 5 Change 1 was reviewed and is still valid. Two items that require evaluation are the deletion of Relief Request 3.1.43 and 3.1.56.

Reason for Change

Relief Request 3.1.43 was submitted when TSs required all containment fan coolers (CFC) to be operable in modes 1-4. Amendment 39 to TSs no longer requires all CFCs to be operable, and thus the basis for this relief request is no longer applicable. The CCW valves to the containment fan coolers are tested quarterly in accordance with ASME Boiler and Pressure Vessel Code Section XI.

Relief Request 3.1.56 meets the requirements for NRC approval per Generic Letter 89-04. The area of code deviation concerns increased frequency testing criteria. By establishing allowable time limits, a more conservative approach is utilized to increase the monitoring frequency of potentially degraded valves.

Safety Evaluation

This change does not affect the design or operation of safety-related equipment and thus the probability of occurrence or the consequences of an accident is unaffected. Similarly, the possibility for an accident of a different type evaluated in the FSAR is not created. This change increases the monitoring frequency of CCW valves to containment fan coolers. Equipment degradation will be detected sooner, reducing the probability of occurrence of an accident. The method of testing is unchanged, thus, the design and operation of safety-related equipment is also unaffected. The consequences of a malfunction are unchanged. No new possibility of malfunction is created. This change will not result in a decreased margin of safety as the operation and design of safety-related equipment is unaffected.

Special Issue

Waterford 3 Pump and Valve Inservice Test Plan (Revision 6 Change 2)

Description of Change

Two manual gate valves (ACC-116A and ACC-116B) and two check valves (ACC-114A and ACC-114B) are being added to the scope of testing. Two relief requests, 3.1.57 and 3.1.58, have been written to address testing of these four valves during refueling outages when the lines can be drained, flushed and refilled with water from the condensate storage pool.

Reason for Change

It is undesirable to test these valves while the plant is in power operation or in cold shutdown due to the probability of chemical contamination of the Emergency Feedwater (EFW) System.

Safety Evaluation

Testing two additional manual gate valves and two check valves in mode 6 has no effect on any assumed accidents. ACCW and EFW are not required in mode 6. Testing the valves will ensure that the consequences of an accident are no worse than originally assumed. The probability of equipment (valve) malfunction is reduced and the consequences of equipment malfunction remains unchanged. Additional testing reduces the possibility of a malfunction. Testing the valves does not introduce any new accident scenarios. The margin of safety will actually be increased because of the additional testing.

Special Issue

SPEER 88-462, Spare Parts Equivalency Evaluation Report (SPEER) Emergency Diesel Generator Thermocouple Box Replacement (JB-P) (Revision 1)

Description of Change

This evaluation documented replacement of both EDG thermocouple junction boxes (JB-P) with larger boxes. The replacement boxes are used to terminate field thermocouple cable to the pre-wired thermocouples supplied with the EDGs.

Reason for Change

The EDG thermocouple junction boxes are being replaced with larger boxes for maintenance enhancement.

Safety Evaluation

The thermocouples housed in the new junction boxes do not perform a safety-related function. FSAR Table 3.2-1 indicates that the EDG "Alarms and Computer Interfaces" are non-nuclear safety (NNS). This indicates that the boxes are not required to function during and after a seismic event. However, FSAR Section 3.2.1 requires that NNS components in the area of safety-related equipment are to be seismically mounted. This is to preclude NNS equipment from damaging safety-related equipment during a seismic event. In addition, FSAR Table 3.2-1, Note 17, states that these components will receive 10CFR Appendix B QA during the operation phase of the plant. The boxes were purchased from the original EDG vendor under a qualified 10CFR Appendix B program. The boxes are seismically mounted and were evaluated for seismic loading on the EDG. Therefore, this change will not increase the probability of occurrence of an accident previously evaluated in the FSAR. Since the seismic event is the only accident which requires consideration for the boxes and they are seismically mounted, there is no increase in the consequences of an accident nor is a different accident than previously evaluated in the FSAR created. The thermocouples that terminate in the replacement boxes do not perform a safety-related function. They provide alarm only functions for various EDG system temperatures. Therefore, equipment important to safety previously described in the FSAR is not affected. The change did not affect the design of the thermocouple system, which does not perform a safety-related function. The only event of concern is a seismic disturbance. A potential seismic event has been previously evaluated in the FSAR. Therefore, the possibility of a malfunction of equipment important to safety of a different type than previously described in the FSAR is not possible. The replacement boxes are larger, however they are seismically mounted and do not affect the margin of safety.

Special Issue

PEIR 20000, Spent Fuel Pool Heat Load Calculation

Description of Change

The Waterford 3 FSAR Table 9.1-3 is updated as follows:

a) Design fuel pool eat exchanger heat load value is deleted from the "Operating Parameters" heading, and is added under the "Design Parameters" heading.

b) Actual operating fuel pool heat exchanger heat load is added under the "Operating Parameters" heading, and the corresponding shell and tube side outlet temperatures are revised accordingly.

These changes more accurately reflect plant design; however, there is no change to the plant design basis.

Reason for Change

Due to the eighteen month cycle fuel management scheme for cycle 2 (and beyond), the spent fuel pool heat load calculation for the FSAR has been repeated by Middle South Services. These new results are incorporated into the Waterford 3 documentation.

Safety Evaluation

There are no physical modification involved. Heat exchanger operating parameters remain within spent fuel pool cooling and CCW system design parameters. Consequently, the probability of an accident previously evaluated in FSAR will not be increased; nor will there be an increase in the probability of malfunction of equipment important to safety previously evaluated in the FSAR. Since these changes do not reflect a change to the design basis, then the consequences of a design basis accident will not be increased. As no physical modifications are involved, there are no systems altered; nor are there any accident scenarios created. The only TS that applies to the spent fuel cooling system is for water level which is not a part of this change. The basis for the CCW TS is unchanged.

Design Change Notice, DCN LC 1799

DCN LC 1799, EDG / 4.16 kV Bus - Manual Synchronization Documentation

Description of Change

Contacts, as described in PEIR 61115, are in a circuit that allows either EDG to be manually synchronized to the respective 4.16 kV safety bus. This feature is required for (1) testing, and (2) securing the EDG following restoration of offsite power subsequent to a LOOP. DCN LC 1799 authorizes removing these contacts from the circuit. PEIR 61115 requests revising FSAR Figure 8.3-1, LOU 1564-2176, LOU 5817-9403 to be consistent with as-built condition. DRNs E-8902088 and 2089 will implement this change.

Reason for Change

The original hardware change was implemented prior to plant commercial operation and therefore does not impact the current plant configuration.

Safety Evaluation

The only accident evaluated in the SAR which might be affected by this change is the LOOP due to the interface with the EDGs. The physical arrangement of the generator controls and bus tie breakers will not be modified. In addition, the operating procedure, OP-205-005 provides the necessary administrative controls to ensure that the probability cf LOOP will not be increased.

LOOP is the only accident in the SAR which could have radiological consequences altered by this activity. The operation of the EDGs will not be revised and the interface between the safety and non-safety 4.16 kV buses remains the same. Therefore, there is no effect of this change on mitigating system performance. Station blackout can occur if the diesels are unavailable during a LOOP. This condition has been analyzed in accordance with 10CFR50.63 and determined that no additional radiological release consequences will be created.

The EDGs and 4.16 kV buses are the only equipment which are affected by the change. The only function of the circuit, which is being revised, is manual synchronization to the non-safety 4.16 kV buses. This change will not increase the probability of a malfunction of the EDG or the 4.16 kV buses since the operating methods have not been altered.

This equipment is required to function for LOOP. If the EDG and the associated 4.16 kV buses were unavailable, the LOOP accident would result in a station blackout. This scenario has been partially addressed in the FSAR and a detailed evaluation performed pursuant to 10CFR50.63.

The proposed change does not create any new system interactions, connections, or modes of operation. The manual synchronization will continue to be performed as described in OP-902-005.

The removal of a bus breaker contact, which is not required for the manual synchronization of an EDG will not introduce a failure mode for the EDG that is not already evaluated in the FSAR. A failure of the 4.16 kV safety bus can occur if the procedure is not followed correctly and a single failure (in the safety to non-safety breaker) occurs simultaneously. However, the loss of one bus is already evaluated in FSAR section 8.3.1.1.1(a).

Document Revision Notice, I 8900545

DRN I 8900545, Drawing Corrections

Description of Change

DRN I8900545 R0 upgraded drawing 5817-4265 to the latest revision of the vendor drawing and deleted sample pumps from radiation monitors PRM-IRE-5700, 7050A, 7050B.

Reason For Change

The pumps were removed to improve the system performance and reliability.

Safety Evaluation

The monitors sample the CCW to provide a leak detection capability for components containing radioactivity which may have developed leakage. This system provides a surveillance function only and is fully functional following the changes. The CCW system monitors do not affect the post accident response of the plant in any way, therefore the probability of an accident or its consequences was not increased. The removal of the pumps from the CCW System radiation monitors did not contribute to any new failure modes which could have increased the possibility of an accident. Sufficient differential pressure exists across the monitors to warrant the removal of the pumps. An engineering evaluation verified that the removal of the pumps from the CCW System could be accomplished without any adverse effects on the performance of the subject monitors. Removal of the pumps decreased the number of moving parts associated with the monitors, consequently the probability or consequences of a malfunction were decreased. There were no malfunctions identified that had not been previously evaluated. The monitors are completely functional and the technical basis is unchanged, therefore no margin of safety is reduced.

Document Revision Notice, M 8800077

DRN M 8800077, Flow Diagram Boron Management System

Description of Change

This DRN revises the unique identification (UNID) numbers on five Boron Management (BM) System valves to radioactive waste management (RWM) classification. Also, one line number has a transposed number corrected.

Reason for Change

This change was made to reflect the actual plant configuration and appears on FSAR Figure 11.2-1.

Safety Evaluation

This was a documentation change only. No physical changes to the plant are accomplished. This drawing change will not affect any of the accidents or malfunctions previously evaluated nor will it reduce the margin of safety.

Document Revision Notice, M 8800419

DRN M 8800419, Update of Drawing LOU-1564 G-853 S02

Description of Change

This change merely updates the above drawing by identifying valve CAP-205 (2HV-B155A) which previously was unidentified.

Reason for Change

Drawing previously had no identification for this valve.

Safety Evaluation

This change does not represent any kind of physical change made to the plant. This is a paper change only; does not change the operation of the plant. All procedures remain unchanged. As such, no unresolved safety question is associated with this change.

Document Revision Notice M 8900187

DRN M 8900187, Chilled Water System Low Point Drains

Description of Change

The flow diagram and FSAR Figure 9.2-8 sheet 1, for the Chilled Water (CHW) System are to be revised to depict the drains located at system low points.

Reason for Change

The flow diagram is being revised to reflect as-built conditions. The changes meet the original design intent of having the drains located at the system low points and is merely reflecting actual plant configuration.

Safety Evaluation

This revision to the flow diagram will show the drains in the actual location relative to other system components, in accordance with as-built conditions. The drains are located at the system low points, which is in accordance with the original design intent. Only the flow diagrams are being revised and no changes are being made to the system; therefore, there are no accidents affected by this change.

The normally closed and capped drains exist upstream and downstream of the CHW lines which provide cooling water to the AH-2 air handling coils in the safeguard pump room. These drains have been located at the system low points to allow for maintenance. The flow diagrams showed these drains incorrectly located relative to other system components. The revision to these drawings will allow for the correct schematic representation of the system without affecting the system in any way. The CHW system and associated cooling coils are required to function under all accident conditions. The normally closed drain lines serve no function other than facilitating maintenance and the correction of the flow diagrams to reflect their actual location will have no consequences relating to the system operation or equipment malfunction. There are no new system interaction resulting from this change and no new types of accidents that could result. The drain lines are closed and capped and serve no active or passive function other than to prevent leakage when closed during normal operation or during accident conditions.

This correction to the relative drain locations on the flow diagram is not a physical change to the system, and as the drain lines serve only a maintenance function, this change does not involve or relate to protective boundaries or margins of safety in any way.

Document Revision Notice, M 8900272

DRN M 8900272, Update the Vent and Drain Locations on the CCW Heat Exchanger

Description of Change

DRN M 8900272 is issued to update the vent and drain locations on the CCW heat exchanger. This is a drawing change only, the existing configuration remains in accordance with the original design.

Reason for Change

This revision will allow the drawing to match the as-built condition and conform to system design and intent.

Safety Evaluation

There is no change to the function or operation of the system. The drawing will now reflect the as-built condition which conforms to the original design. There are no accident or equipment malfunction considerations altered by this change. No margins of safety are affected.

Document Revision Notice, M 8901196

DRN M 8901196, Flow Diagram CCW System Valve Position Designation when Chillers are Removed

Description of Change

This DRN documents the position of valves CC-8083A/B and CC-8244A/B as normally opened and valves CC-8246A/B and CC-8085A/B as normally closed when the chillers are removed.

Reason for Change

The valve positions were redesignated to depict the required positions for safety class break when the chillers are removed from the system.

Safety Evaluation

The existing safety class break design is required when the CCW system is in operation, however, when the chillers are removed, only one manual, administratively closed valve is required. The probability of an accident previously evaluated is not increased. This DRN provides notation on the flow diagram as to the required valve position to maintain safety class break requirements and prevent entrapment of fluid between the two valves.

The consequences of an accident previously evaluated is not increased. This DRN does not alter plant configuration, therefore, postulated consequences of CCW loss remain the same and no accident different than already evaluated will be created.

The probability or consequences of malfunction of equipment important to safety previously evaluated is not increased. This DRN provides notation on the flow diagram as to the required valve position to maintain safety class break requirements. This DRN does not alter the actual plant configuration, therefore, postulated consequences of CCW loss remain the same. Similarly no different malfunction than previously evaluated is created.

The margin of safety will not be reduced since the system failure is unchanged.

Document Revision Notice, M 8904202

DRN M 8904202, Flow Diagram Sampling System

Description of Change

This change is to a flow diagram contained in the FSAR as Figure 9.3-2. Nineteen sample taps are rearranged to support installation of an on-line chromatograph.

Reason for Change

This DRN documents a change in the use of sample taps for support of an on-line chromatograph.

Safety Evaluation

The probability of an accident will not be increased because the sample points already exist and are just being utilized differently. The consequences of an accident will not be increased because these sample points exist presently. The possibility of an accident which is different will not be created because no new function is being created. Equipment malfunction probabilities and consequences are unchanged for the same reasons. The margin of safety will not be affected because the function is the same.

This change affects the Secondary Sampling System internal tubing configuration. This system does not affect any safety-related system nor is it required for safe shutdown of the plant.

Document Revision Notice, M 8904221

DRN M 8904221, Flow Diagram Instrument Air

Description of Change

This change is to a flow diagram contained in the FSAR as Figure 9.3-1. An instrument air supply is added to support installation of an on-line chromatograph.

Reason for Change

This DRN documents a change in the use of instrument air for support of an on-line chromatograph.

Safety Evaluation

The addition of instrument air for the Dionex ion chromatograph will not increase the probability of an accident. The consequences of an accident previously evaluated will not be increased by the addition of instrument air. The possibility of an accident which is different will not be created because no new function is being created. Equipment malfunction probabilities and consequences are unchanged for the same reasons. The margin of safety will not be affected because the function is the same.

This change affects the Secondary Sampling System internal panel tubing configuration. This system does not affect any safety-related system nor is it required for safe shutdown of the plant. This is a non-seismic, non-quality and non-safety system.

Special Issue

LDCR 89-0357, Potable Water System and Secondary Access Facility

Description of Change

This change to the FSAR deals with the Potable Water System and the secondary access facility (SAF). The secondary access facility and line 8PW2-33 were never installed; therefore, they are being deleted from FSAR Fig. 9.5.1-1. Line 8PW4-37 was installed under DCN-MP-999 and is shown on Line List 5817.075B but is not depicted on Figure 9.5.1-1 and will be added to the figure.

Reason for Change

The change is being made to provide the latest plant configuration to the annual FSAR update.

Safety Evaluation

Deleting the SAF and the associated potable water line 8PW2-33 and adding potable water line 8PW2-33 to FSAR Figure 9.5.1-1 is a documentation change only to reflect current plant configuration. No accidents that may have radiological consequences will be caused or the consequences increased by the changes. No equipment important to safety is affected by the changes; therefore, there is no increase in the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR. There are no new system interaction or connections introduced by these changes and thus no possibility of an accident of a different type than previously evaluated. No new methods of failure can be created by deleting the facility and pipeline from the FSAR figure since they were not installed. Adding existing line 8PW4-37 to the figure will not create any new methods of failure since the line already exists and has been evaluated. This change will have no affect on the protective boundary, accident response, or the margin of safety since the change is related to the potable water system.

Special Issue

LDCR 90-0002, Deletion of Fire Area RAB 4 and Rezoning to Fire Area 1

Description of Change

LDCR 90-0002 revises the FSAR to show the cable vault as a fire zone of the control room. A fire in the cable vault or control room or both has the same impact (alternate shutdown via LCP-43).

Reason for Change

During penetration seal inspection and rework activities, several seals penetrating the floor of the control room were identified as inaccessible due to the existence of a steel plate on the bottom side. These seals were evaluated and it was determined that they did not meet 10CFR50 Appendix R requirements. The fire areas involved included the cable vault area and the control room. The associated circuits analysis analyzed a common fire in the control room and cable vault. All alternate shutdown circuits/controls are located outside of these areas. Thus, an unmitigated fire in either area, or both would result in alternate shutdown via backup control panel LCP-43.

Safety Evaluation

The probability of an accident previously evaluated in the FSAR will not be increased because alternate shutdown for both areas due to a common fire has been considered. In addition, the pressure boundary of the control room is not impacted. The consequences of an accident previously evaluated in the FSAR will not be increased because there are no changes to the previous evaluations. No new accident scenarios are introduced. The previous evaluation considered a total burn-up of the control room and cable vault. The worse case affects including complete loss of circuitry, hot shorts, shorts to ground, short circuits and internal shorts were considered. Analysis also considered availability of either on-site or off-site power, whichever was most limiting. Therefore, the probability or consequences of equipment malfunction will not be increased and the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR will not be created. Since compliance with 10CFR Appendix R is maintained the margin of safety is not reduced.

Special Issue

LDCR 90-0071, Update FSAR Table 3.9-10

Description of Change

License Document Change Request (LDCR) 90-0071 revises FSAR Table 3.9-10 to indicate an increase from 110 psig to 121 psig for the operating condition relief valve setpoint on CCW Code Class 2 safety relief valves 2CC-R21, 2CC-R22, 2CC-R23, 2CC-R24.

Reason for Change

LDCR 90-0071 updates FSAR Table 3.9-10 to reflect changes made by DCN-MP-981.

Safety Evaluation

DCN-MP-981 increased the setpoint for four relief valves on lines supplying containment fan coolers. The setpoint increased from 110 psig to 121 psig. That setpoint is still below the pipe design limit of 125 psig. Therefore, this change does not affect any previously analyzed accident in the FSAR.

Because the setpoint remained below the pipe design limit, no new pipe break events can be postulated to occur in the CCW system. By keeping the setpoint below the pipe design value, this change also does not affect the probability of a CCW pipe break.

This change does not affect the consequences of a containment fan cooler water supply line failure. No common mode failure can occur because all four relief valve set points stay below the design limits.

This change assures that the relief valves reseat if they were to open, the normal system pressure could have kept the relief valve open once it popped. The new setpoint ensures the valve spring can push the seat back against the static and dynamic forces present when the valve opens. Once the relief valve reduces static system pressure, the reset spring force (altered by DCN-MP-981) can easily reset the valve.

This change does not affect any limiting condition for operation, safety limit, or surveillance requirement listed in TSs. Therefore, the margin of safety remains unchanged.

Special Issue

LDCR 90-0086, Fuel Pool Heat Exchanger Setpoint

Description of Change

FSAR Table 9.2-2 and 9.1-3 are being revised to correct the temperature setpoint for the fuel pool heat exchanger.

Reason for Change

This is being done to put the FSAR in agreement with plant design documentation.

Safety Evaluation

This change revises information in the FSAR in order to make it internally consistent as well as with other plant design documents and calculations. There are no changes to any plant equipment or procedures and no accidents affected.

This change corrects the FSAR to reflect the correct fuel pool heat exchanger setpoints. The actual setpoints are not being changed for the heat exchanger and therefore there is no effect on the equipment or on any associated accidents.

The fuel pool heat exchanger is not being physically or procedurally affected by this change. This change is only to correct the FSAR to reflect the actual setpoints at which the heat exchanger maintains the fuel pool.

Accidents related to the fuel pool include the fuel handling accident, spent fuel cask drop and liquid / Gaseous Waste System leak. However, none of these accident scenarios are affected as this change merely makes corrections to the setpoints listed in the FSAR.

This change revises the FSAR to bring it into agreement with itself and with plant design calculations / documentation. There are no actual changes to the plant equipment or procedures.

This change will provide for consistency in the information on fuel pool heat exchanger setpoints in the FSAR and bring it into full agreement with plant design documentation and calculations. There are no actual changes to any plant equipment, systems or procedures and therefore no impact on any possible equipment malfunctions.

This change corrects FSAR information to agree with design documentation and does not physically affect any plant equipment, systems, or procedures and hence, does not impact any margin of safety.

Special Issue

LDCR 90-0087, CHW Setpoint

Description of Change

The FSAR is being revised to correct the temperature setpoint for the Essential Chillers from 105°F to 102°F. There are no actual changes being made to any plant equipment or procedures.

Reason for Change

This is being done to put the FSAR in agreement with plant design documentation.

Safety Evaluation

This change revises the FSAR to change the CHW setpoint from the misstated value of 105°F to the correct value of 102°F, which will bring the FSAR into agreement with actual design and operating conditions. This change will not affect any plant equipment, systems or procedures and will not impact any accident scenarios.

The correction of the setpoint in the FSAR is the only change being made. The actual setpoint remains as it is and the design documents all utilize the correct value. No physical changes are being made to any plant equipment systems or procedures and no accidents are being affected.

The essential chillers are neither affected nor changed in any way by this revision to the FSAR. The actual setpoints are correct a they are and are properly documented in the design documents.

The essential chillers are required to function under all accident conditions (LOCA, post-LOCA, and shutdown) and the setpoint of 102°F will allow for their proper functions to continue. The revision to the FSAR will correct the listed setpoint from 105°F to 102°F and thereby bring the FSAR into agreement with plant design documentation as well as the design intent of the system. There are, therefore, no increases in the consequences of any equipment malfunction resulting from this change.

This change to the FSAR involves no actual change to the essential chillers as the existing setpoint is correct as it is, and there are no system interactions which could result at all from this change, and therefore no possibilities for any new types of accidents.

The essential chillers are not physically or procedurally affected by this change as only the setpoint value given in the FSAR is being revised, and there are, therefore, no possibilities of failure or malfunctions of any different type to occur.

The misstated value for the CHW setpoint of 105°F would have provided no margin of safety for the essential chillers when accident conditions might have required the switch to ACCW cooling water, but this value for the setpoint was given nowhere else but the FSAR, as all other design documents gave the correct setpoint of 102°F. This correct setpoint, 102°F, as given by the design documents, incorporates a margin of safety in order to initiate the switch to ACCW cooling water before CCW temperature exceeds 105°F in order to guarantee the cooling capacity of the chillers will meet design requirements. This margin of safety remains unchanged as only the setpoint value in the FSAR is being revised, which, from the point of view of the FSAR, is an increase in the margin of safety.

Special Issue

LDCR 90-0182, Drain Relocation

Description of Change

The flow diagram, FSAR Figure 9.2-8 (Sheet 2) is to be revised to show the correct location of drain lines with the associated drain valves and caps.

Reason for Change

This change is to bring the drawing into conformance with the as-built conditions.

Safety Evaluation

This change only affects the flow diagram and does not make any physical or procedural changes to the plant. There are, therefore, no accidents which may be caused or affected by this change.

This change affects only the drain locations as shown on the flow diagrams. There are no changes to the plant or procedures and no resulting impact on any accidents previously evaluated.

The actual drain locations are unchanged, only the drawing is revised. Hence, there is no impact on any equipment in the system.

The system and all associated equipment are unaffected by this change as only the flow diagram is revised. There are, therefore, no accidents affected and no increase in the consequences of a malfunction.

This change is to the flow diagram only and does not make any physical changes to the plant. Therefore, there are no new system interactions or possibilities for accidents created.

As this change only brings the flow diagram into accordance with the design intent and as-built conditions, there are no new possibilities for malfunctions being created.

This change affects information on the flow diagram and FSAR figure, only. There are no actual changes to the plant or procedures and no resulting impact or any margins of safety.

Special Issue

LDCR 90-0226, Pressurizer Ambient Heat Loss Revision

Description of Change

FSAR Section 5.4.10 currently refers to the calculated pressurizer heat loss to ambient. This calculated value differs from and is a small fraction of the actual measured value. This section should, instead, reference the results of the Waterford-3 startup tests. The currently referenced pressurizer heat loss to ambient is a calculated value of 82,500 BTU/hr. The value measured during startup testing was 356,000 BTU/hr. (heat loss without spray).

Reason for Change

The change is being made to provide actual test data which is different than the calculated value currently referenced in the FSAR.

Safety Evaluation

The proposed change documents the actual heat loss of the pressurizer to containment (356,000 BTU/hr.), which is less than the assumed value of 400,000 BTU/hr. used in the determination that natural circulation conditions can be maintained with 50°F subcooling after a LOOP. Therefore, the change does not increase the probability of occurrence of an accident or increase the consequences of any accident analyzed. The documentation change does not affect the probability of occurrence or alter the consequences of equipment malfunctions important to safety. The change does not result in any system interaction or connections which did not previously exist and has no impact upon how the plant is operated. Per TS 3.4.3.1.b, the pressurizer must have at least two groups of pressurizer heaters powered form class 1E buses, each having a nominal capacity of 150 kw. From the TS bases, this requirement exists to enhance the capability to control RCS pressure and establish and maintain natural circulation. If a LOOP occurs with the single failure of one EDG to start on demand, then only one set of the heaters with a capacity of 150 kw is available. As described in the FSAR, this is a sufficient capacity to allow the RCS to be maintained at hot standby assuming a 0.5 gpm pressurizer safety valve leakage and a 400,000 BTU/hr heat loss to ambient. Because the actual measured heat loss of 356,000 BTU/hr is less than the assumed value, there is no effect on any margin of safety for the plant.

Special Issue

LDCR 90-0228, Material Upgrade to the Rotating Face Body of the Reactor Coolant Pump Seal Assembly

Description of Change

The current material, ASTM A-351 GR. CF8, for the rotating face body of the reactor coolant pump (RCP) seal assembly will be upgraded to ASTM A-182 GR.F304. Table 5.4-1 of the FSAR is being revised to reflect this change and will allow the option to use the old material or the new material.

Reason for Change

The change is being made to reflect a material upgrade which Byron Jackson has made for the rotating face body (part of the RCP seal assembly which controls leakage around the RCP shaft). The new material which is a forging, is stronger, more shock and fatigue resistant, and more durable than the current material which is a casting.

Safety Evaluation

This change is a material upgrade and will enhance the performance of the rotating face body. As such, there will be no increase in the probability of occurrence or consequences of an accident previously evaluated in the FSAR. The material upgrade is a forging and stronger than the existing material. These characteristics assure that the probability of a malfunction will not increase. The RCP seal assembly safety function is to act as an RCS pressure boundary. The consequences associated with the failure of the RCP seal assembly will remain unchanged by this material upgrade. No new system interactions or connections are being created. Failure of the RCP seal assembly is covered by a small break LOCA (FSAR 15.6.3.3.3.2). This material upgrade will only change the material that the part is being made from. The design of the part will remain unchanged and will not create any new methods of failure or reduce the margin of safety as defined in the FSAR.

Special Issue

LDCR 90-0473, Deletion of Type "C" Test Requirement for CAP1032 and CAP2032

Description of Change

This changes FSAR Table 6.2-32 to agree with current local leak rate procedure.

Reason for Change

FSAR Section 6.2.6.3 describes containment isolation valve leak tests. However, that description allows test connection valves to be administratively controlled and capped providing a double isolation barrier. Therefore, the requirement for type "C" testing for 2HV-V635 (CAP1032) and 2HV-V636 (CAP2032) is no longer necessary.

Safety Evaluation

The accidents potentially affected by this change are all those that rely on containment integrity for mitigation. This change will not effect the likelihood of an accident occurring because these test connection valves are manually operated and administratively locked closed and capped during normal operations.

The accidents potentially affected by this change are all those that rely on containment integrity for mitigation. This change will not effect the radiological release consequences because these valves are manually operated and are closed during normal operations. These valves are only opened during performance of LLRT for penetrations 10 and 11. In the unlikely event of an accident during performance of this LLRT, the valves would be locked closed and capped.

The equipment potentially affected by this change are containment penetrations 10 and 11. This change will not affect the performance of penetration 10 and 11 because only the test requirement for CAP1032 and CAP2032 is being changed. CAP1032 and CAP2032 are manual operated valves and will be locked closed and capped in the event that penetration 10 and 11 fulfill their safety function.

The accident potentially affected by this change are all those that rely on containment integrity for mitigation. This change does not affect the consequences of equipment important to safety because only the testing requirement for two normally closed manual test connection valves is being changed.

The protective boundary potentially affected by this change is containment penetration 10 and 11. CAP1032 and CAP2032 are administratively controlled in the locked closed and capped position. Therefore, the protective barrier provided by penetration 10 and 11 and the associated margin of safety is not reduced. CAP1032 and CAP2032 are manual valves with capped ends. This passive design can not affect accident response. In the unlikely event these valves are open for an LLRT and an accident occurs, the valves will be locked closed and capped. Thus, there is no impact on the margin of safety provided by this protective boundary.

Special Issue

Special Test Procedure, High Pressure Safety Injection Pump (HPSI) B Performance Data Collection

Description of Change

The test procedure will run HPSI pump B on recirculation to the RWSP through the SIT fill and drain lines.

Reason for Change

The purpose of this special test is for data collection to determine HPSI pump B operating characteristics.

Safety Evaluation

The Safety Injection (SI) System will perform as designed under accident conditions. This test will not cause the SI system to be operated outside normal SI parameters experienced during performance of existing procedures. Because the test procedure will operate the SI system in a manner consistent with normal operating procedures, the probability of an accident occurring is not increased. The test runs the HPSI pump B on recirculation through the SIT 1A fill and drain valves back to the RWSP. In the event of an accident which would generate a SIAS or CIAS, the SI system would perform as designed. After receipt of an SIAS, HPSI pump B would remain running (if already running) or would start (if it had been secured). For a SIAS, the injection valve, SI-225B, will automatically open to its predetermined throttled position and SI-303A will automatically close (isolating the recirculation path to the RWSP). For a CIAS, the SIT drain to the RWSP isolation valve SI-343 will automatically close to ensure containment integrity. The repositioning of the above valves restores the system lineup to ensure adequate ECCS design performance. For this test, no changes have been made to equipment installed in the portion of the SI system affected by this test. The design analysis assures single failure criteria for the SI system which has redundant trains to ensure SI flow from at least one train. The automatically operated components affected by this test are redundant. The failure of SI-343 to shut would be backed up by the closing of SI-303A (both valves fail shut and SIAS/CIAS actuate on the same plant parameters.) Thus performance of this test does not impede the ability or increase the consequences of the SI system to perform its design function. There are no new systems or components added to the existing system to perform this test. Therefore, the accidents previously evaluated in the SAR are adequately addressed. The TS bases for ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure. Either subsystem operating in conjunction with the SITs is capable of supplying sufficient core cooling. This test procedure ensures that SIT 1A is still available and a SIAS could realign the portion of the SI system affected by this test to ensure adequate SI flow.

Special Issue

Special Test Procedure, Fuel Pool Heat-Up Rate (WA 01043590)

Description of Change

This special test stops fuel pool pump "A" and" B" by placing their start-stop switches to the stop position. The fuel pool temperature is then recorded until either a 10°F rise on either fuel pool temperature indication or after 6 hours, whichever is shorter. At this time the Fuel Pool Cooling (FPC) System is returned to service. Calculations are then performed to obtain the fuel pool heat-up rate.

Reason for Change

The purpose of the special test is to obtain the fuel pool heat-up rate empirically in order to identify time constraints for performing fuel pool system hydrostatic tests.

Safety Evaluation

The probability of occurrence or consequences of an accident previously calculated in the FSAR will not be increased because both spent fuel pool (SFP) pumps will remain operable and the SFP temperature and level will be maintained within the constraints of the FSAR. The consequences of a loss of SFP cooling or failure of fuel in the SFP will be bounded by FSAR analysis. A source of make-up water to the SFP will be maintained.

The probability of occurrence or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. All systems will be operated normally with the exception of having both SFP pumps secured.

This test will not impact any TS since SFP water level will be maintained above the minimum required by TS.

Special Issue

TAR 89-15, Return to Service of Temperature Loop RC-IT-112CB

Description of Change

The compensation loop on TE-112CB will be replaced with a substitution resistor. The use of a fixed resistor could introduce an error of ±0.25°F to the loop.

Reason for Change

The compensation loop on TE-112CB (temperature loop RC-IT-112CB) has an intermittent ground that affects the temperature signal for loop 1 "T_{cold}" to core protection calculator "B". This temporary alteration (TA) will allow the temperature loop to function properly.

Safety Evaluation

The resistor could introduce a ±0.255°F error. This is within the uncertainty analysis. The Cycle 3 setpoints are based on a ±3.0°F total cold leg temperature uncertainty, while the currently calculated uncertainty is ±2.655°F. Therefore, an additional ±0.255°F will result in a calculated uncertainty of no more than ±2.91°F which is bounded by the current setpoints. The core protection calculator function and safety function will remain unchanged. Therefore, the probability or consequences of an accident previously evaluated in the FSAR will not be increased. The fixed resistor allow the compensation loop to resistor thermal detector to function as assumed in the FSAR; thus the possibility of an accident which is different than any previously evaluated in the FSAR will not be created.

The probability of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The resistor installed is of safety-related quality and installed to meet seismic requirements. A malfunction probability is not greater than originally evaluated. A malfunction will cause the channel to trip and, thus, protect the reactor core. The consequences of malfunction will not be increased. The resistor will not increase the possibility of a malfunction of equipment in the process cabinet. It is of negligible weight and securely fastened to the transmitter.

The temperature error is within the uncertainty analysis of the core protection calculator. There will be no effect on the margin of safety.

Special Issue

TAR 89-17, Temporary Alteration for Temporary Chiller Piping

Description of Change

Temporary chillers were located on the East side of the B dry cooling towers. Temporary piping was routed to a manifold system and supported by scaffolding. An eight inch emergency shut off valve was installed between the manifold and the supply line. The system was flushed to meet cleanliness compatibility of the CCW system. All temporary piping was insulated to prevent condensation from forming on the pipe. Electrical service was supplied by temporary power station.

Reason for Change

This TA provided chilled water to the CCS system to support the refueling 3 outage.

Safety Evaluation

Credit was not taken for the class 7 pipe that the temporary chillers are being attached to. There are two isolation valves, CC-8085A/B and CC-8242A/B that protect the safety-related portion of the system from the non-safety portion. The containment cooling system is not required to operate while in modes 5 and 6. The temporary chillers will not be activated until the plant is in modes 5 or 6. The Containment Cooling System is not credited in any accidents for modes 5 and 6. Therefore, the consequences are not affected nor is the possibility for a different type of accident created. To minimize the loss of CCW water, if a leak occurs, a twenty-four hour personnel watch will be posted. To prevent any possible environmental and radiological release, Health Physics and Chemistry will test water samples prior to disposal.

The temporary chillers, associated piping, and scaffolding were evaluated. No equipment will be affected in the event of a malfunction. The quality of water used is the same as normal CCW water, therefore, there is no effect on the CCW system. The chillers are not located near any safety-related equipment. The power supplied is from offsite and does not pose any other electrical concerns.

Special Issue

TAR 89-19, Temporary Access Control Measures For Containment During Refueling Outage

Description of Change

TAR 89-19 established temporary access contr›l measures (electronics) for the duration of refueling outage into conti inment. Existing cardreaders at door 34A was disconnected and wiring for the door will be utilized at temporary cardreaders entering and exiting the Health Physics outage trailer. Door 34A was removed.

Reason for Change

This TA facilitated outage work in the reactor containment while maintaining security for the area.

Safety Evaluation

The door which is removed is used for security purposes and does not affect nuclear safety because security will be maintained at the door. These security measures are the same special security measures used during refueling outage 2 and agreed to by the NRC Region IV personnel. Chapter 10 of the Waterford 3 physical security plan allows for special security measures during outages. The special security measures will assure that accident, equipment malfunction and margin of safety considerations are not adversely affected.

Special Issue

TAR 89-21, Service Air to the Reactor Containment Building

Description of Change

This TA connects temporary air compressors to a leak rate testing piping flange, removes blind flanges for jumper installation, gags relief valve LRT-107, installs a pressure gage, and repositions various valves.

Reason for Change

The purpose of the TAR is to provide an additional source of service air to the RCB during refueling outage number 3.

Safety Evaluation

The air line will be pressurized; however, administrative controls are to be placed upon the system to isolate penetration 63 via containment isolation valve LRT-109 in the event that the line loses pressure. This applies during core alterations and loss of SDC. The line will be isolated in case of loss of system pressure during a loss of SDC scenario or during core alterations. Administrative controls are to be placed upon the system to close LRT-109 in that event. This action will isolate the penetration to form an effective containment closure. The line has been evaluated for air pressures of up to 120 psig and the air compressor reliefs will be set at 120 psig. Since the pressurized line provides an effective boundary to a potential release of radioactivity, and the administrative controls will close containment isolation valve LRT-109 for events stated above, the possibility of an accident which is different is unlikely.

Penetration 63 can still be effectively isolated to form containment closure should the need arise by closing containment isolation valve LRT-109. This capability and the setting of the compressor relief valves assure that the probability or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased. The TA will not in any way prevent the effective closure of containment isolation. Two air compressors will be used to lessen the likelihood of loss of line pressure. This will form an effective boundary to the release of radioactivity. The pressurized boundary and controls to isolate the line assure that the margin of safety is not adversely affected.

Special Issue

TAR 89-23, Placing Gagging Clamp on Valves SI-405A and SI-405B

Description of Change

TAR 89-24 allowed a gag or physical restraint to be placed on valves SI-405A and SI-405B to hold the valves open in modes 5 or 6. The gag is to be removed before entry into mode 4.

Reason for Change

Valves SI-405A and SI-405B were restrained open during modes 5 and 6 to prevent a loss of SDC due to either operator error or inadvertent actuation of the auto-closure interlock (ACI). The ACI is required only for modes 1 through 4.

Safety Evaluation

Gagging SI-405A and SI-405B open defeats the ACI for these valves. The ACI automatically closes SI-405A and SI-405B and SI-401A and SI-401B when pressurizer pressure is above 700 psia. The ACI is described in FSAR sections 9.3.6.2.2.d and 7.6.1.1.1. The purpose of the ACI is to prevent plant heatup without double valve protection between the high pressure RCS and low pressure SDC system. Operator error could allow RCS pressurization with only single valve isolation between RCS and SDC. Failure of the one valve would overpressurize the SDC piping. The SDC piping could rupture outside containment resulting in direct release of the RCS outside containment. ECCS water would also escape containment and would not be available in the containment sump for recirculation for long term core cooling. The ACI is intended to prevent this type of event. The TSs require the ACI to be operable in modes 1,2,3 and 4. Controls are established to prevent heatup to mode 4 unless the gags are removed and verified. The ACI is designed to prevent an accident. It has no effect on the consequences of an accident.

The only accident affected by gagging open SI-405A and SI-405B is over-pressurization of the SDC during plant heatup. Over-pressurization of the SDC and RCS during transient events is protected by the low temperature overpressure protection (LTOP) relief valves SI-406A and SI-406B. Gagging open SI-405A and SI-405B increases LTOP reliability.

The probability of failure of the SDC piping is not affected because the gags will be removed prior to plant heatup to mode 4 when the ACI is required. The reliability of the SDC system is increased because gagging the valves open eliminates a common cause of loss of SDC. The ACI is provided to prevent consequences due to operator error. No consequences due to malfunction of equipment are considered. The only equipment which could be affected is the low pressure SDC piping, which has already been evaluated.

The margin of safety as defined in the basis to any TS will not be reduced. The ACI is required for modes 1,2,3 and 4. Controls are established to ensure the gags are removed prior to mode 4. These controls are equivalent to those established to ensure ACI operability.

Special Issue

TAR 89-24, Placing Gagging Clamp on Valves SI-405A and SI-405B

Description of Change

TAR 89-24 allowed a gag or physical restraint to be placed on valves SI-405A and SI-405B to hold the valves open in modes 5 or 6. The gag is to be removed before entry into mode 4.

Reason for Change

Valves SI-405A and SI-405B were restrained open during modes 5 and 6 to prevent a loss of SDC due to either operator error or inadvertent actuation of the ACI. The ACI is required only for modes 1 through 4.

Safety Evaluation

Gagging SI-405A and SI-405B open defeats the ACI for these valves. The ACI automatically closes SI-405A and SI-405B and SI-401A and SI-401B when pressurizer pressure is above 700 psia. The ACI is described in FSAR sections 9.3.6.2.2.d and 7.6.1.1.1. The purpose of the ACI is to prevent plant heatup without double valve protection between the high pressure RCS and low pressure SDC system. Operator error could allow RCS pressurization with only single valve isolation between RCS and SDC. Failure of the one valve would overpressurize the SDC piping. The SDC piping could rupture outside containment resulting in direct release of the RCS outside containment. ECCS water would also escape containment and would not be available in the containment sump for recirculation for long term core cooling. The ACI is intended to prevent this type of event. The TSs require the ACI to be operable in modes 1,2,3 and 4. Controls are established to prevent heatup to mode 4 unless the gags are removed and verified. The ACI is designed to prevent an accident. It has no effect on the consequences of an accident.

The only accident affected by gagging open SI-405A and SI-405B is over-pressurization of the SDC during plant heatup. Over-pressurization of the SDC and RCS during transient events is protected by the LTOP relief valves SI-406A and SI-406B. Gagging open SI-405A and SI-405B increases LTOP reliability.

The probability of failure of the SDC piping is not affected because the gags will be removed prior to plant heatup to mode 4 when the ACI is required. The reliability of the SDC system is increased because gagging the valve open eliminates a common cause of loss of SDC. The ACI is provided to prevent consequences due to operator error. No consequences due to malfunction of equipment are considered. The only equipment which could be affected is the low pressure SDC piping, which has already been evaluated.

The margin of safety as defined in the basis to any TS will not be reduced. The ACI is required for modes 1, 2, 3 and 4. Controls are established to ensure the gags are removed prior to mode 4. These controls are equivalent to those established to ensure ACI operability.

Special Issue

TAR 89-30, Open Breakers to Valves SI-401A and 401-B During modes 5 and 6 Only

Description of Change

This TAR opened the circuit breakers (SI-EBKR-311A-8D and SI-EBKR-311B-8D) to valves SI 401A and 401B while open during modes 5 and 6. This action removed power to these motor operated valves. The breakers were required to be closed prior to mode 4.

Reason for Change

The alteration was required to prevent the valves from inadvertently closing during modes 5 and 6 causing a possible loss of SDC.

Safety Evaluation

Opening the breakers to motor operated valves SI-401A and SI-401B defeats the ACI for these valves. The ACI automatically closes SI-405A and SI-405B and SI-401A and SI-401B when pressurizer pressure is above 700 psia. The ACI is described in FSAR sections 9.3.6.2.2.d and 7.6.1.1.1. The purpose of the ACI is to prevent plant heatup without double valve protection between the high pressure RCS and low pressure SDC system. Operator error could allow RCS pressurization with only single valve isolation between RCS and SDC. Failure of the one valve would overpressurize the SDC piping. The SDC piping could rupture outside containment resulting in direct release of the RCS outside containment. ECCS water would also escape containment and would not be available in the containment sump for recirculation for long term core cooling. The ACI is intended to prevent this type of event. The TSs require the ACI to be operable in modes 1, 2, 3 and 4. Adequate controls are established to prevent heatup to mode 4 unless the breakers are closed and verified. The ACI is designed to prevent an accident, it has no effect on the consequences of an accident.

The only accident affected by opening the breakers powering SI-401A and SI-401B is over-pressurization of the SDC during plant heatup. Over-pressurization of the SDC and RCS during transient events is protected by the LTOP relief valves SI-406A and SI-406B. Opening SI-401A and SI-401B increases LTOP reliability. The LTOP valves will not be inadvertently isolated from the RCS.

The probability of failure of the SDC piping is not affected because the breakers will be closed prior to plant heatup to mode 4 when the ACI is required. The reliability of the SDC system is increased because opening breakers ensures that SI-401A and SI-401B cannot close. This eliminates a potential loss of SDC. The ACI is provided to prevent consequences due to operator error. No consequences due to malfunction of equipment are considered. The only equipment which could be affected is the low pressure SDC piping, which has already been evaluated.

The margin of safety as defined in the basis to any TS will not be reduced. The ACI is required for modes 1, 2, 3 and 4. Controls are established to ensure that the breakers are closed prior to mode 4. These controls are equivalent to those established to ensure ACI operability.

Special Issue

TAR 89-33, Maintaining Power Supplied to 3AB1 Battery Charger

Description of Change

This TAR modified the plant to a configuration different from that described in the FSAR. An "AB" load was supplied from an "A" train bus. The 3AB1-S battery charger was de-terminated from its normal power supply, 480V MCC 3AB311-S, CUB2D and an electrical jumper connected the charger to 480V MCC 3A311-S, CUB 3M.

Reason for Change

The TAR was necessary to supply power during a bus outage.

Safety Evaluation

The separation and independence between the "A" train and "B" train class 1E distribution systems are maintained so that the probability of a single failure degrading both systems was not increased. The limiting conditions for operation applicable during modes 5 and 6 indicate that a loss of battery charger 3AB1-S and 480V MCC 3A311-S does not decrease the ability to mitigate the consequences of a design basis event provided that "B" train class 1E systems are operable. 480V MCC 3A311-S and/or battery charger 3AB1-S may possibly fail due to this TAR. This would be considered a single failure which has already been evaluated in the FSAR.

The limiting conditions for operation applicable for modes 5 and 6 indicate that battery charger 3AB1-S and 480V MCC 3A311-S are not important to safety provided all "B" train class 1E components are operable. The ability of the "B" train class 1E systems to mitigate the consequences of a design basis accident is not decreased by a malfunction of this TA. Also, this TA will not cause a malfunction of "B" train class 1E equipment.

The operability of the minimum specified AC and DC power sources and distribution systems during shutdown and refueling was not jeopardized. Therefore, the margin of safety was not reduced.

Special Issue

TAR 89-35, Instrument Air (IA) Installation for Containment LLRT

Description of Change

This describes the installation of a rubber hose between IA-5902 (outside of containment near personnel airlock) and IA-915 (inside containment) during performance of the LLRT on the IA containment penetration. Since this hose will run through the personnel hatch, install a quick disconnect on the RAB side to allow for quick isolation of containment, if required.

Reason for Change

This provides IA to containment during the LLRT of the IA containment isolation valves. No equipment will be affected. The mechanical jumper will be installed for approximately four hours. This is approved for use during modes 5 and 6, only.

Safety Evaluation

The instrument air system is not safety related. Complete loss of the system does not reduce the ability of the reactor protective system or the engineered safety features and their supporting systems to safety shutdown the reactor or to mitigate the consequences of an accident. Therefore a total failure of the hose would not cause an unsafe condition to exist. If containment integrity is required, the hose can be disconnected and the personnel airlock closed.

The instrument air system has no effect on the consequences of any accident evaluated in the FSAR, because it serves no safety function. A complete failure of the system does not prevent the reactor protective system or the engineered and safety features and their supporting systems from mitigating the consequences of an accident.

The instrument air system is not safety related. A complete failure of the system could not cause the possibility of an accident.

The instrument air system is not safety related. It has no effect on any equipment important to safety.

The instrument air system has no effect on any equipment important to safety that is used to mitigate the consequences of an accident.

The instrument air system is not TS related. This does not affect containment integrity because the hose can be removed to close the personnel airlock if required.

Special Issue

TAR 89-39, Operating the Containment Fan Cooler Motors in Low Speed

Description of Change

A switch was added to the auxiliary panel controls for containment fan coolers A, B, C, and D to lock fans in low speed. A switch was added to the containment fan cooler trains A and B safety discharge dampers to keep them closed. Instructions were provided to return the fans and dampers to normal conditions if desired.

Reason for Change

The purpose of this TAR is to reduce noise levels in the containment and is only used in modes 5 and 6.

Safety Evaluation

This TAR will not be implemented unless the plant is in modes 5 and 6. The containment fan coolers are not required in these modes in accordance with the FSAR and TS requirements. Therefore, there are no accident probability, consequences or possibilities previously evaluated which are adversely affected. Similarly, malfunctions of equipment important to safety are unaffected. The margin of safety as defined in the basis is not reduced because the coolers are not required by TSs in modes 5 and 6.

Special Issue

TAR 89-41, Maintaining Power Supplied to 3A2 Battery Charger during the 3A31, and 3A312 Bus Outage

Description of Change

This TAR modified the plant to a configuration different from that described in the FSAR. An "A" train component was powered from the "B" train. The 3A2-S battery charger was de-terminated from its normal power supply, 480V MCC 3A311-S and an electrical jumper was connected to the charger to the input terminals of battery charger 3AB1-S.

Reason for Change

The TAR was necessary to supply power during 3A31 and 3A312 bus outage.

Safety Evaluation

This installation does not compromise the separation and independence between components and circuits of the "A" train and "B" train 125V DC systems. Double isolation is provided between the "AB" train and the "B" train 480V busses. The current limiting feature of the 3A2-S battery charger along with the circuit breaker in 480V MCC 3AB311-S will prevent a fault in the "A" train 125V DC system from degrading the "AB" train and "B" train AC and DC system. The battery chargers of the "A" train remain independent of those of the "B" train. This TAR was restricted such that installation was prior to mode 5 and removal prior to mode 4, therefore, it does not represent a violation of TSs.

The separation and independence between the "A" train and "B" train class 1E distribution systems was maintained so that the probability of a single failure degrading both systems was not increased. The limiting conditions for operation applicable for modes 5 and 6 indicate that a loss of battery charger 3A2-S and 480V MCC 3AB311-S does not decrease the ability to mitigate the consequences of a design basis event provided that "B" train class 1E systems are operable. 480V MCC 3AB311-S and/or battery charger 3A2-S may fail due to this TAR. This would be considered a single failure which has already been evaluated in the FSAR.

The limiting conditions for operation applicable for modes 5 and 6 indicate that battery charger 3A2-S and 480V MCC 3AB311-S are not important to safety provided all "B" train class 1E components are operable. The ability of the "B" train class 1E systems to mitigate the consequences of a design basis accident are not decreased by a malfunction of this TA. Also, this TA does not cause a malfunction of "B" train class 1E equipment.

The operability of the minimum specified AC and DC power sources and distribution systems during shutdown and refueling are not jeopardized. Therefore, the margin of safety was not reduced.

Special Issue

TAR 89-43, Replacement of Two Defective Heated Junction Thermocouples (HJTC) with Resistors (Same as TAR 90-06)

Description of Change

TAR 89-43, installs pressure caps on all the guide tubes of the moveable incore instruments and also removes the lower section of guide tube. The moveable in-core instruments have never been operable.

Reason for Change

The caps are required because of RCS leakage and boric acid buildup on the missile shield, vessel head lift rig, reactor head and cavity.

Safety Evaluation

The moveable incore instruments are used to evaluate reload cycle power distributions. This TAR does not change this, therefore it cannot affect or increase the probability of an accident. Pressure cap installation does not affect the function of the fixed ICIs. The movable incore system is a potential path of RCS leakage. The leakage from one tube would be 2.5 gpm and would not exceed the break size evaluated in the FSAR paragraph 15.6, "Decrease in Reactor Coolant Inventory." The cap is being used as intended, to remove the leakage path. The pressure rating of the cap is adequate and recommended by Combustion Engineering, the NSSS vendor. No equipment important to safety is affected. The caps block the movable incore instrument path; however, the function is not required at this time and is considered inoperable. There are no new system interactions. There is a very remote possibility that the cap could become a missile. This potential problem is bounded by the control element assembly ejection evaluated in FSAR paragraph 15.4.3.2. and other missiles in the head area in table 3.5.4. The margin of safety is not adversely affected by this TA.

Special Issue

TAR 89-46, Isolation of Unit Aux Transformer Instrumentation

Description of Change

TA-89-46 isolated sudden pressure relay, unit auxiliary transformer B differential relay, and 87GMT current transformers from the generator protective circuitry.

Reason for Change

The TA was utilized to facilitate maintenance on unit auxiliary transformer B while the unit was on line.

Safety Evaluation

The probability of an accident or its consequences were not increased because power was available to operate or safely shutdown the plant via the startup transformer. The availability of the startup transformer also assured that a different accident than already evaluated was not created and that no malfunction of equipment or its consequences would be increased. The margin of safety was not reduced since TS 3.8.1.1 was met by having power from the startup transformer.

Special Issue

TAR 89-50, Remote Fill For RCP Lower Reservoirs

Description of Change

A temporary red hose (approximately 1 inch) was routed from the RCB el. 46', RCP upper reservoir remote fill station through existing RCP upper remote fill pipe chase down to RCB el. 11' local RCP lower reservoir fill pipe. The connection uses a drill/tap pipe cap hose fitting at RCB el. 11' fill pipe. The connection at the RCB el. 46' RCP upper remote fill station to the existing hand pump will be utilized as needed for fill, otherwise it is plugged.

Reason for Change

The purpose of this alteration is to provide a temporary RCP lower oil reservoir remote fill line to avoid plant shutdown to add oil. Declining oil levels require periodic refill.

Safety Evaluation

The RCP motor lube system is not required for safe shutdown, nor has it any connection to a safety boundary. This TA does not increase the probability of an accident because it is totally detached from safety components. The RCP motor has no RCS pressure boundary and would shutdown prior to damage to any safety-related component.

The probability or consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased because no safety components are involved and the motor malfunction results in shutdown; thus eliminating the impact or consequences. There is no effect on the RCP motor from a minor change to a pipe cap. Therefore, the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR will not be created. There are no TS references or impact from the oil system.

Special Issue

TAR 90-03, Install Pressure Caps for In-Core Instruments (ICI)

Description of Change

TAR 90-03, installs pressure caps on all the guide tubes of the moveable incore instruments and also removes the lower section of guide tube. The moveable in-core instruments have never been operable.

Reason for Change

The caps are required because of RCS leakage and boric acid buildup on the missile shield, vessel head lift rig, reactor head and cavity.

Safety Evaluation

The moveable incore instruments are used to evaluate reload cycle power distributions. This TAR does not change this, therefore it cannot affect or increase the probability of an accident. Pressure cap installation does not affect the function of the fixed ICIs. The movable incore system is a potential path of RCS leakage. The leakage from one tube would be 2.5 gpm and would not exceed the break size evaluated in the FSAR paragraph 15.6 "Decrease in Reactor Coolant Inventory." The cap is being used as intended, to remove the leakage path. The pressure rating of the cap is adequate and recommended by Combustion Engineering, the NSSS vendor. No equipment important to safety is affected. The caps block the movable incore instrument path; however, the function is not required at this time and is considered inoperable. There are no new system interactions. There is a very remote possibility that the cap could become a missile. This potential problem is bounded by the control element assembly ejection evaluated in FSAR paragraph 15.4.3.2. and other missiles in the head area in table 3.5.4. The margin of safety is not adversely affected by this TA.

Special Issue

TAR 90-06, Replacement of Two Defective Heated Junction Thermocouples (HJTC) with Resistors

Description of Change

TAR 90-06, replaces two defective HJTC sensor heater with resistors. HJTC #5 and channel 2 probe heater #7 were replaced with a 25 ohm, 75W resistor.

Reason for Change

This temporary alteration was performed to allow channel 2 HJTC probe to be operable.

Safety Evaluation

The function of the HJTC system is to allow determination of the water inventory in the reactor vessel above the fuel alignment plate. There are two channels of HJTC instrument monitoring. With the channel 2 sensors #5 and #7 out of service, the information is still available from channel 1 sensors #5 and #7. Also, the remaining six sensors in channel 2 will remain operable. The HJTC provides information to the operators and will not affect the ability to shut down the plant. No other equipment will be affected. The reactor vessel level monitoring will remain operable. The change is confined to wiring changes in a monitoring system outside of containment and cannot cause a different accident than previously evaluated. The change will lower the number of operable sensors in the HJTC channel 2 probe from eight to six (three in the head and three in the plenum). The TS allows the channel to remain operable with a minimum of 1 sensor in the head and 3 in the plenum. Thus, the TS basis and the margin of safety is not reduced.

Special Issue

TAR 90-08, CD-230B Actuator Alteration

Description of Change

TAR 90-08 installs a mechanical collar as a clamping device to hold condensate valve CD-230B open. This valve is a manually operated valve which isolates condensate flow to the "B" feedwater pump. During normal plant operation this valve remains open. To hold CD-230B open, the drive nut is rundown the stem to the yoke where the drive nut normally is positioned. To further hold the valve open, two U-bolts attached to an angle iron are clamped around the stem above the gland follower.

Reason for Change

The actuator for Valve CD-230A was damaged and replaced with the actuator from CD-230B. This action was taken in order to close CD-230A for maintenance on steam generator feedwater pump (SGFP) A.

Safety Evaluation

There are no accidents affected by the operation of valve CD-230B. This valve provides isolation of condensate flow to Feedwater Pump "B". The valve remains open during all normal plant operations and the temporary inability to close this valve would only impact maintenance activities of feedwater (FW) pump "B". The valve has no function in the safe shutdown of the plant and does not impact any accident situations. The valve and this TA do not increase the probability or consequences of a malfunction of safety-related equipment. The alteration to CD230-B will not affect any protective boundary. The margins of safety as related to protective boundaries remain unchanged. There are no TSs related to this valve and none of the bases or safety analysis are affected.

Special Issue

TAR 90-11, Vent Hose Installed to Vent Xenon Gas from Demineralized Water Storage Tank

Description of Change

TAR 90-11 installs five hundred feet of one inch rubber hose from the DWST through an existing penetration into the RAB to vent slightly radicactive gas via the RAB ventilation system.

Reason for Change

This TA provides a monitored release path for radioactive gases which have accumulated in the DWST. The release of these gases into the RAB will ensure the monitoring and control required to process through normal operations since the DWST system does not include the venting of radioactive gases into the RAB.

Safety Evaluation

The TAR will alter the normal operation of the system and components of the Demineralized Water System. It will not present any hazards to any of the equipment as the operation will only vent off gasses under very low pressure. The system is equipped to remain under a very slight pressure (0.0 to 2.5 inches of water) to prevent intrusion of air into the DWST. The pressure will be vented off to a pressure of one inch of water at which time the pressure regulator which maintains the tank pressure will be allowed to return the tank to the normal pressure. The gases will be released into the ventilation system of the RAB where the gases can be monitored and controlled prior to release through the plant stack. The gases have been analyzed and determined that the radiological releases and exposures could not exceed the limitations of the design basis accident limits given by FSAR chapter 15 Table 15.7-2. Thus even if all of the gases were to escape to the atmosphere uncontrolled the event would not exceed the consequences of an accident that has already been evaluated within the SAR. Further, the release would not exceed the normal release limitations given by TSs 3.11.2.1 and 3.11.2.2 The gases are currently under very low pressure and are not likely to leak from the hoses or connection that will be made to the RAB.