50-331 DUANE ARNOLD IEL&PC

1992 ANNUAL REPORT OF FACILITY CHANGES,
TESTS, EXPERIMENTS, AND SAFETY AND RELIEF
VALVE FAILURES AND CHALLENGES

Rec'd w/ ltr dtd 2/26/93......9303090309

Table of Contents

SECTION A - PLANT DESIGN CHANGES

This section contains brief descriptions of and reasons for Plant design changes completed during the calendar year 1992 and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR Section 50.59(b). All changes were reviewed against 10 CFR 50.59 by the Duane Arnold Energy Center (DAEC) Operations Committee. None of the changes involved unreviewed safety questions.

The basis for inclusion of a modification in this report is site closure of the associated modification at the DAEC in the calendar year 1992. Portions of some of the Plant Modification Packages (PMP), Minor Modifications (MM), or Design Change Packages (DCP) which are listed were partially closed in previous years.

PMP 0012 Cooling Tower Fill and Drift Eliminator Replacement

Description and Basis for Change

This modification replaced the fill in the original 10 cells of both cooling towers, replaced the original drift eliminators in the 'B' cooling tower, and replaced nineteen of the twenty fan assemblies. Damaged, eroded or decayed structural members in each tower were also replaced.

Summary of Safety Evaluation

This modification has not changed the operation or function of the cooling towers. The cooling towers are not safety-related and cannot initiate any of the accidents previously evaluated in the Safety Analysis Report (SAR). They do not mitigate the consequences of any previously evaluated accident. However, they do provide the heat sink necessary to maintain adequate vacuum in the Plant's main condensers. The modification improved cooling tower capacity which results in increased condenser vacuum during normal Plant operations. Failure of the cooling towers to dissipate this heat load would result in an increase in condenser back pressure and ultimately a Turbine trip if Reactor load was not decreased accordingly. This, in turn, would result in an undesirable increase in nuclear steam system pressure. This scenario, however, is already presented in SAR Section 15.2.

All existing transient/accident analyses in the SAR remain valid. Normal operation (with respect to condenser vacuum) was changed in the conservative direction (i.e., we operate further from the trip setpoint for high condenser backpressure).

PMP 0013 Installation of Insulated Spline Coupling

Description and Basis for Change

This modification involved the installation of an insulated spline coupling in the Turbine front standard area. The insulated spline coupling design was installed to electrically insulate the Auxiliary Control Rotor Gear Assembly from the Turbine-Generator. The coupling protects the gear shaft teeth journals and bearings from electrical pitting damage.

Summary of Safety Evaluation

The only related Technical Specifications are those which address Turbine Stop Valve Closure, Control Valve Closure or Recirculation Pump Trip. These scenarios are not affected by this modification.

This modification is not safety-related. It will not initiate nor increase the consequences of any of the accidents previously evaluated in the SAR, nor increase the probability of occurrence of a malfunction of equipment important to safety. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR has not been increased. No new system requirements or limitations have been created by this modification. This modification will not cause a radioactive release to the environment.

PMP 0016 Hot Tool Crib Mezzanine in Low Level Radwaste Processing & Storage Facility (LLRPSF)

Description and Basis for Change

This modification installed a mezzanine in the Hot Tool Crib which is located in the LLRPSF. The mezzanine provides additional storage and issuing space for the Material & Test Equipment (M&TE).

The free standing, pre-engineered mezzanine is approximately 20' X 26'. It is supported by 14 columns of structural steel tubes welded to base plates which are anchored to the floor. The mezzanine conforms to the requirements of the Uniform Building Code (UBC), and Occupational Safety and Health Act (OSHA).

Summary of Safety Evaluation

The mezzanine is located in the Hot Tool Crib of the LLRPSF which does not house any component that processes low-level radwaste nor does it provide any storage area for spent resins or dry active wastes.

There is no equipment important to safety located in this area of the LLRPSF. Therefore, the erection of the mezzanine did not degrade the performance of any safety system assumed to function in the accident analysis, nor did it impede any equipment which would be needed to mitigate the consequences of an accident.

This activity did not create the possibility of an accident or malfunction of a different type than previously evaluated in the SAR. No equipment changes were performed, and the mezzanine was installed in accordance with UBC and OSHA criteria. There is no accident associated with the modifications that would involve an initiator or failure not already analyzed in the SAR.

PMP 0017 Air Ejector Temperature Indicating Switch Removal

Description and Basis for Change

Previously the Air Ejectors would isolate if a high temperature at the Air Ejector outlet was sensed. This modification removed the isolation function and the associated temperature indication. This improves the reliability of the Offgas System by reducing the potential for unnecessary isolations. The temperature switches were part of a continuous combustion protection system which is not required for systems having catalytic recombiners.

Summary of Safety Evaluation

Removal of an Air Ejector isolation signal has not increased the probability of occurrence of an accident previously evaluated in the SAR because none of the accidents evaluated in the SAR are originated by or caused by the Offgas System.

The Offgas System is not required to mitigate the consequences of any previously evaluated accident. The Offgas System uses steam to remove air from the main condenser and treat the gases before they are exhausted. In the case of the Loss Of Coolant Accident (LOCA), Main Steam Isolation Valve (MSIV) closure will remove the Offgas System from service.

The Offgas System removes dissociated water from the main condenser which potentially could recombine or burn if the correct conditions were present. The temperature indicating switches were installed to monitor for combustion at the outlet of the air ejectors and isolate the line from the main condenser if a temperature of 280°F was sensed. An evaluation of DAEC design determined that these switches were unnecessary. The isolation feature was important on earlier BWRs which did not have catalytic recombiners in the Offgas System. The Offgas System is designed to store the air for 30 minutes to allow for radioactive decay of the short lived isotopes. This can be a large volume of dissociated water that could detonate. The DAEC Offgas System causes catalytic recombination before it is held up for 30 minutes, so the detonation protection is of limited value. Removal of the temperature switch isolation reduces the potential for unnecessary system challenges and improves system reliability.

The Offgas System will isolate to limit offsite releases and is not in service any time the MSIVs are closed or Reactor pressure is below 150 psi. This change in the isolation logic removed a non-radioactive isolation and therefore will not impact the availability of the Offgas System.

The removal of the temperature switch isolation on air ejector discharge temperature will not create any new accidents. An analysis of the complete loss of the Offgas System is encompassed by the MSIV closure event analysis. Additionally, the system is designed to be detonation resistant.

PMP 0018 <u>The Addition of Branch Line Connections To The Cooling and Heating System For The Plant Air Supply</u>

<u>Description and Basis for Change</u>

This modification changed the Cooling & Heating System for the Plant Air Supply, added piping supports to the exterior of the Reactor Building and Recombiner Building, and added temporary exhaust fan support bracket anchor holes to the exterior of the Turbine Building outside the Diesel Generator rooms.

Branch connections were added to the Well Water System supply headers to supply coil banks that cool the Reactor and Turbine Buildings. The modification allows 100% of the Well Water flow to these coil banks through the branch line connections. In addition, these branch line connections may be utilized in the future to provide additional Reactor Building cooling or cooled Well Water for other purposes.

In addition, a series of pipe supports were attached to the precast panels on the exterior of the Reactor Building and recombiner Building to allow temporary piping to be run from the Well Water branch line connections, to a temporary chiller, and back again.

Anchor holes for power cable support brackets were drilled at various locations on the exterior of the

Turbine Building. These holes were used to anchor supports that allowed temporary power cables to be run.

Support anchor bolt holes were drilled into the Turbine Building just below the Diesel Generator room louvers. These holes will be utilized by bracket assemblies to support temporary exhaust fans which will be used to facilitate air flow through the Diesel Generator rooms.

Summary of Safety Evaluation

The Well Water System interfaces with the safety-related Emergency Service Water (ESW) System. During normal Plant operation the Well Water System supplies the Control Building Chillers. During a design-basis accident cooling water is supplied to these units from the ESW System. The systems are separated via check valves. Implementation of this modification did not effect the operation of these check valves. Therefore this modification did not prevent the ESW System from operating correctly. Additionally during normal Plant operation this modification does not influence or cause performance degradation within the Control Building Chillers as this modification does not influence normal flow distribution or Well Water chemistry.

The precast panels on the exterior of the Reactor Building and Turbine Building serve a safety function. However, the addition of piping supports to the Reactor Building panels has not affected their strength capabilities. The bolt holes added to the Turbine Building panels does not significantly affect their strength characteristics.

The installation of pipe supports and support anchor bolt holes to the Reactor and Turbine Buildings did not impact any existing safety systems. The modification activities were carried out in a location removed from any sensitive equipment.

PMP 0023 OSC Relocation

Description and Basis for Change

The former location of the Operational Support Center (OSC) did not lend itself to an efficient working arrangement. It had been recommended that the OSC be relocated in order to alleviate any possible interference caused when the OSC is in operation, especially during a security alert. Relocation of the OSC enhanced its operation during any radiological accident at DAEC.

The existing OSC was relocated from the Security Control Point to the Health Physics (HP) Foreman's

Office/HP Lunchroom. This involved constructing a sound proof enclosure for the Self Contained Breathing Apparatus (SCBA) fill station; removal of the existing glass wall, and erecting a stud wall; and installation of permanently mounted status boards, remote control speakers, and slave clocks. Electrical wiring was run from the master clock and amplifier located in the Technical Support Center (TSC) as necessary.

Summary of Safety Evaluation

The Administration Building is adjacent to, but physically separate from, the Reactor and Control Buildings. These changes to the administration Building have not affected the separation of the buildings or the building structurally. This remodelling merely allowed the OSC to be relocated which improves its ability to function when required.

This modification did not involve any changes to load bearing walls or major power systems. Some communication tie-ins were made to the TSC. This relocation allows the OSC to communicate better with the crews, to be closer to the Plant, and to not interfere with security activities.

These changes to the Administration Building had no effect on any power generation or safety-related equipment. The Building does not house any safety-related equipment or accident mitigation equipment.

PMP 0025 Machine Shop Power Panel

Description and Basis for Change

A new grinder has been purchased for use in the Machine Shop. The existing power panel for the Machine Shop did not have enough power available to operate this new grinder. This modification installed a power panel to provide power for the new grinder and for equipment which will be added to the Machine Shop in the future.

Summary of Safety Evaluation

The Motor Control Center which supplies the power panel is not safety-related and does not interface with any part of the Plant that could cause an accident. The new power panel and all other new equipment installed by this modification does not interface with any part of the Plant that could cause an accident.

The Motor Control Center and all the new equipment installed by this modification do not impact any system important to safety. The equipment involved in this modification does not interface electrically and/or mechanically with equipment important to safety.

PMP 0026 Turbine Building Sample Sink Chiller Replacement

Description and Basis for Change

This modification improved the temperature control in the Turbine Building and the Reactor Building Sample Stations by replacing the existing Turbine Building Sample Chiller with a larger unit. In addition, the condenser piping was re-sized along with the feeder breaker and motor starter to accommodate the larger chiller. The change was designed to improve the performance of the conductivity monitoring instrumentation. In addition, a temperature control valve was relocated and a level indicator was installed to aid in performance monitoring of the chiller. The sample stream piping for the Steam Packing Exhauster was modified to add a line to the sample sink hood that taps off after the rack isolation valves. This allows the sample cooler to be used to provide additional cooling to grab samples taken at the hood.

Summary of Safety Evaluation

The new sample chiller and sample conditioning changes improved the temperature control at the sample stations. The ability to continuously monitor condensate and Reactor conductivity is retained and improved by the improved temperature control.

The Turbine Building Sample Station is located on the Turbine operating deck and is not near any safety-related equipment. The Reactor Building Sample Station is located in the Reactor Building. The physical and electrical separation from equipment important to safety was adequate to ensure that there would be no effect by the modification.

The Technical Specifications provide Reactor water conductivity limitations and require continuous monitoring of condensate and Reactor water conductivity. This modification enhances the ability to continuously monitor conductivity in condensate and Reactor water. This modification is consistent with Technical Specifications requirements.

PMP 0031 Cleanup Phase Separator Tank A&B Sparger Replacement Modification

Description and Basis for Change

The Solid Reactor Water Cleanup Phase Separator Tank spargers provide a means to ensure mixing of the contents of the tank to a homogeneous mixture for transfer. The spargers were plugged and did not allow the contents of the tank to be adequately mixed. In order to alleviate this problem, the sparging system was modified to reduce the likelihood of future plugging problems.

Directional elbows which will not easily plug were installed. The directional elbows will provide the motive force to recirculate the contents of the tank and provide for a consistent slurry for resin transfer.

Summary of Safety Evaluation

The Phase Separator Tanks are part of the solid radioactive waste system which is not a safety-related system. The sparging system does not interface with a safety-related system.

The modification has not affected the integrity of the cleanup phase separator tanks, nor the ability to prevent releases from exceeding 10 CFR 20 limits. The probability of a tank rupture has not been increased since the sparger modification has not impacted the tank integrity.

Enhancement of the operation of the sparging system has not increased the possibility of a malfunction since it improves the mixing function of the sparging system in the tank and causes the tank to operate more effectively.

PMP 0038 EHC Pump Replacement

Description and Basis for Change

The Electro-Hydraulic Control (EHC) Oil System pumps were obsolete and replacement parts were no longer available from the original manufacturer. General Electric recommended replacing the pumps because of possible thrust bearing failures relating to metal fatigue due to an overload condition.

The EHC pumps were replaced with new pumps, and a conversion kit including piping, tubing flanges and fittings was installed.

Summary of Safety Evaluation

The purpose of the EHC Oil System is to provide hydraulic fluid to operate the Main Turbine valves and to provide for rapid closure of these valves under certain emergency conditions.

Chapter 15 of the SAR has analyzed the consequences of Turbine control valve fast closure (without bypass) and

Turbine stop valve closure (without bypass). The consequences of a failure of the EHC pumps would not increase the probability of occurrence of a transient, nor would it result in a more severe transient than any previously analyzed in the SAR. Environmental qualifications, seismic criteria, fire protection, heavy loads and design requirements have all been analyzed.

The margin of safety as defined in the Technical Specifications Bases is not reduced because the MSIVs will continue to meet the required closure time.

PMP 0039 Replace MSIV N_2 Piping in Steam Tunnel

Description and Basis for Change

This modification replaced the non-safety copper nitrogen supply piping to the outboard MSIV accumulators with stainless steel piping. The reason for this modification was to replace the sweated joints in the copper piping with stainless steel piping with swagelock fittings.

The Reactor Building 2-inch copper piping was replaced with 3/4-inch stainless steel piping. In the steam tunnel, 3/4-inch stainless steel piping replaced the 2-inch and the 1/2-inch copper piping. The 2-inch and the 3/4-inch copper piping in the CRD Repair Room was replaced with 3/4-inch stainless steel piping. Several nitrogen supply valves were also replaced.

Summary of Safety Evaluation

The probability of occurrence of an accident previously evaluated in the SAR was not increased. The MSIVs do not initiate any of the accidents, but they do mitigate the consequences of certain accidents once they have occurred.

The consequences or probability of a malfunction of equipment important to safety previously evaluated in the SAR were not increased because the modification did not affect the ability of the outboard MSIVs to close. Regardless, the failure of the outboard MSIVs to close does not increase the consequences of a malfunction of the outboard MSIVs because the redundant inboard MSIVs will still isolate the steam lines. The margin of safety has not been reduced because the MSIVs will continue to meet the Technical Specifications requirements of closing in 3-5 seconds.

PMP 0040 Fire Brigade Room Expansion

Description and Basis for Change

This modification expanded the existing Fire Brigade Assembly & Storage Area by removing an existing wall and building another wall. This provides additional space for equipment storage and dressing requirements for the fire brigade members.

Emergency lighting and area radiation monitoring components were relocated to accommodate the expansion. The Plant Public Address System was modified to enhance its capabilities in the Fire Brigade Room (FBR). The security door intrusion alarm components were transferred to the new door location. The Administrative Building Heating, Ventilating and Air Conditioning (HVAC) System was modified to provide better air flow to the modified FBR. Also, the temperature controller was relocated.

Summary of Safety Evaluation

The Fire Brigade Room is located in the Administrative Building which is a non-safety-related structure. It houses no safety-related equipment or systems. The modification did not affect any safety-related system or impact the original design intent of DAEC. No new failure modes for equipment were introduced by this modification. The structural integrity of the Administration Building was not impaired by this modification.

PMP 0042 Digital Imaging Power and Cables

Description and Basis for Change

This modification installed power outlets for the Digital Imaging computer in the TSC library computer room. A regulating transformer was installed in the Data Acquisition Center (DAC) mechanical room and power cables were run to the TSC library. Coaxial cables were run from the TSC library to the Control Room for the terminals and printers associated with Digital Imaging and the computerized tagout system. A new 30 amp breaker was installed to supply the regulating transformer. Cable was run from the transformer to a distribution panel, which was installed in the TSC library. This required a core drill through the TSC block wall.

Summary of Safety Evaluation

The SAR does not identify any accidents associated with this equipment. The new transformer, distribution panel and power cables do not interface with any part of the Plant which could cause an accident. The cables from the TSC library to the Control Room use existing routings for computer and phone cables. The routings do not interface with any cables which are important to safety. None of the new equipment installed or involved in the modification impacts any system important to safety. There are no failure modes associated with this system that effect equipment important to safety. The electrical and/or mechanical interfaces are not with equipment important to safety.

PMP 0043 <u>Air Compressor Load Center Circuit Breaker Installation</u>

<u>Description and Basis for Change</u>

The modification replaced three load side 480 volt 30,000 amp short circuit rated breakers at each Plant Air Compressor load center, with 42,000 amp short circuit rated breakers. The new breakers are sized per IEEE/ANSI standards which recommend sizing circuit breakers large enough to handle the worst case three-phase fault current.

<u>Summary of Safety Evaluation</u>

This modification involved changes to the electrical power supply to the Plant Air Compressors. These modifications to the Instrument Air System did not change the functional operation of the system and was not safety-related. There was no potential for increasing radiation exposure or for affecting the Design Basis Accident (DBA) analysis.

The circuit breaker change-out does not affect the SAR analysis which evaluated the complete loss of the Instrument Air System. This modification does not decrease the reliability of the Instrument Air System. Technical Specifications are not affected since they do not specifically address the Instrument Air System or its power supply.

PMP 0050 <u>Installation of Sliding Link Disconnect Terminal Block on IST Instrument Loops</u>

<u>Description and Basis for Change</u>

This Plant modification installed sliding link disconnect terminal blocks with banana jack connector adapters on the Inservice Testing (IST) instrument loops for the following: High Pressure Coolant Injection (HPCI) Pump suction and discharge pressure indicators, Reactor Core Isolation Cooling (RCIC) Pump suction and discharge pressure indicators, Core Spray Pumps discharge pressure indicators and ESW Pumps flow indicators.

The sliding link terminals provide a reliable means of opening the IST instrument loops for connection of digital multimeters without removing any wires from the terminal points.

Summary of Safety Evaluation

This modification did not change the function or the operation of the related system. This modification only affected the electrical portion of non-safety-related instruments whose failure does not initiate any accident.

The sliding link disconnect terminal blocks which were installed can only malfunction if they were to inadvertently open. The construction of these sliding link disconnect terminal blocks makes this possibility extremely unlikely. Even if all the sliding links were to open they would only affect the indicators and these indicators do not initiate any malfunctions.

PMP 0055 Turbine Auxiliary Upgrades

Description and Basis for Change

Several modifications were made to the Main Turbine support equipment to increase monitoring capabilities, ease of operation, and maintenance of the support equipment. Piping from the unused steam blanketing system was installed to allow venting of the Moisture Separator Reheater (MSR) 2nd stage. The EHC pump vent line was relocated from upstream of the high pressure discharge filter to downstream of the filter. This change allows for the purging of air from the pump, piping, and filter after change out of the filter cartridges, thus significantly reducing the danger of air-induced hydraulic transients. Additional drain and vent valves were installed on the EHC fluid reservoir.

The delta P switch which indicates proper vacuum in the Turbine Main Lube Oil Tank was modified to provide Control Room indication. The switch was previously installed but not connected to the Control Room panels. A local gauge was installed at the tank to allow for setting and trouble shooting of the switch. A pressure gauge was installed to indicate the Emergency Bearing Oil Pump (EBOP) discharge pressure when the Main Turbine is at full speed.

EHC filter cartridges were replaced with new filters proven to be more effective at maintaining the hydraulic fluid within specifications. Also, a vent was installed on the filter recirculation tank. The vent serves to prevent potential damage due to over pressurization when new filters are installed and packed prior to being put into service.

Differential pressure indication on the EHC pump discharge filters was installed to provide positive indication of the condition of the filters and when they need to be replaced. Differential pressure indication across the stator water cooling deionizer pre-filter was also installed. This indication is needed to determine the filter cartridge condition.

Summary of Safety Evaluation

The modifications did not create new accidents or malfunction possibilities, nor affect accidents or malfunctions discussed in the SAR. Startup testing demonstrated that MSR 2nd stage vents could pass only an insignificant quantity of steam if connected directly in line due to the small piping size and the pressure drop through that piping, valves and orifices. Reactor pressure or the condenser vacuum pressure could not be adversely affected. This modification did not affect the Turbine trip associated with a high level in the MSRs.

The Nuclear Safety Operational Analysis (NSOA) and the SAR evaluate several operational transients. Two of those transients which could be affected by this modification are Turbine-Generator trip with bypass and Turbine-Generator trip without bypass. This modification utilized components and installation methods that met the same quality and standards as the original equipment, such that the probability of the above transients was not increased. These modifications did not affect the transients. Additionally, the modifications did not prevent the actions required to mitigate those transients (i.e. isolation of the Main Turbine). The modifications are limited to the Turbine-Generator, therefore there will be no effect on the probability of an accident as discussed in the NSOA or the SAR.

PMP 0056 LLRPSF HVAC Control Modification

Description and Basis for Change

The ability of the LLRPSF HVAC controllers to control temperature was improved by changing the controller's feedback from supply air temperature to return air temperature. Since supply air temperature did not vary with changes in area temperature, the system was unable to respond to changing conditions. By monitoring return air temperature and supplying this information to the controller as feedback, the system is now able to respond to changing area temperatures and maintain the area temperature within acceptable ranges.

Summary of Safety Evaluation

Since this modification did not alter the intended function of the system it was considered to be within the original design basis. Additionally, temperature control of areas of the LLRPSF is not required to prevent or mitigate any accident as defined in the SAR. It is also not required to maintain any equipment operable which is important-to-safety.

PMP 0057 Relocation of Production Well No. 1 "B" Pump

Description and Basis for Change

A new well to serve as Well No. 1 was drilled because the output of the old well had fallen from 750 gpm to between 200 and 400 gpm. The total flow capacity from Well No. 1 was increased to allow "B" pump discharge to meet all current system demands, should it be necessary to shut down all other pumps. This also allows for additional flow demands to be added in the future.

The new well was drilled deeper to draw from an aquifer which is recognized to have a larger capacity and greater reserves than the old source. Most of the original components were moved to the new well's location.

Other system upgrades, such as resizing the well pump and drive motor, increased the flow capability of Well No. 1 and added flexibility to the rest of the system.

Summary of Safety Evaluation

None of the components or equipment related to this system are safety-related. The Well Water System neither directly or indirectly affects any of the accidents investigated in the SAR. Well Water is not one of the Plant systems considered in the NSOA.

No components or equipment installed by this modification were located such that failure of a restraining device could cause damage to any safety significant equipment.

The equipment and components which were installed will not jeopardize the operation of any existing safety-related components, either by their failure to function as required, or by their ability to stay restrained during a seismic event.

The Technical Specifications do not address the Well Water System, nor do the Well Water System or any of the proposed changes affect any of the systems that are covered by the Technical Specifications. Therefore, no operational requirements were affected by the changes and margins of safety were not decreased.

PMP 0058 Power Feeds for MT&E & Office Space

Description and Basis for Change

Permanent power feeds from lighting panels in the Administration Building to the M&TE tool crib and the additional office space trailer were installed by this modification.

Summary of Safety Evaluation

This modification did not increase the probability of occurrence of an accident previously evaluated in the SAR because none of the accidents in the SAR are originated or caused by power feeds from lighting panels in the Administration Building.

The power feeds supply loads outside of the power block. There is no interface with any equipment important to safety. These power feeds have a circuit breaker in the panels that will trip if a fault occurs in the trailers.

PMP 0060 Leak Repair of Well Water Chemical Injection Piping and Check Valve Addition

Description and Basis for Change

This modification enhanced the Well Water chemical injection path and added a new check valve to prevent backflow of chemicals into the Demineralized Water System. Injection piping was changed from carbon steel to stainless steel to reduce corrosion. The larger piping required replacement of the 3/8" check valve with a 1" valve. Repairs were made to portions of the injection piping which were temporarily leak repaired earlier.

Summary of Safety Evaluation

Changing carbon steel material to stainless steel to reduce corrosion, changing check valve size and adding a check valve to prevent chemicals from being drawn back into a non-safety-related system did not increase the probability of occurrence of an accident previously evaluated in the SAR because none of the accidents evaluated in the SAR are originated or caused by the chemical injection to the Well Water System. Nor is such an injection to the Well Water System utilized in the mitigation of any accident scenarios described in the SAR.

Adding a check valve to prevent backflow of chemicals and fixing the leak has not affected the safety of the Plant. There was no increase in the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR. This modification enhanced the ability of the Well Water System to perform its non-safety-related function. The modification did not compromise system integrity.

MM 115 120 VAC Utility Outlet Removal

Description and Basis for Change

This modification removed the 120 VAC utility outlet inside the Reactor Vessel Level and Pressure Instrument Rack. The outlet was for the convenience of maintenance personnel, but it interfered with access to terminal strips. The removal of the outlet did not affect the functions of any of the other circuits contained in the instrument rack.

Summary of Safety Evaluation

The utility outlet that was removed is completely independent of and has no effect on the operation of the other circuits contained in the junction box.

No system functions or characteristics were changed. The outlet was removed to allow for access to the terminal strips in the junction box. No new failure modes were introduced.

MM 129 Miscellaneous Lighting Upgrades for the Reactor Building and HPCI Room

Description and Basis for Change

The function of the lighting in the Reactor Air Intake Room, the HPCI access platform, and the Outer Drywell access area is to provide illumination for ingress and egress of personnel and to facilitate inspection and maintenance activities.

This modification raised lighting levels in these areas. Fluorescent lights which have reduced output in low temperatures were replaced with incandescent lights.

such an injection to the Well Water System utilized in the mitigation of any accident scenarios described in the SAR.

Adding a check valve to prevent backflow of chemicals and fixing the leak has not affected the safety of the Plant. There was no increase in the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR. This modification enhanced the ability of the Well Water System to perform its non-safety-related function. The modification did not compromise system integrity.

MM 115 <u>120 VAC Utility Outlet Removal</u>

<u>Description and Basis for Change</u>

This modification removed the 120 VAC utility outlet inside the Reactor Vessel Level and Pressure Instrument Rack. The outlet was for the convenience of maintenance personnel, but it interfered with access to terminal strips. The removal of the outlet did not affect the functions of any of the other circuits contained in the instrument rack.

<u>Summary of Safety Evaluation</u>

The utility outlet that was removed is completely independent of and has no effect on the operation of the other circuits contained in the junction box.

No system functions or characteristics were changed. The outlet was removed to allow for access to the terminal strips in the junction box. No new failure modes were introduced.

MM 129 <u>Miscellaneous Lighting Upgrades for the Reactor Building and HPCI Room</u>

<u>Description and Basis for Change</u>

The function of the lighting in the Reactor Air Intake Room, the HPCI access platform, and the Outer Drywell access area is to provide illumination for ingress and egress of personnel and to facilitate inspection and maintenance activities.

This modification raised lighting levels in these areas. Fluorescent lights which have reduced output in low temperatures were replaced with incandescent lights.

Summary of Safety Evaluation

The change of fixture types and the addition of fixtures in the Reactor Building was not safety-related and did not affect the function of any safety-related systems. The added lights are not located near safety-related equipment. The falling of fixtures has been prevented by supporting the fixtures with rigid steel pendants, or by mounting the fixtures with anchors.

MM-272 Security Microwave System Upgrade

Description and Basis for Change

The Perimeter Intrusion Detection System (PIDS) consisted of microwave systems and E-Field systems. The microwave units had an unacceptable false alarm rate when the sensitivity was set high enough to detect a crawling intruder. The E-Fields were a high maintenance item which caused many nuisance alarms from birds landing on the E-Field wires.

This modification replaced these systems with two diverse types of microwave systems. The two new units have been stacked on the existing microwave poles to give better coverage against intruders. By setting each type of microwave unit to detect the threat for which it was designed, the false alarm rate has been reduced.

The four E-Field systems at the intake structure were replaced with a microwave system. This system alleviates maintenance problems and nuisance alarms.

Summary of Safety Evaluation

The security modification was done in accordance with the guidelines of applicable sections of 10 CFR 73, "Physical Protection of Plants and Materials," Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage," ANSI N18.17, "Industrial Security for Nuclear Power Plants" and the DAEC Security Plan.

None of the equipment involved has any seismic or environmental qualification requirements. There was no change to any DAEC Licensing Analysis or NRC compliance program as a result of this modification.

The Security System interfaces with other Plant systems are through approved isolation devices which are coordinated to prevent the propagation of malfunction. There is no direct interface with any Reactor control system, engineered safety system, or safety supporting system.

The new perimeter intrusion detection systems have equal or better detection capability and reliability. The new systems were connected into the Security System in the same manner as the existing system.

MM-274 Condensate System High-Point Vent

Description and Basis for Change

A tygon hose connected to the Condensate System high-point vent was replaced with permanently installed piping. The modification rerouted the condensate vent piping discharge to an existing feedwater heater open radwaste drain trough. The manual vent valve was repositioned in a manner that allows an Operator to see when the condensate high-point vent has completed venting.

Summary of Safety Evaluation

The original condensate high point vent system function has not been changed. The condensate vent system is non-safety related and no safety-related equipment was affected by this modification. The failure of the rerouted vent piping will not cause a malfunction or failure of any safety-related system.

This modification meets the same codes and standards as the other vents in the Condensate System and is essentially identical to the existing vent systems; therefore, no new failure modes have been introduced.

DCP 1284 Diesel and Electric Fire Pump Modifications

Description and Basis for Change

This modification affected the Diesel and Electric Fire Pumps, and portions of their ancillary equipment. The Diesel Fire Pump engine fuel supply shut-off valve extension rod/handle was relocated to an area where a fire on the diesel engine will not prevent fuel shutoff. A wye strainer was added to the Diesel Fire Pump water-cooled oil cooler supply line to prevent plugging of the oil cooler supply pressure regulator.

The pressure sensing lines for the Electric and Diesel Fire Pumps were modified by removing redundant valving in each pump pressure sensing line and replacing the remaining valves with visual indicating type valves. This change increased the reliability of each Fire Pump to start automatically and to supply water pressure when system pressure is reduced.

The water supply tap for the sprinkler system protecting the Diesel Fire Pump Room and the Hose

Stations in the Pump House was relocated. This prevents having to isolate other fire suppression systems when the Electric or the Diesel Fire Pump is isolated.

An isolation root valve for a pressure indicator at the Fire Main Pressure Surge Tank was installed. This allows the pressure indicator to be removed for calibration without isolating and depressurizing the surge tank.

Modifications were made to the Fire Pump piping to reduce flow oscillations and pressure surges to the Fire Protection System. The recirculation flow control valve was changed from a six-inch valve to a one-and-one-half inch valve. The sensing line for the controller was relocated. The Jockey Fire Pump discharge line was relocated so that the Fire Protection System could remain pressurized while fire pump tests are being conducted.

Summary of Safety Evaluation

The Electric and Diesel Fire Pumps and the encompassing Fire Suppression Water System are not safety-related systems. The safety-related systems for which the Fire Protection System provides protection do not require fire protection to be able to perform their safety-related functions. The modifications meet the same codes and standards and are located in the same area as existing fire protection equipment.

The changes to the Electric and Diesel Fire Pumps have, however, improved the reliability and functionality of the Fire Protection System without changing the way the system operates. The probability of a malfunction in the operation of the Fire Protection System due to these modifications has not been increased, therefore there has been no affect on the ability of safety-related systems to perform their design function.

DCP 1396 MO-2700, V-14-02 and V-14-04 Packing Configuration Modification

Description and Basis for Change

MO-2700 is a Primary Containment isolation valve for the Reactor Water Cleanup (RWCU) System. V-14-02 and V-14-04 are feedwater isolation valves located downstream of feedwater check valves V-14-01 and V-14-03. This modification improved the packing configuration of MO-2700 by reducing the number of packing rings to five and eliminating the use of braided asbestos. Leakoff ports on V-14-02 and V-14-04 were plugged and the existing leakoff piping removed.

The leakoff port plug on MO-2700 was replaced with a stainless steel plug. All plugs were seal welded to their respective bonnets around their entire perimeter. The leakoff piping from V-14-02 and V-14-04 tied into drain piping at a point further downstream. At the point where this piping tied into downstream piping, it was plugged to maintain operability of the remaining drain piping.

Summary of Safety Evaluation

The improved packing configuration of MO-2700 does not alter the operation or design intent of this valve. The design of the new packing configuration ensures that the possibility of system leakage or valve failure via packing failure has not been increased. The consequences of packing failure and resulting stem leakage have not been modified by this change. The consequences of system leakage were not changed.

Changing the packing configuration did not alter the isolation logic or the Technical Specifications maximum closing time for MO-2700. The reduced packing-induced stem loads actually decreases the probability of MO-2700 exceeding the Technical Specifications maximum closing time.

The packing configuration of MO-2700, V-14-02, and V-14-04 cannot affect any of the SAR Chapter 15 analyses. Seal welding was performed in accordance with applicable codes and procedures.

DCP 1429 Additional 161 KV Breaker in Switchyard

Description and Basis for Change

An additional 161 KV breaker was installed in the switchyard. This connected the West bus with the Startup Transformer via a new Breaker (K). Previously the East bus powered the Startup Transformer via Breaker J, leaving the DAEC vulnerable to a Loss Of Offsite Power (LOOP) due to a single failure. The new Breaker K provides a breaker-and-one-half scheme which eliminates the possibility of a LOOP due to a single failure.

The modification also installed breaker controls in the Control Room for Breaker K which are identical to those for Breaker J, (a control switch, status lights and control failure annunciation). Cables required for these Control Room modifications were pulled out to the switchyard.

Summary of Safety Evaluation

The modifications were designed and installed using design criteria that are the same or more stringent than the design criteria used in the original Plant design.

Adding the breaker-and-one-half scheme to the startup transformer eliminates vulnerability to a LOOP due to a single failure and provides another source of site power. The ability of offsite power to support equipment required to mitigate the consequences of an accident have not been degraded. Therefore, this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Breaker K was incorporated into the breaker failure logic. Breaker K was included in the transformer differential relay scheme to maintain West bus protection. Synchronizing switches and sync-check relaying ensure that the East and West buses are in phase prior to closing breakers J and K.

DCP 1431 Administration Building HVAC Controls Modification

Description and Basis for Change

A programmable logic controller (PLC) was installed to provide all temperature and humidity controls and electrical interlocks for the Administration (Admin) Building HVAC System. A terminal was installed and programmed to provide interface with the PLC to operate all pumps and fans, control all temperature settings and monitor the complete HVAC system. This equipment replaced all of the pneumatic controls and hard wired hand switches and indicating lights.

Summary of Safety Evaluation

This change affected the method of controlling the humidity and temperature in the Administration Building. No equipment important to safety or equipment which can affect the accident analysis previously evaluated in the SAR was affected by this system or system modification. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR has not been increased.

All electrical interlocks continue to operate as before. The Administration Building HVAC controls have no effect on safety-related systems or equipment in the Plant. Thus, the possibility of an accident or malfunction of a different type than any previously evaluated in the SAR has not been created.

DCP 1435 Move KAMAN Software to VAX 8600

Description and Basis for Change

This modification moved the radiological monitoring (KAMAN) software that was formerly on the PDP 11/34 computer to the Plant Process Computer (PPC), a VAX 8600. This change was made to consolidate computer devices at the DAEC. New software was written for the VAX 8600 that emulates the software that was formerly used in the PDP 11/34. After the VAX software change had been verified and tested, as required by DAEC quality assurance controls, the PDP 11/34 was removed.

Summary of Safety Evaluation

The KAMAN system operation was not affected by these changes. The KAMAN software changes were independent of the KAMAN "Data gathering" hardware. The software changes only affected the transfer of the data from one computer to another and not the operation of the KAMAN hardware. The software conversion did not involve any changes to the existing software that performs the existing computer routines. Offsite dose calculations will still be performed using the KAMAN data.

The software conversion from the PDP 11/34 to the VAX 8600 did not involve any interfaces with safety-related equipment. The software added to the VAX 8600 simply replaces the software used in the PDP 11/34; it performs the same functions as the PDP 11/34 did.

DCP 1444 Replacement of EHC Supply Flex Hoses

Description and Basis for Change

In 1989 a crack developed on a one inch high pressure supply flex hose to Main Turbine Control Valve 3.

An in-depth investigation of the flex hoses determined that the flex hose would dampen out vibration and/or dynamic movement in only a one dimensional plane. However, if movement in two or more dimensions were to occur, additional flex hoses would have to be installed to dampen out each dimension of movement. Since thermal, dynamic, fluid pulsation, and vibration movement could exist in all three dimensions, and because the cracked flex hose showed signs of both torsional and mechanical bending stresses, the high pressure EHC supply flex hoses were replaced with an improved type of hydraulic hose.

Summary of Safety Evaluation

The worst-case failure of the flexible hoses would be a leak or rupture, resulting in a loss of EHC fluid. If the leak was severe, the loss of fluid and decrease in EHC oil pressure would eventually result in an automatic closure of the Main Turbine Stop Valves, Control Valves, and Combined Intermediate Valves. These valves are designed to close on a loss of system pressure (bypass valves fail as-is).

Chapter 15 of the SAR has analyzed the consequences of Turbine Control Valve fast closure (without bypass) and Turbine Stop Valve closure (without bypass). The consequences of a failure of the EHC hoses would not result in a more severe transient than previously analyzed in the SAR. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased by this modification.

The manner in which the EHC System performs was not changed by this modification. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the SAR was not created.

The Technical Specifications Bases for fuel cladding integrity evaluate the results of Turbine Stop Valve closure and Turbine Control Valve fast closure and the margins required for the Safety Limit Minimum Critical Power Ratio. These margins of safety would not be reduced by a worst-case failure of the Turbine EHC oil system. Technical Specifications define the requirements for Reactor Protection System (RPS) actuation for Turbine Control Valve fast closure and Stop Valve closure. These requirements were not affected by the replacement of EHC hydraulic hoses.

DCP 1445 Upgrade KAMAN Monitoring Skid

Description and Basis for Change

This modification made several changes to the KAMAN monitoring skids and their associated microcomputers (micros).

The particulate and iodine detectors were removed. Particulate and removable iodine filters were installed, and an upgrade of the 'firmware' (chips and documentation) on the monitoring skids and associated microcomputers was made. These four skids and micros were Reactor Building Exhaust Vent #1, #2, and #3 normal range, and Low Level Radwaste Building normal range monitors.

An upgrade of the firmware in the other seven KAMAN microcomputers was performed. This change was made to keep the software revision in all eleven micros the same. The micros involved were the Turbine Building accident and normal range micros, the Reactor Building accident range micros, and the Offgas Stack accident and normal range micros.

Summary of Safety Evaluation

The function of the KAMAN radiation monitoring system is to acquire and display radiation data taken from the Turbine Building and Reactor Building vents, the Offgas Stack, and the Low Level Radwaste Processing & Storage Facility (LLRPSF). The monitor in the LLRPSF ventilation exhaust initiates shutoff of the facility fans as described in SAR Section 9.4.8. This modification did not affect the monitoring capability or the isolation capability of the Kaman system. It only removed unused detector assemblies and replaced them with filter assemblies which collect particulate and iodine samples for off-skid analysis. The on-line monitoring of particulates and iodine on the accident range monitor skids remains. The firmware changes removed the alarms associated with the particulate and iodine detectors, and improved the availability of the KAMAN system. Therefore, this modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously analyzed in the SAR.

The design functions of the existing KAMAN system were not changed. Therefore, this modification did not create the possibility of an accident or malfunction of a different type than any previously addressed in the SAR.

The information taken from the existing particulate and iodine detectors is not used in dose rate calculations. Therefore, this modification has no effect on dose rate or dose calculations.

DCP 1453 Modifications to Control Building Chillers

Description and Basis for Change

This modification to the Control Building HVAC Chilled Water System helped resolve functional, operational and maintenance concerns. Existing dp type chilled water loss-of-flow switches and associated sensing lines were replaced with qualified Class 1E electronic flow switches. These electronic flow switches were located so that turbulent flow would not interfere with the switches' operation. They have a higher degree of overall reliability since the electronic flow switch design involves no moving parts.

The condenser water flow switch was removed from each chiller and a separate temperature sensor input signal was wired to each Temperature Load Controller (TLC).

The TLCs control compressor loading and stabilize equipment operation during normal and emergency operation. The flip-pak and reed switch that were formerly installed on the "A" Chiller were replaced by a Class 1E flip-pak and reed switch.

Summary of Safety Evaluation

The chillers maintain comfort for Operators in the Control Room and maintain ambient temperatures below the level that may cause equipment damage in the Control Building. These modifications improved the reliability of the chiller. No failure mode of the chillers can cause any of the credible accidents evaluated in the SAR, or any new accidents.

The elimination of the switch contacts in the sensor circuits by adding another identical temperature sensor does not change the operation or the function of the chiller system. It does eliminate the unwanted variation in input resistance which was caused by the switch contacts changing resistance due to corrosive effects.

Replacing the currently installed flip-pak and reed switch with a Class 1E flip-pak and reed switch improved the reliability of the chiller. It did not change the operation or the function of the chiller system.

Relocating the chill water flow switch to an area where flow is less turbulent improved the stability of the monitoring process, thereby providing a more reliable output. This provides a more stable operation of the flow switch and thereby the chiller. Changing from a dp type flow switch to a direct insertion type flow switch allows monitoring of the actual flow and not just an indirect variable of the process. This provides better protection of the chiller against freeze-up.

Eliminating the unwanted varying relay contact resistance by providing a separate temperature sensor for each TLC also increased chiller stability.

The added temperature sensor is a type identical to those presently in use. This temperature sensor performs a safety-related function in that it is responsible for the input signal to the temperature controller of the chill water to the Control Room HVAC system. All basic design functions of the Chilled Water System remain unchanged by this modification and reliability has been improved.

Based on the above, the modification did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. Since the chillers are not accident initiators, no failure mode of the chillers can cause any of the credible accidents evaluated in the SAR.

DCP 1456 <u>Radwaste Dewatering Station Relocation</u>

<u>Description and Basis for Change</u>

The purpose of this change was primarily to facilitate resin dewatering operations. The Radwaste Dewatering Station was relocated from the Radwaste Building basement to the LLRPSF. The dewatering equipment was also relocated, and the existing resin feed line was extended to a convenient location for filling of the High Integrity Container (HIC) liners. The dewatering return was routed back to the Radwaste Building Conveyor Drain Sump. Radiation (gamma) detectors were located in the dewatering pit with local readout located near the dewatering equipment for use during its operation. In addition, the area formerly used for dewatering was modified to facilitate future use.

<u>Summary of Safety Evaluation</u>

In general, all changes were operational or location enhancements which have not increased the probability of occurrence or the magnitude of the consequences of an accident. This change only moved the location where the HICs will be dewatered. The dewatering process itself has not been changed, and the amount of radioactive material on site has not been increased. The HICs will be dewatered in a pit, which is located below grade in the LLRPSF versus on a trailer in the Radwaste Building. A worst-case malfunction of the HIC dewatering equipment in the Radwaste Building would have resulted in the release of the liquid contents of a full HIC. By moving the equipment to the LLRPSF, any spills during routine dewatering would be contained because the HIC would be located in a pit. The LLRPSF was designed and constructed to the same design requirements as the Radwaste Building.

The probability of a spill during normal operations was not increased because the equipment used was not changed. The addition of radiation monitors to the dewatering pit was a monitoring enhancement which will not affect the control or release of radioactive wastes. By not needing to use hand held monitoring, ALARA practices are enhanced. Coating the dewatering pit with epoxy sealer improved the radioactive cleanup characteristics of the pit, and therefore had no adverse affect on safety.

DCP 1459 Removal of Radwaste Drum Conveyor System

Description and Basis for Change

The purpose of this change was to remove the Radwaste Drum Conveyor System and associated equipment in order to reclaim the space for more productive uses such as mixed waste storage.

The modification isolated and removed the Drum Conveyor System and associated panels, electrical equipment, and hardware equipment. The Radwaste Centrifuges, Radwaste Centrifuge Hoppers and their associated equipment were isolated but left in place. The associated annunciators and switches were also removed.

Summary of Safety Evaluation

The Radwaste Drum Conveyor System removal in no way adversely affected the analyzed SAR events. Since the Radwaste Drum Conveyor System was isolated at its points of interface with other equipment, its removal has not adversely affected the operation of the interfacing equipment. The isolation of the Radwaste Centrifuges has not adversely affected the operation of interfacing equipment since the centrifuges were no longer being used. No new forms of processing were introduced by this project.

The presently used method of solids processing and dewatering was not changed, and the amount of radioactive materials on site was not increased.

DCP 1466 EOP 3 Water Level Indication

Description and Basis for Change

An entry condition for Emergency Operating Procedure 3 (EOP 3), "Secondary Containment Control", considers the water level of the HPCI Room, RCIC Room, Torus Room, and the RHR/Core Spray Corner Rooms. EOP 3 also requires the levels be monitored anytime entry into this EOP is required.

To prevent having to visually (locally) verify the water levels, this modification installed water level elements in the HPCI Room, RCIC Room, Torus Room, and the RHR/Core Spray Corner Rooms. These levels are indicated on meters in the Control Room.

Summary of Safety Evaluation

The function and operation of this modification is indication only and does not effect any safety systems. The seismic requirements for the safety-related panel which holds the level indicators, level switches and power supply have been met. The core bores which were made to facilitate routing conduit were made per the guidance of "Guidelines for Cutting Penetrations in Existing Masonry Block Walls, Poured Concrete Slabs and Walls and Structural Steel." Because the modification does not affect the operation or reliability of other systems in the Plant, the Plant margin of safety as defined in the Technical specifications has not been reduced.

DCP 1468 Intake Structure Circuit Breaker Replacement

Description and Basis for Change

Load Centers (LC) 1B-9 and 1B-20 supply the River Water Supply (RWS) pumps and MCCs 1B-91 and 1B-21, respectively. LC breakers serve to protect the RWS pump motors from a fault condition, isolate faults on or downstream of MCC 1B-91 and 1B-21, and serve as a backup if an MCC breaker fails to properly isolate a fault. MCCs 1B-91 and 1B-21 supply Intake Structure lighting, security and heating, as well as power receptacles and RWS equipment.

This modification replaced the existing essential LC breakers with new breakers. The new breakers were upgraded by equipping them with a "state-of-the-art" solid-state trip unit. The old breakers used a mechanical trip unit that was obsolete. The bus bar stabs were increased by one-fourth inch to fit the replacement breakers.

This modification also removed the instantaneous overcurrent trip devices from breakers 1B-903 and 1B-2003 to achieve better breaker coordination. The time delay settings of relays 150/151-312 and 150/151-412 were increased, thus further enhancing coordination efforts.

The IST Program requires testing of solenoid valves SV4934 and SV4935 quarterly. In the event of high Drywell pressure or low-low Reactor water level these solenoid valves serve to maximize water to the stilling basin. Previously, it was necessary to lift leads to test these valves. Two 3-position switches were installed to eliminate the need to lift leads.

Summary of Safety Evaluation

The only purpose of the bus bar modification was to ensure proper fit and to maintain the form, fit, and function of the breakers.

Increasing the time delay settings on relays 150/151-312 and 150/151-412 did not change the form, fit or function of the relays. These relays still serve to protect upstream cable and bus. The fault clearing time of these relays remains fast enough to prevent damage to associated cabling and other devices. The solid-state trip units are more reliable than the previously existing mechanical trip units. The added test switches maintained the separation between division I and division II and non-divisional power for the RWS solenoid valves SV4934 and SV4935. The modification did not alter the function of the circuit; it merely added a testing capability to the circuit for SV4934 and SV4935.

Since the function of these breakers is to mitigate, not prevent an accident, the modification does not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the SAR.

DCP 1469 Perimeter Intrusion Detection System Upgrade

Description and Basis for Change

This modification reconfigured portions of the existing protected area (PA) fencing and installed new fencing parallel to the existing fencing to create a double fence configuration. It also installed new security kiosks, closed circuit television towers, and microwave posts, modified isolation zones, and installed a new raceway around the perimeter to interface with the existing Security System.

Summary of Safety Evaluation

The security modification is in accordance with the guidelines of applicable sections of 10 CFR 73, "Physical Protection of Plants and Materials;" Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage;" ANSI N18.17, "Industrial Security for Nuclear Power Plants;" and the DAEC Security Plan.

None of the equipment involved has any seismic or environmental qualification requirements. There was no change to any DAEC Licensing Analysis as a result of this modification.

The Security System interfaces with Plant systems are through approved isolation devices which are coordinated to prevent the propagation of malfunction. There is no direct interface with any Reactor control system, engineered safety system, or safety supporting system.

DCP 1477 Reactor Building Vent Shaft Housing Cover Modification

Description and Basis for Change

During an engineering walkdown of the Reactor Building Ventilation System, sheet metal damage was found on a section of the Reactor Building Vent Shaft. This modification repaired this and other related damage.

Summary of Safety Evaluation

The safety functions of the Standby Gas Treatment System (SBGT) or Reactor Building Ventilation/Isolation System as described in SAR Section 6.5.3.3 have not been affected by the modification. This modification qualified the Reactor Building vent shaft housing as a Seismic Class I structure as defined in the original design specification. This qualification ensures that the vent shaft housing will maintain its integrity during a seismic event and the Reactor Building Ventilation/Isolation System will be able to perform its design bases functions. Its ability to minimize radioactive releases which may result from an accident is ensured. The modified vent shaft housing provides a higher degree of confidence that the Secondary Containment Isolation System will function as designed to mitigate the consequences of all analyzed transients and accidents in the Reactor Building (as evaluated in Chapter 15 of the SAR). The probability of occurrence or consequences of an accident or malfunction of equipment important to safety as previously analyzed in the SAR has not been increased.

Safety-related material has been used in the construction of the cover frame structure. Although the sheet metal panels and stiffener members have been procured as non-safety-related, which meets the requirements of the original design specification, installation was controlled as a safety-related activity. Functional testing of the vent shaft housing was performed by ensuring that each train of SBGT can, with the Reactor Building isolated, hold the Reactor Building at a sub-atmospheric pressure of 0.25 inches of water as required by the Technical Specifications.

DCP 1478 SRV Tailpipe Pressure Switch Wiring EQ Mod

Description and Basis for Change

The tail pipes of the Safety/Relief Valves (SRVs) and Safety Valves are instrumented with pressure switches. Three pressure switches are mounted on each tail pipe. The actuation of two of the three switches provides an indication that the associated valve is open and also provides an input to the Low-Low Set arming circuitry. The switches are safety-related and environmentally qualified.

This modification installed a seal at the connection to the junction box, to protect the pressure switch wiring and the external connection to the switch from moisture intrusion.

Summary of Safety Evaluation

Although the pressure switches are not specifically described in the SAR, Section 7.6.5 of the SAR describes Low-Low Set and the arming input from the SRV open indication. This indication is provided by the tail pipe pressure switches in a two-out-of-three logic arrangement. The safety function of Low-Low Set was not affected by this modification since the switch actuation will be more reliable under accident conditions. Section 5.2.2 describes the SRV and Safety Valve functions and how they act to mitigate Reactor Pressure Vessel (RPV) overpressurization events. Section 5.4.13 presents the design bases of the Safety/Relief Valves and the safety evaluation of the transients for which the valves must function. The tail pipe pressure switches have no input to the actuation of the Safety Valves or the SRVs that actuate for the Automatic Depressurization System (ADS). Other than the input to Low-Low Set Logic, the switches provide an indication in the Control Room that the associated valve is open. The modification has not impacted the evaluation presented in these sections.

Qualifying the tail pipe pressure switch wiring and wire terminations to E.Q. criteria for moisture intrusion provides a high degree of confidence that the pressure switches will actuate as needed under accident conditions. The safety functions of the Safety Valves and Safety/Relief Valves to mitigate the consequences of an RPV overpressurization transient were not affected by this modification.

Technical Specifications Bases for Section 3.6.D describes the safety function of the Safety Valves, Safety/Relief Valves, and Low-Low Set. The safety functions described were not affected by the modification. The margin of safety has not been reduced.

DCP 1482 Motor Operated Valve (MOV) Thermal Overload Detection

Description and Basis for Change

This modification incorporated an MOV thermal overload (TOL) detector in the control/indication circuitry of each motor operated valve indicated in the Control Room. The modification was made in response to NRC Information Notice 84-13, "Potential Deficiency in Motor-Operated Valve Control Circuits and Annunciation".

Additionally, failure of the sensing circuit in the TOL detector would be indicated. An integral indicating light on the device provides the dual function of indicating proper operation of the device via an energized light, and during alarm conditions, flashing concurrently with the valve position light(s). A spring-return self-test pushbutton of the device provides for on-line testing of the device function/operation by simulating an open TOL contact. The device has been mounted inside the MOV motor control center (MCC) and wired into the control and indication circuitry for the MOV.

Summary of Safety Evaluation

The devices do not affect the operability or reliability of the MOV control circuits and cannot preclude the success of any valve's safety function. None of the accidents analyzed in SAR Chapter 15 can be initiated by the failure of this device.

The device is integrally fused at a rating which is approximately 1/2 the rating of the smallest upstream control circuit fuse. An unlikely fault in the device will blow the device fuse prior to the control fuse. Although a blown fuse may render the indicating lights inoperative, this loss of indication in no way affects the ability of the valve to complete its safety function. The probability of a device fuse opening is no greater than the probability of a blown fuse in existing indication circuitry.

The installation of the device eliminated the possibility of an undetected failure of the MOV due to a thermal overload (TOL). Additionally, the device helps mitigate the consequences of events associated with this condition by providing Control Room Operators with immediate indication of a TOL condition, and the resulting inoperative valve's position.

DCP 1484 Drywell Radiation Monitor CONAX Connector EQ Modification

Description and Basis for Change

This modification sealed the Drywell Radiation Monitor high-range detector's stainless steel flex hose, thus providing an hermetic seal against moisture for the detector connector. Moisture proof CONAX connectors were installed for each detector.

The two Drywell high-range detectors are designed to provide Plant status for accidents that exceed the DBAs discussed in Chapter 15 of the SAR. They are located in containment to "view" a large segment of the containment atmosphere and accurately reflect and monitor accident conditions. The two high-range detectors are not safety-related but are required to be environmentally qualified.

The high-range detectors are mounted close to their associated junction box. Two stainless steel flex hoses run from each detector to a junction box. The wiring runs through conduit to another junction box. The conduit was sealed, so that under accident conditions moisture could not enter into the conduit or through the coaxial cable outer jacket and cause a failure of the high-range detectors.

Summary of Safety Evaluation

The Drywell detectors provide indication only and there are no associated Plant control functions. CONAX connectors have been sufficiently tested at high voltages to ensure that the dielectric strength was sufficient.

The addition of the CONAX connectors has no effect on the detector's response. Calibration of the detectors was performed before and after the modification to verify that the modification did not adversely affect the detectors.

Technical Specifications Bases for Section 3.2.H provides specifications for operations for the containment high-range detectors. The functions described are not affected by the modification. The modification ensures that the installed configuration conforms to the tested configuration.

DCP 1485 Drainage Modifications for Radwaste Spill

Description and Basis for Change

This modification ensures that a postulated radwaste spill would not reach the Cedar River in less than 1,000 days (a requirement of Section 11.2.3.5 of the SAR).

In 1985, storm sewers, catch basins, and associated drainage piping were installed in the yard west of the radwaste Building and LLRPSF, and in the parking lots. This new drainage system created a short-circuit path to the Cedar River for any standing water in the yard area or parking lots.

Section 11.2.3.5 of the SAR describes the results of the analysis of a Non-Mechanistic Failure (NMF) of the liquid radwaste tanks in the radwaste Building. The effect of the NMF was a postulated radwaste spill that would be absorbed into the ground and travel through groundwater reaching the Cedar River in 1,000 days. The drainage system installed in 1985 provided a pathway that would allow a postulated radwaste spill to reach the Cedar River within a few hours.

The SAR assumption of 1,000 days was restored by blocking the flow path to the Cedar River with a new retention structure. Rain water and any postulated radwaste spill can now be first retained and then drained to a new retention pond south of the Plant.

The retention pond is plugged at the south end with a sluice gate and retention structure. Periodically, the gate is opened to drain the retention pond but remains closed the balance of the time. This requires manual action which is administratively controlled by a drainage procedure. The complete scope of work was to install the retention structure, sluice gate, and retention pond.

Summary of Safety Evaluation

The design criteria for the retention structure and sluice gate meets the seismic requirements for equipment installed in the yard area. The sluice gate remains closed during a seismic event and does not need to be operated.

The retention pond and structure have not been designed for the effects of tornadoes and tornado missiles. However, the Radwaste Building design is such that a tornado or tornado missile would not cause a radwaste liquid spill. The combined effects of a seismic event causing a radwaste liquid spill and a tornado destroying the retention pond are highly improbable. The sluice gate and retention structure have, however, been installed underground eliminating the possibility that the sluice gate or retention structure could be struck by a tornado missile during normal operation.

Installation of the sluice gate and retention structure will not affect the method of operation or safety function of any other equipment. The retention pond has sufficient volume to contain normal rainfall in addition to tank contents. Technical Specifications regulate radioactive liquid effluent to maintain the concentration of discharges below the limit of 10 CFR 20. Following this modification, the assumptions of the original analysis of radwaste spills remains valid. Therefore, the margin of safety with respect to 10 CFR 20 limits and as defined in the Technical Specifications is not reduced.

DCP 1489 Transformer Annunciator Modification

Description and Basis for Change

This modification involved the four Main Transformers, and the Auxiliary, Startup, and Standby Transformers and improved the transformer annunciator/alarm system by removing alarm masking in the Control Room. Prior to this modification, multiple alarms were wired as a single input to the Control Room annunciator window associated with the transformers. This did not provide reflash capabilities to make Operators aware of any additional alarms after the first was acknowledged. The ability of non-critical alarms to mask critical alarms such as transformer cooling parameters was an operational and safety hazard which could result in equipment damage and loss of generation of electricity.

Existing annunciator hardware at the single phase Main Transformers, Auxiliary, Startup & Standby Transformers was removed and a self-contained multi-microprocessor based twelve point LED display annunciator module was installed at each transformer control cabinet. Transformer alarm inputs were wired to the modules, and existing circuits to the Control Room annunciator windows were utilized. For the Main Transformers an additional "reflash" alarm module was required to accept the output of each individual alarm module.

Summary of Safety Evaluation

The primary operation and function of the Auxiliary AC Power System as described in the SAR has not been altered. Only annunciator and alarm functions not critical to the shutdown of the Plant have been affected. Control functions associated with the transfer of the Startup and Standby transformers as outlined in the SAR and Technical Specifications were not affected. The upgraded monitoring capabilities increase Operator awareness of transformer status, thus decreasing the probability of an accident.

DCP 1493 DCRDR Phase 4 Modifications

Description and Basis for Change

General design modifications were made to the DAEC Control Room, remote shutdown panel, and related systems to correct associated human engineering deficiencies identified in the Detailed Control Room Design Review (DCRDR) summary report. These modifications resulted in a significant reduction in the probability of human error and improved the Operator's ability to respond during emergency events and abnormal operational conditions.

At Control Room Panel 1C07, pressure indication was added for Turbine Lube Oil and the Electro-Hydraulic Control (EHC) System. Several existing indicators and controls were rearranged. A remote control switch was added for the Reactor Feed Pump Stuffing Box Transfer Pump. The existing local switch was modified to allow two-switch operation. The Turbine thrust bearing wear detector test buttons, test lights, and position indicator were removed. Several handswitches and indicating lights were also rearranged.

Controllers for valves that fail open were modified so that the same Operator action is required to manually open these valves as is required for fail-closed valves (i.e., in all cases turning the manual control clockwise will open the valve).

Modifications were made to annunciator windows, to produce a "blackboard" effect in which no alarms are illuminated during normal operations. The seismic annunciator horn in panel 1C-35 was replaced with a 24 VDC horn identical to the existing Control Room annunciator horns. This allows volume adjustment. A new power supply identical to the power supplies for existing Control Room annunciator horns was installed.

To eliminate Rod Block Monitor (RBM) nuisance alarms, a time delay before annunciation was implemented. This delay is introduced into the annunciator circuit only. A time delay relay was also added to the APRM annunciator circuit to eliminate Average Power Range Monitor (APRM) upscale nuisance alarms at high power.

Modifications to the Rod Worth Minimizer (RWM) had been previously accomplished using temporary lifted leads. A permanent modification was accomplished by removing panel wiring, and abandoning in place relays, alarm cards, and cables in the RWM circuit. This modification is designed to enforce rod programming through all power levels, and eliminates the ability to bypass the RWM automatically at 30%.

A pressure indicator was installed to indicate RHR shutdown cooling suction piping pressure so that Operators can ensure that sufficient pump suction head is present to avoid cavitating the pumps.

Several other changes were made including the repositioning of indicating lights, installation of replacement scales and rearrangement of indicator banks. Selected Control Room fluorescent lights were supplied with battery backup emergency power capable of supplying the lights for 8 hours. A portable oxygen monitor was added to the Control Room to monitor possible changes in oxygen level which might occur if the cable spreading room CARDOX system is initiated.

Summary of Safety Evaluation

The design bases for this modification were based on the SAR, Human Factors Considerations of the Control Room, and the DCRDR summary report. Instruments were arranged and grouped according to system and in the order as recommended in the DCRDR summary report. Indicator scales were changed to allow the Operator to read the instruments more accurately and quickly. The modifications improved the ability of the Operator to interpret Plant operating conditions and aid in the response to those conditions.

The accuracy of new instruments in the Control Room is equal to or better than existing instruments. Modifications to eliminate nuisance alarms by the addition of time delay relays or operational interlocks to existing annunciator circuits did not affect the function of any automatic actuation or safety interlock circuit.

Electrical separation in accordance with IEEE Std 279 was maintained between channels in accordance with the original design. This includes power and control cables, and wiring in the main Control Room panels.

The new equipment is not required to operate in a harsh environment. Installation did not degrade any environmental conditions for existing equipment.

The weight of the new Control Room instruments have been evaluated and determined not to adversely affect the seismic qualification of the control panels.

The method of operation and safety function of equipment important to safety were not changed. The elimination of the RWM bypass capability reduces possible failure modes. The modifications to Control Room lighting provides additional capability to light the Control Room during power failure without loss of any existing capability.

DCP 1494 Lifting Lugs for RHR Full Flow Test Valve

Description and Basis for Change

During maintenance of the RHR full flow test valve, the motor operator could not be removed because of space limitations. This modification added two lifting lugs, angle iron and a channel on an imbedded channel directly over the valve. The lugs are used to perform a two point lift of the motor operator.

Summary of Safety Evaluation

The lifting lug configuration is independent of any safety-related system and has been attached to an original embedded channel designed to carry loads of any future additional attachments (lugs, pipe supports, hangers, etc.). The maximum load on the lug is well below the design load. The possibility of a new malfunction of equipment important to safety has not been created.

The lifting lug configuration has been attached such that in a seismic event the loads seen at the lug configuration will be considerably less than the lifting loads from the motor operator. The seismic design will prevent the structure from falling on any safety-related equipment. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR has not been increased.

The material for the lifting lug configuration is certified material with heat traceability. The steel was welded using approved Iowa Electric welding procedures. These welds have been sized per original design calculation and are within allowable stresses.

Installing the new lifting lug configuration allows for safer and faster removal of the motor operator. The design of the lifting lugs satisfies NUREG 0612 and AISC standards.

DCP 1496 RWS Pumps Restart Logic Modification

Description and Basis for Change

The RWS pumps restart logic was modified to enforce a two minute time delay prior to restart of any RWS pump after manual or automatic tripping. The time delay allows the tripped RWS pump to coastdown to standstill, thus avoiding an automatic or manual restart while it is either coasting down or running in the reverse direction due to drainage of water through the column.

The RWS pumps RED and GREEN indicating lights on panel 1C06 associated with the selector switches were changed to WHITE indicating lights and wired to indicate "PUMP AVAILABLE" status.

The modifications do not affect any instrument requirements directly or indirectly as defined by the Technical Specifications. Surveillance requirements, limiting conditions for operation, and basis statements for the systems covered by these sections are not affected.

Summary of Safety Evaluation

The modifications were designed and installed using design criteria that are the same or more stringent than the design criteria used in the original Plant design. New relays were seismically mounted. The restart logic modification which introduced a two-minute delay in starting does not violate the original assumptions for the probability or consequences of accidents analyzed in the SAR. A system calculation has determined that sufficient inventory exists to allow the RWS System to survive a complete loss of river water for at least six minutes without any adverse effects. The logic changes eliminate the possibility of damage to the pump and drive motor due to restart while coasting down or while rotating in the opposite direction.

The use of WHITE indicating lights at the RWS pump selector switches to indicate that the pump is "available" for restart enhances the human factors design considerations for the system. The function and method of operation of the RWS pumps have been improved. The margin of safety for the operation of RWS pumps has not been reduced because additional protection is provided to the RWS pumps by this modification. No other equipment or systems were affected by this modification. No new accident scenarios resulting from this modification were identified.

DCP 1499 The Removal Of Equipment From Unused Residual Heat Removal (RHR) Modes

Description and Basis for Change

The RHR system was originally designed to provide multiple modes of operation for both essential and supplemental cooling of the Reactor Core, Suppression Chamber (torus), Containment and Fuel Pool.

The steam condensing mode of RHR was originally designed to be used when the Reactor Vessel is isolated from the Main Condenser and the Main Steam Relief Valves are blowing down steam to the Suppression Pool. The steam condensing mode had not been used since Plant startup and had been removed.

The head spray feature of the shutdown cooling mode of the RHR System was decommissioned by a previous modification. It was determined to be a source of extensive maintenance and had the potential to violate the rate of cooldown as listed in the Technical Specifications. The Primary Containment isolation valves were closed and de-energized.

HPCI was isolated from RHR by removing the piping tee connection in the HPCI steam line and replacing it with an elbow. The RHR piping was cut and capped.

Summary of Safety Evaluation

Since none of the accidents previously evaluated in the SAR could be initiated by a failure of a portion of the RHR system, the modification of unused portions of RHR has not increased the probability of an accident to occur.

The removal of these modes of RHR and associated equipment have not affected the operability of the Low Pressure Coolant Injection (LPCI) and Containment Spray modes which are the Technical Specifications required portions of RHR that mitigate the consequences of an accident. The HPCI System operability has not been affected by this modification. The remaining modes of RHR were unaffected by the removal of this equipment.

The removal of the steam condensing and head spray modes and equipment reduced the probability of occurrence of a malfunction of equipment since the equipment which was removed interfaced with the HPCI and Primary Containment Systems. The steam condensing mode of RHR interfaces with the HPCI steam supply and the head spray mode interfaces with Primary Containment. Both of these modes of RHR were unused as they were considered supplemental and were not required for the safe shutdown of the Reactor.

DCP 1500 Lifting Lugs for Various MOVs

Description and Basis for Change

Lifting lugs were installed near various MOVs to allow removal of the motor operators for maintenance.

The lugs were welded to plates anchored in the walls and ceiling as required. Lugs were also attached to existing embedded steel and supports as necessary for each valve.

Summary of Safety Evaluation

The lifting lugs are independent from any safety-related systems. The design of the RHR and Core Spray Systems (for which the motor operators are used) was not affected in any manner, nor was safety-related equipment in the vicinity affected. The lugs have been seismically designed to prevent the lifting structures from falling on any safety-related equipment.

The lifting lugs were attached to anchored plates, existing embedded steel, and/or existing supports. The addition of lifting lugs has not changed the original design intent of the concrete, embedded steel, or supports. Although none of the motor operators are considered a heavy load (ie, $\geq$1000 lbs.), the guidelines of NUREG 0612, Control of Heavy Loads at Nuclear Power Plants, have been met.

DCP 1506 RWCU Return Line Leak Detection

Description and Basis for Change

This modification installed a high ambient temperature monitoring system which would detect a RWCU System piping leak in a previously unmonitored area. Four dual element thermocouples were installed to monitor the ambient temperature in the area above the TIP room on the first floor of the Reactor Building.

The thermocouple pairs operate identically to the existing ambient leak detection system. The outputs from the Division I temperature switches have been tied into the Division I RWCU isolation valve logic and initiate an isolation signal to valve MO2700 (inboard supply). The outputs from the Division II temperature switches have been tied into the Division II RWCU isolation valve logic and initiate an isolation signal to valve MO2701 (outboard supply) and MO2740 (outboard return). Each thermocouple location has a redundant set of elements to provide divisional components that are capable of providing the isolation signal. The temperature switches associated with these components have been provided with burnout protection to ensure isolation if the cable or thermocouple are damaged.

The temperature switch setpoints have been set at a value that is high enough to avoid spurious isolation, but low enough to provide assurance that a minimal leak in this portion of piping is detected.

Summary of Safety Evaluation

The ambient temperature monitoring capability is designed to isolate leaking portions of the RWCU system and maintain the design physical barriers, and/or limit the reduction of coolant inventory and subsequent radioactive release. The modifications have no effect on the related accidents evaluated in the SAR.

The quality of the components added by this modification meets original design specification, including seismic, separation, and EQ. All electrical components were installed as safety-related.

The failure mechanisms for the affected important-to-safety equipment were not changed from the original analysis. The TEs function as the original design intended. Because this modification does not affect system performance criteria (flow, piping design, function, etc.) there is no mechanism for an accident of another type. The functional design of the affected systems has not been changed by this modification. The equipment added performs a monitoring function and only adds another initiation signal to a pre-existing logic string to isolate RWCU.

DCP 1508 Replace RWS Pump 1P-117D

Description and Basis for Change

The RWS pumps are routinely subjected to less than ideal water conditions. This had caused degradation to their performance which required the pumps to be refurbished or rebuilt on a regular basis. The purpose of this modification was to install a replacement pump assembly into the 'D' RWS Pump; install a vibration probe at the pump bowl assembly; install Crane type packing onto the newly installed pump; and enlarge the discharge head drain port to one inch and provide drain piping back to the sump.

Summary of Safety Evaluation

The replacement of the 'D' RWS pump had no effect on the overall ability of the RWS System, in combination with the ESW and RHRSW Systems, to perform design safety functions. The replacement pump was procured to meet or exceed all performance requirements contained in the original specification for the RWS pumps. The installation of the vibration probe added an inconsequential amount of mass to the overall pump assembly. It will not degrade the pumps' ability to perform its safety-related function. The installation of the John Crane type packing had the effect of reducing the packing bypass leakage rate over time, and extends the life of certain consumable components, thus increasing the overall reliability and maintainability of the pump. The discharge head drain port was enlarged but this has been analyzed and determined to have no detrimental affect on the structural integrity of the pump assembly. The overall effect of these modifications is to enhance the RWS pump's long term reliability, and therefore increase availability for the mitigation of any accidents evaluated in the SAR.

DCP 1509 CAV-LPRM Computer Input Modification

Description and Basis for Change

A previous modification removed 2 Local Power Range Monitors (LPRMs) in order to install two Double Cantilever Beam (DCB) sensors for crack arrest verification. Part of the original intent of that modification was for the computer to read the removed LPRMs as failed, thereby forcing it to provide substitute values when doing its P1 calculation. The logic for those 2 LPRMs was disabled, but a noise-induced floating voltage across the Plant computer inputs resulted in a false signal to the PPC. The extended modification was installed to ground a 0.0 volt signal to the computer, which caused the signal to be viewed as a failed LPRM, and be dropped from the P1 calculation, as intended.

This modification made permanent the grounding of the computer inputs from the 2 LPRMs.

Summary of Safety Evaluation

No safety systems, or design bases were affected. The PPC has no safety margin defined in the Technical Specifications. Therefore, the Plant margin of safety as defined in the Technical Specifications Basis has not been reduced. This modification only affects inputs to the PPC from LPRMs which have already been removed. The PPC does not initiate any actions.

DCP 1512 SCRAM Relay Replacements

Description and Basis for Change

The contactors in the Reactor Auto Scram Trip Logic and the Reactor Manual Scram Trip Logic relays were replaced. The contactors used previously in the system are no longer manufactured and the installed units were reaching the recommended service life. Upgraded contactors were used.

Contactors and auxiliary contacts were mounted in the existing enclosures in the Control Room.

This modification also provided a voltage sensing device for detecting a closed contact for the back-up scram valves. This ensures that an inadvertent scram does not occur from a stuck contact.

The annunciator windows for this back-up scram half scram will alarm each time there is a half-scram. When the half-scram is cleared, the associated back-up scram half scram annunciator should also clear. If this annunciator window does not clear when the half-scram is cleared, this indicates that a contact for the back-up scram valve is still closed and a half-scram for the back-up scram valve still exists.

The voltage sensing device is not safety-related. It has been connected to a safety-related circuit. Double isolation fuses are used to isolate this device from the safety-related circuit.

Summary of Safety Evaluation

None of the accidents previously evaluated in the SAR were affected by this modification. The replacement contactors were seismically qualified. These replacement contactors are not a like-for-like replacement; they are an upgrade. The contactors are located in a mild environment; therefore, they are not required to be environmentally qualified. The separation criteria has not been changed. The power supply has not been changed.

The specification of setpoints, minimum number of trip systems and sensor contact response time measurements are discussed in the Technical Specifications. The response time and reliability of the new contactors meet or exceed that of the original contactors. The interfaces of the RPS equipment have not been changed.

The new contactors have the same failure modes as the previously existing contactors. The RPS will not fail in any new way as a result of this modification. Since the design or interfaces did not change, the replacement contactors could not cause a malfunction of a different type than evaluated previously in the SAR.

The addition of the voltage sensing device to the back-up scram circuit did not change the function of the circuit. Isolation fuses ensure that the operation of the circuit will not be affected by a fault on the non-safety related portion of the circuit. The fuses have been sized to allow the voltage-sensing devices to function and to blow prior to having any effect on the circuit during any fault condition.

DCP 1517 Control Building Chiller Enhancements

Description and Basis for Change

Modifications were made to the Control Building Chillers to allow for performance monitoring, testing/calibration of safety switches, and ease of maintenance. Components used for control of the hot gas bypass feature of the 'B' Chiller were obsolete and unlike those used on the 'A' Chiller. The 'B' Chiller was modified to provide easier maintenance, improved performance monitoring and to upgrade the obsolete hot gas bypass controls.

Pressure points were provided for the chilled water system inlet and outlet to the evaporator. These pressure points were located such that they could be used to drain and vent the chilled water side of the evaporator as needed for maintenance and operation. Isolation valves and connections to the sensing lines were added to facilitate the periodic testing and calibration of pressure gauges and switches without removing the refrigerant charge. A flip-pac and reed switch kit was installed in the 'B' Chiller's hot gas bypass control system. The reed switch senses low compressor load. Several timing relays were replaced.

In addition, equipment IDs were added to skid-mounted equipment which had not been labeled. The anti-recycle timer on both chillers was replaced.

Summary of Safety Evaluation

The Control Building Chillers and the closed loop chilled water system are not accident initiators. They control the ambient temperatures in the Control Building. This provides for Operator comfort in the Control Room and precludes equipment degradation due to overheating during both normal and Control Room isolation conditions.

The pressure boundary modifications of the Control Building Chiller and the Chilled Water System meet the requirements for Seismic Category I. The electrical control modifications that perform active functions meet the requirements for Class 1E. Utilizing a Class 1E flip-pac and reed switch system for the hot gas bypass control improves the reliability of the Control Building Chiller. The function and operation of the Control Building Chiller have not been affected. Therefore, the consequence of an accident previously evaluated in the SAR has not been increased.

The ability to do predictive and preventive maintenance has improved the Control Building Chiller reliability.

The replacement of the timer relays has not created the possibility of a new malfunction. The relays are highly reliable, safety-grade with no new failure modes.

DCP 1518 Feedwater Control Enhancements

Description and Basis for Change

Backup air supplies were provided for valves in the Feedwater and Condensate Demineralizer Systems that were formerly supplied only by the Instrument Air System. The backup air supplies were designed to allow approximately 30 minutes of instrument/valve operation after loss of Instrument Air. The Feedwater Regulating Valves and Minimum Flow Valves were provided with appropriately sized line pressure accumulators. The Condensate Demineralizer valves were provided with compressed air cylinders as the back-up air supply source. This provided redundancy to allow continued operation of the components in the event of a loss of Instrument Air, thus enhancing DAEC Scram Frequency Reduction efforts.

The Feedwater Pumps Minimum Flow Valves were replaced with more restrictive valves designed to reduce the excessive flow rate when the valve is fully open. The existing valves were replaced with valves manufactured for this severe service. In addition, the existing discharge piping from these valves was replaced with like-for-like carbon steel components. The exception was that the blind flange material was changed to stainless steel which is much less susceptible to erosion and impingement damage.

Several electrical modifications were also implemented.

The logic of the Turbine Building Instrument Air isolation valves was upgraded. Previously, these valves failed in the closed position and isolated the Instrument Air Supply. This resulted in the Feedwater Minimum Flow Valves and Feedwater Regulating Valves failing closed on a loss of power. This modification allows the control valves to "fail open" on loss of power.

Previously, upon a loss and return of power to the condensate demineralizer logic, the demineralizers would go into "hold", isolating the suction source for the Feedwater Pumps. This automatic "hold" sequence was eliminated to reduced the possibility of a Reactor trip due to loss of feedwater flow.

The "A" and "B" Feedwater Pumps' low suction pressure time delay-to-trip setpoints were staggered. This reduced the likelihood of a Reactor scram on loss of feedwater flow in the event that one Feedwater Pump can maintain operation under low suction pressure conditions for two pump operation, e.g., loss of one Condensate Pump.

The power supply for each Feedwater Pump's Minimum Flow Valve has been separated to ensure that a loss of power to one division of Instrument AC will not cause both minimum flow valves to fail open, causing a low suction pressure trip. Previously, a loss of one division of Instrument AC would have caused both Circulating Water Pumps' discharge valves to close, thus causing the pumps to trip. This would result in a loss of Condenser vacuum, Turbine trip and scram. The power supply for these valves' circuitry has been separated to prevent this.

Summary of Safety Evaluation

Backup air supplies provide greater reliability for the components and do not increase the probability of a Feedwater Controller failure or loss of all feedwater. Check valves were installed between each air cylinder and the main Instrument Air line to prevent an air cylinder/air regulator failure from disabling the Instrument Air System.

The cylinder locations were evaluated to ensure that a seismically induced support failure would not adversely affect equipment important to safety. Each cylinder is adequately supported to preclude inadvertent toppling that could lead to a broken shutoff valve and subsequent air discharge.

The replacement of the minimum flow valves will enhance system operation during events when the valve fails open and during normal operation. The replacement of these valves and downstream piping did not increase the probability of a loss of feedwater event, nor did it increase the probability of a loss of condenser vacuum. The new valves are more flow restrictive, thereby decreasing the likelihood of a Feedwater Pump trip if the valve failed open. The piping replacement is the same configuration with the exception of the stainless steel blind flange which is more resistant to erosion and impingement damage.

The modification of the Instrument Air isolation valve control logic improved the reliability of the Instrument Air System. This ensures a more reliable air supply to the Feedwater and Condensate System valves on a loss of power. Even a total failure of the compressed air system will not affect the operation of safety-related equipment because all safety-related components serviced by the Instrument Air System have either backup compressors or accumulators. The addition of backup supplies for non-safety-related components serviced by Instrument Air created a more reliable air supply to those individual components than had previously existed.

The change of the Feedwater Pump low suction pressure trip time delay settings allow for a staggered trip of the pumps. In the case of one Condensate Pump tripping or another event that would allow one Feedwater Pump operation, this change would allow continued Reactor operation (at reduced power) without a Plant trip.

The separation of the power supplies for the Feedwater Pump Minimum Flow Valves and their flow indicating controllers reduced the likelihood of a loss of feedwater event because both valves and both controllers would not be subject to a single point failure upon a loss of power. This change enhanced the Feedwater System's ability to operate during loss of power events.

The separation of power supplies for the Circulating Water Pump discharge valves reduced the likelihood of a loss of circulating water event. The Circulating Water System is better able to operate during a loss of one division of Instrument AC. This separation allows at least one Circulating Water Pump to remain in operation for maintenance of condenser vacuum and possible prevention of a Turbine trip.

The modifications did not affect any systems that serve to mitigate an accident. Therefore, the consequences of an accident previously evaluated in the SAR was not increased. The probability of abnormal events or malfunctions of equipment important to safety as listed in the NSOA and as analyzed in the SAR are not increased.

DCP 1519 <u>Turbine Supervisory Instrumentation (TSI)/Scram Frequency Reduction</u>

<u>Description and Basis for Change</u>

The purpose of this modification was to reduce Turbine trip vulnerability due to single sensor failure. This was accomplished by adding redundant sensors, connecting them in a fault tolerant arrangement (such as 2-out-of-3 logic), or by eliminating automatic Turbine trips. This modification resulted from DAEC Scram Frequency Reduction group recommendations.

The following five Turbine trip circuits were modified:

- LP Exhaust Hood High Temperature
- Turbine-Generator High Vibration
- Thrust Bearing Wear Detector (TBWD)
- Low Bearing Oil Pressure via TBWD
- Moisture Separator and Reheater (MSR) High Level

The low pressure Turbine Exhaust Hood high temperature alarm and trip circuits were modified to eliminate the Turbine trip at 225 degrees F. The associated annunciators were also modified.

The Turbine-Generator Bearings were each monitored by a vibration channel in the Turbine Supervisory Instrument (TSI) cabinet. Any channel would activate a Turbine trip if the vibration at any of the bearings exceeded 10 mils. This trip signal was removed. Several annunciator windows were also relocated.

The Thrust Bearing Wear Detection System was replaced with a proximity probe system. The Thrust Bearing Wear Detection System continuously detected the axial position of the Turbine shaft with respect to the Thrust Bearing casing using a pilot valve/pressure switch arrangement. If the wear on the Thrust Bearing exceeded the setpoint, the Turbine would trip. The Proximity Probe System consists of three independent Thrust Position Monitoring Channels. The channels are arranged in a 2-out-of-3 trip configuration and are connected to the Turbine master trip bus input. An associated annunciator window was also added.

The Loss of Bearing Oil Pressure Trip was modified to a 2 out of 3 trip configuration. The setpoints on the new pressure switches remain unchanged. The trip signal from the bearing oil pressure switches was connected in series with a time delay pickup relay which provides a delay of 2 seconds before tripping the Turbine.

A Moisture Separator and Reheater high level trip occurred when the Moisture Separator and Reheater (MSR) water level increases to the trip setpoint. Formerly, the high level trip signals and PPC inputs were supplied directly from one level switch on each MSR. The existing level switch on each MSR was replaced with three level switches. The Turbine trip logic from each MSR has been arranged in a 2-out-of-3 configuration and connected to the existing Turbine master trip bus. The level switch indication circuit provides position indication of each level switch at a local control panel.

Summary of Safety Evaluation

This modification reduced Turbine trip vulnerability by adding redundant sensors or by eliminating automatic Turbine trips. The addition of redundant 2 out of 3 sensors to the TBWD, MSR level switches, and low bearing oil pressure switches reduces the vulnerability of Turbine trips due to single sensor failure or spurious actuation. The high vibration and exhaust hood high temperature trips were removed and replaced with Control Room annunciation. This allows Operators to take appropriate action while further reducing Turbine trip vulnerability. These modifications helped decrease the probability of a spurious Turbine trip resulting in a nuclear system pressure increase.

The radiological consequences of the accidents evaluated in the SAR are not altered by the modifications. The modifications were to Turbine instrumentation and controls, annunciators, and the PPC. The modification to these systems reduce Turbine trip vulnerability due to single sensor failure. The modifications do not alter any assumptions previously made evaluating the radiological consequences of an accident described in the SAR, nor does it degrade or prevent actions described or assumed in an accident discussed in the SAR.

The SAR evaluates the consequence of Turbine-Generator missiles. The Reactor Building walls and slabs and control Building walls and slabs are designed to withstand the loading due to missiles generated by the failure of the Turbine-Generator. The modifications to the Turbine trip system have not altered the existing worst-case Turbine missile analysis, and hence have not affected existing equipment important to safety. The consequences of a Turbine-Generator missile remain unchanged with respect to the existing analysis.

This modification does not affect the margin of safety as defined in the basis for any Technical Specifications because no margin of safety is defined in the Technical Specifications.

DCP 1520 Installation of Keylock Switches and Banana Jack Adapters to Facilitate Surveillances

Description and Basis for Change

Some of the DAEC Surveillance Test Procedures (STPs) formerly required lifting leads, pulling fuses or installing jumpers in order to accomplish the desired test. These practices greatly increased the possibility of poor connections and human error. The requirement to lift the leads or pull the fuses was facilitated by the use of keylock switches. The keys for these switches are controlled by the Operations Department and are only removable in the normal (closed) position. As an extra precaution against leaving the switch in the wrong position, a light was provided to indicate if the switch is not in the normal position.

Some STPs required the use of jumpers on relays. This need was eliminated by installing banana jacks in place of the existing terminal screws.

Summary of Safety Evaluation

The addition of the keylock bypass switches provides a more reliable method of opening the circuit for surveillance testing. It does not affect the function or operation of any system. The keylock bypass switches meet the same or better design, material and construction standards that apply to the previous systems and Control Room panels. The switches and indicating light are seismically qualified.

The terminal board screws were replaced with banana jacks. This did not change the function or operation of any system. The probability of occurrence of an accident or a malfunction of equipment important to safety that has been previously evaluated in the SAR has not been increased. These modifications did not change, degrade or prevent any actions described in the SAR.

The additional weight of the switches was analyzed to show that they did not adversely affect the seismic qualification of the panels in which they were installed. Separation criteria were also met.

DCP 1521 Miscellaneous Fire Protection Piping Modifications

Description and Basis for Change

Several independent modifications were made to improve fire protection capabilities. The existing deluge system for the main transformers was extended to provide coverage underneath the isophase bus ducts. The existing sprinkler system in the LLRPSF mask cleaning room was extended to provide coverage underneath oversized HVAC ducts. A hose reel station was installed in a stairwell of the Administration Building to provide coverage for the hot chemistry lab. An odorizer was added to the CO_2 line for the Cable Spreading Room CARDOX System to alert Control Room personnel in the event of system actuation.

Summary of Safety Evaluation

No systems or equipment other than fire suppression systems were modified. Interaction with fire suppression systems is not assumed or required for any of the accidents previously evaluated in the SAR or NSOA. Therefore, the modifications will not alter the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR.

The sprinkler/deluge and hose reel piping does not protect equipment important to safety. No work was performed in areas that contain equipment important to safety. Systems modified are not required to perform a safety function or to support a system which is needed to perform a safety function. Therefore, the consequences of an accident previously evaluated in the SAR is not increased.

Modified systems meet the same codes and standards and are located in the same areas as existing suppression systems. The CO_2 odorizer unit is a passive device. It is identical to other installed units and does not adversely affect CARDOX System operation. The scent added by the odorizer unit will provide Control Room Operators with an additional means to detect the presence of CO_2. This will enhance protection against loss of Control Room habitability. Therefore, the possibility for an accident of a different type is not created.

Requirements for fire protection systems are covered in Technical Specifications Section 3.13. None of the water systems affected by this modification are required by the Technical Specifications.

DCP 1522 Turbine-Generator Fire Suppression - Phase II

Description and Basis for Change

This modification provided fire suppression capability by extending the existing sprinkler system to provide coverage of the lube oil supply and return lines for the Turbine bearings beneath the Turbine skirt. Piping, pipe supports, sprinkler heads, heat detectors, conduit and cabling were installed.

Summary of Safety Evaluation

Previously evaluated accidents are described in the SAR and the NSOA. The probability of occurrence of these accidents is based on initial conditions and assumptions which do not depend on the end use of or interactions with fire protection systems. No systems or equipment other than the sprinkler system were modified and the sprinkler piping which was modified does not protect equipment important to safety. No work was performed in areas that contain equipment important to safety. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR has not been increased.

Fire Protection capability has been improved. The modified system meets the same codes and standards and is located in the same areas as existing equipment. The modification will not degrade or prevent any action assumed to occur in the SAR to mitigate the consequences of a malfunction of equipment important to safety.

DCP 1523 RPS Power Supply Upgrade

Description and Basis for Change

Modifications as described below were made to the RPS power supply system. These changes significantly reduced the probability of inadvertent scrams arising from the operation of the RPS power supply, ensured the continued reliability of critical RPS power supply components, and enhanced the maintainability of the RPS power supply.

The RPS power supply transfer scheme was modified so that no single failure would lead to the loss of both RPS buses. The existing RPS power supply transfer switch and the three indicating lights that monitored the RPS power supply were removed from Control Room panel 1C16. A transfer switch for the RPS 'A' bus, with indicating lights to monitor normal and alternate power supplies to the RPS 'A' bus, was added to panel 1C15. A transfer switch for the RPS 'B' bus, with indicating lights to monitor normal and alternate power supplies to the RPS 'B' bus, was added to panel 1C17.

The operating coils of the four RPS power supply contactors were nearing the end of their operating life and were replaced with new coils as General Electric Service Information Letter (SIL) 508 recommended.

The Main Steam Line (MSL) radiation monitor isolation and trip functions which close the Main Steam Isolation Valves (MSIVs) on high radiation were removed. This Technical Specifications revision was implemented by Amendment No. 182.

Summary of Safety Evaluation

General Electric Service Information Letters SIL-508 and SIL-445 provided a basis for this modification.

The modifications implemented by this change do not affect the operation of any Plant system other than the RPS. No other equipment was affected by this modification.

The function and method of operation of RPS has been improved to comply with the original design intent. The design and installation met design basis and safety parameters. The RPS power supply system has been enhanced by the control switch replacement since it has eliminated the possibility that a single control switch failure could result in the loss of both trains of RPS.

The weight of new relays, seismically qualified switches and indicating lights is very small compared to the panel weights, so the seismic qualification of the Control Room panels remains unaffected.

The revised components meet or exceed the requirements originally established for their function. The control logic modifications have no impact on the previously analyzed consequences of an RPS malfunction. NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor" was previously reviewed by the NRC and found acceptable.

DCP 1524 Containment Hard Vent Modification

Description and Basis for Change

NRC Generic Letter 89-16 requested that each licensee with a Mark I containment install a hardened wetwell vent. By letter, NG-89-2886, Iowa Electric agreed to the installation of such a vent.

The hard vent system performs two different functions depending upon the type and severity of the event for which it is relied upon to operate: Primary Containment integrity and Primary Containment venting.

An 8" branch connection was installed on the 18" line between the existing inboard and outboard torus purge and vent isolation valves. New 8" piping off of this connection was routed into the condenser bay where it ties into the Steam Packing Exhauster discharge line which leads to an elevated release point provided by the Offgas Stack. A new outboard Primary Containment isolation valve was installed in the new 8" piping in the NE Corner Room. Both the inboard torus purge and vent isolation valve and this new outboard isolation valve are required to be opened to use the hard vent path.

An AC-independent design was provided to ensure operability of the hard vent path during a station blackout event. An accumulator was installed in the air supply line to provide a dedicated source of air for operation of these valves. A check valve was installed in the air line between the accumulator and Instrument Air System to prevent loss of air due to back leakage during a station blackout event.

An 8" rupture disk was installed. The rupture disk will prevent the opening of a vent path from the Primary Containment directly to the environment unless the Primary Containment pressure limit of 53 psig is threatened. During a DBA, the rupture disk will provide a zero-leakage barrier between the Primary Containment and the environment to prevent any small amount of leakage from bypassing the Secondary Containment.

Operation of the hard vent is possible from the Control Room using a handswitch in conjunction with another keylock handswitch. This gives the Control Room Operator the capability to open the vent valve using DC power during a station blackout event.

This modification has not been site closed. It is included in this report because the hard vent portion of the modification has been installed and has been operationally released.

Summary of Safety Evaluation

The hard vent system modifications are restricted to the condenser bay in the Turbine Building, the NE corner room in the Reactor Building, and the main Control Room. None of these areas contain piping or equipment associated with accidents discussed in Chapter 15 of the SAR. Hard pipe vent system components are sufficiently separated from the Reactor Vessel and associated piping, control rod drive hydraulics and controls, main steam lines and refueling equipment to make any significant interaction impossible.

Establishment of Primary Containment integrity is a required safety action to protect against the release of radioactive materials to the environment during specified DBAs. The hard vent system is designed as a passive isolation barrier that ensures the integrity of the Primary Containment and prevents Primary Containment leakage from bypassing the Secondary Containment and standby gas treatment system through the hard vent path during a DBA. The hard vent will be a normally closed, sealed closed barrier, capable of performing as an effective Primary Containment barrier consistent with Primary Containment design requirements. Likewise, the volume formed by the rupture disk and associated piping downstream of the vent valve performs a passive function to collect any leakage, no matter how minute, that may pass through the valve during a DBA.

A failure of electrical power or air pressure does not prevent the hard vent system from performing its isolation function.

The Primary Containment isolation function associated with the inboard torus purge and vent isolation valve is unaffected by the hard vent modification when the new keylock handswitch is in the NORMAL position. The handswitch would be placed in the BYPASS position only when necessary to prevent a loss of integrity due to rupture of the Primary Containment as dictated by approved emergency operating procedures. Bypassing of Primary Containment isolation functions associated with the inboard torus purge and vent isolation valve will only occur in response to beyond design basis events, and even then, only when necessary to protect the integrity of the Primary Containment as controlled by Emergency Operating Procedures.

The key for the keylock handswitch is administratively controlled by the Operations Shift Supervisor in the Control Room. In addition, fuses are removed from the vent valve's control power to prevent inadvertent operation. These fuses must be installed prior to valve operation.

Electrical components and cables installed for the hard vent system are divisionally separated. In this configuration, the probability of a hot short or other electrical fault causing the inadvertent opening of both valves during an accident is no greater than any other electrical failure which would prevent an accident mitigation system from failing to perform its accident mitigation function.

The valves are designed to fail closed on loss of air and therefore are fail safe. Use of a common supply of air does not invalidate any assumptions or requirements pertaining to redundancy of Primary Containment isolation valves. No common component failure can cause both valves to open or to fail open.

The hard vent system is designed to withstand the most severe pressure and temperature that would be expected during a design basis LOCA. Therefore, the mitigative properties of the Primary Containment will not be degraded due to reduced pressure retaining capability.

The ability of the Primary Containment to withstand the effects of the postulated metal-water reaction following any DBA is not degraded by this modification.

The portions of the Containment Hard Vent System contained in the Turbine Building is entirely passive piping. No electrical components associated with the hard vent are located in the Turbine Building. Equipment important to safety or abnormal events which is considered by the SAR and which is located in the Turbine Building, include pressure regulators, Main Turbine control and stop valves, feedwater heaters and feedwater piping. The only way that the hard vent system could cause a malfunction of these systems is by falling on them. The probability of the hard vent piping falling on equipment in the Turbine Building is remote since piping design meets or exceeds original design standards.

The NE corner room contains safety-related Primary Containment vacuum breakers and air accumulators. Mechanical interaction between the Containment Hard Vent System and Primary Containment vacuum breakers is improbable. Therefore electrical modifications performed for the hard vent system do not make the failure of the Primary Containment vacuum breakers any more probable.

Containment Hard Vent System modifications in the Control Room are purely electrical. All electrical equipment installed complies with the electrical separation requirements of IEEE-279.

Since one of the handswitches requires a key to operate, and because two handswitches must be operated, the probability of inadvertent operation of the hard vent path under any Plant condition due to Operator error is improbable.

The effects of the design basis tornado, flood, high energy line break, and internally and externally generated missiles are not a concern with respect to the hard vent design. Neither are the effects of rail, chlorine, river and aircraft hazards. The hard vent, as an extension of Primary Containment, is shielded from each of these events in a manner consistent with the location of other Primary Containment components.

The only postulated environmental event of concern to the Containment Hard Vent System is vulnerability to a design basis earthquake. Accordingly, containment hard vent piping and components up to and including the first pipe support in the Turbine Building, and the penetration between the NE corner room and the Turbine Building, is designed to Seismic Category I criteria.

The containment hard vent modification provides a path for relief of Primary Containment pressure which bypasses the Secondary Containment radioactive material barrier. The rupture disk blocks the hard vent path to prevent the occurrence of bypass leakage until exposure to a pressure higher than its burst rating (50 psig). The burst pressure is above the peak pressure experienced by containment immediately following a loss of coolant accident (42.7 psig) and is below the Primary Containment pressure limit (53 psig).

The hard vent system, by virtue of its physical location, lack of interaction with other systems and passive nature during normal operations, abnormal operational transients and accident scenarios, can not cause an accident, event or malfunction of equipment important to safety not previously evaluated in the safety analysis report.

DCP 1525 Feedwater Check Valves

Description and Basis for Change

The purpose of this modification was to improve the containment isolation capability of the feedwater check valves and to verify this improvement through Local Leak Rate Testing (LLRTs). The check valves serve as the in-board isolation valves on the feedwater lines. Improvements were accomplished by modifying the check valves internally and removing the actuator/indicators.

A high point vent line was also added downstream of the check valves to facilitate conducting LLRTs and to add flexibility to the test procedure.

The valve discs were re-set to improve the alignment of the valve disc to the valve body, to ensure that internal clearances were correct, and to ensure that interferences between these components were eliminated.

Summary of Safety Evaluation

The modifications did not change the basic operation of the valves. The check valves still function as simple check valves. The SAR has already evaluated the system with simple check valves. The actuators were intended only to supplement the valve's ability to seal. Valve operation is controlled exclusively by feedwater flow. Indication of valve disc position was helpful, but not required for safe system operation. By design, the check valves are verified open by the passage of feedwater, and closed by the loss of feedwater flow.

The addition of the vent line to the Feedwater System did not increase the probability of a loss of feedwater flow accident, since failure of a 3/4" line would not significantly decrease flow to the Reactor. The SAR addresses all the concerns associated with line breaks inside the Drywell. The installation of new small-bore vent lines in the Drywell will not initiate any previously unaddressed malfunctions.

Because valve performance is improved, this modification decreases the probability of an incident involving the Feedwater System and as such does not affect the accident analysis in SAR Chapter 15 or the NSOA. The modifications will improve the ability of the system to perform its containment isolation function. Installing blind flanges over the stuffing box area decreased the likelihood of a radiological release through the valve stuffing box. Therefore, the radiological consequences of an accident evaluated in the SAR were not increased.

DCP 1526 HPCI and RCIC Flow Instrumentation Modification

Description and Basis for Change

The HPCI steam line high flow isolation and RCIC steam line high differential pressure isolation setpoints were raised to provide more operating margin during system startup in order to avoid spurious isolations.

The setpoints for the flow switch for the HPCI pump low discharge flow, were changed to reflect the change in the flow orifice done under a previous modification. Changing the setpoints ensures that the HPCI pump minimum flow valve will operate at 300 and 600 gpm.

A flow transmitter was replaced and instrument sensing lines were rerouted to allow for wet calibration.

The level switch for the HPCI Turbine lube oil tank high/low level was modified to directly power an annunciator window in the Control Room.

Summary of Safety Evaluation

The new setpoints for HPCI and RCIC optimize the margin for system reliability while maintaining a conservative margin for the isolation function. The new setpoints and corresponding steam flow values are bounded by the calculations for 300% of rated flow as described in the SAR.

The modifications made the HPCI and RCIC Systems more reliable. The setpoint changes increased the reliability of the HPCI and RCIC Systems by reducing the likelihood of an inadvertent isolation during system startup. The modifications to the flow transmitter and level switch also increased their reliability. The modification to the flow switch ensures that the minimum flow valve operates as originally designed.

The modifications have not resulted in any functional change to the affected instrumentation. The changes have not created the possibility of a new or different kind of accident or a different type of malfunction of equipment important to safety.

DCP 1528 Safety-Related MSIV Nitrogen Piping Replacement

Description and Basis for Change

This modification replaced the safety-related portion of the copper nitrogen supply piping for the outboard MSIVs with stainless steel piping. The soldered joints in the copper piping were susceptible to leakage which had resulted in loss of nitrogen pressure causing inadvertent MSIV closure and subsequent Reactor scram.

The check valves in the supply piping for the MSIV accumulators were also replaced. The 1.25" copper piping was replaced with 1.5" schedule 40 stainless steel piping. The 3/4" copper tubing was replaced with 3/4" stainless steel tubing.

SUMMARY OF SAFETY EVALUATION

The safety function of the MSIVs is to provide isolation of the Primary Containment and Reactor pressure vessel to limit the release of radioactive materials during design basis events. Nitrogen is supplied to the MSIVs to provide motive force for operation. Piping replacement does not alter any of the inputs or assumptions previously evaluated and described in the SAR or the NSOA. This modification does not prevent or degrade the capability of the MSIVs to perform their safety function.

The effects of the replacement check valves on system LLRT requirements; the effects of this modification on Technical Specifications 3.7.D, Primary Containment Power Operated Isolation Valves; and the susceptibility of the replacement stainless steel piping to stress corrosion cracking were analyzed.

Piping replacement did not affect MSIV isolation logic or valve closure times. There is no credible failure of the nitrogen supply piping which can prevent the outboard MSIVs from closing upon receipt of an isolation signal. Stainless steel piping meets the same codes and standards (including seismic design) and is located in the same areas as the original copper piping. Nitrogen flow requirements are maintained.

DCP 1529 Breaker 1B4223 and Panel 1L08 Load Reduction

Description and Basis for Change

The purpose of this modification was to correct a breaker tripping problem by reducing the load connected to a Lighting Panel. Power to Supply Transformer 1XL80 was switched to another panel via a new 100 amp circuit breaker. The mounting configuration and seismic qualification of the new circuit breaker are the same as the previously existing spare which it replaced. The reduction in the inrush current to the Lighting Panel and an optimum instantaneous setting of the associated breaker allows the system to be re-energized without tripping the breaker.

Summary of Safety Evaluation

No new loads were added or removed by this modification. The redistribution occurs inclusively within a safety-related system. However, the modification has no effect on the accidents listed in the NSOA and analyzed in the SAR. This modification only changed the distribution of power and has not introduced any new systems interaction. This modification has not increased the probability of occurrence of an accident previously evaluated in the SAR.

Malfunctions of equipment important to safety as listed in the NSOA and as analyzed in the SAR that relate to this modification are inadvertent breaker trips and/or failure to trip. The modification has not increased the probability of these abnormal events. All original design specifications for material and construction practices have been met (seismic specifications, separation criteria, environmental qualification).

DCP 1531 RPV Level Instrumentation Piping Reroute

Description and Basis for Change

During review of the effect of high Drywell temperature on the Reactor Vessel level instrumentation, it was determined that the HPCI and RCIC System automatic high level trips may not occur with the existing setpoints under specific DBA conditions. In order to assure that these trip functions will occur prior to water covering the reference leg taps, the existing variable legs are being rerouted to reduce the vertical drop in the

Drywell and in turn reduce the calculated error in the system. Modifications to the 'B' side have been completed. 'A' side modifications will be completed during the next refueling outage.

The change implemented the requirements of Generic Letter 84-23 for Reactor Vessel level indication.

Summary of Safety Evaluation

The method of operation and safety function of equipment important to safety were not changed. The variable legs associated with the identified instruments were rerouted to reduce the vertical drop inside the Drywell. The new pipe routing reduces the error induced in the system from Drywell temperature fluctuations during various events. The error reduction enhances the system capabilities to reflect levels closer to the actual level. The modification does not change the input to the analysis performed for events previously evaluated in the SAR.

The modifications meet all requirements of instrument accuracy, mechanical reliability, seismic interactions, fire protection, and environmental qualification. The unattached pipe length is not adequate to create the impact energy necessary to exceed the energy required to penetrate the containment. The design bases for the Reactor Vessel level instrumentation have also been maintained. The piping was designed to original Plant standards and the existing excess flow check valve, isolation valve, and flow orifice were reused, therefore, the probability of a variable leg rupture which could not be isolated has not been increased. The rerouted variable leg piping inside the Drywell reduces the amount of piping susceptible to system interaction inside the Drywell. Analysis confirms that new failure mechanisms have not been introduced. The associated trips as listed in Table 3.2-B of the Technical Specifications have been maintained as specified.

3-D Monicore

Replacement of P-1 Core Monitoring Software

Description and Basis for Change

The P-1 Core Monitoring Software, which has been in use since initial startup, has been replaced with 3-D Monicore Software to provide better modeling of the core. 3-D Monicore uses diffusion theory while P-1 employed fit coefficients to determine the three-dimensional solution of the power distribution. The diffusion theory of 3-D Monicore has been shown to be superior to P-1 and is used in a majority of GE BWR's for determining thermal limits.

Summary of Safety Evaluation

As discussed in Section 7.7.4 of the SAR, the PPC has neither a safety objective nor a safety design basis. The process computer does not play a role in safely shutting down the Reactor after the start of an accident or transient. However, accident and transient analysis assume that initial conditions are within operating limits as defined by the Core Operating Limits Report (COLR). Therefore, the accuracy with which thermal parameters are determined is important in maintaining the integrity of the fuel. The parameters that are monitored for the purpose of protecting fuel integrity are Minimum Critical Power Ratio (MCPR), Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), and Maximum Linear Heat Generation Rate (MLHGR).

The accuracy of 3-D Monicore is therefore depended upon to insure that fuel operating limits are maintained. GE addressed 3-D Monicore accuracy in NEDE-20340-3, Revision 1, 'Process Computer Performance Evaluation Accuracy'. In this report, GE compared gamma scan data taken at Plant Hatch 1 to power distributions calculated by 3-D Monicore. GE concluded, "...the 3-D Monicore model is more accurate than the allowances made for the current process computer model, thereby justifying its use with current margins."

Although the process computer is not discussed in the Technical Specifications, the thermal limit outputs are part of the COLR, which is referenced in the Technical Specifications. The function of the Core Monitoring Program has not been changed. The change only impacts the way in which the core operating parameters are calculated and does not increase the probability of occurrence of an accident or failure of equipment which could initiate an accident previously evaluated in the SAR. This change does not decrease the accuracy with which operating limits are determined.

HPCI/RCIC Setpoints

Changes Due to Potential Adverse Consequences of High Drywell Temperatures

Description and Basis for Change

As a result of engineering validation efforts involving setpoint calculations, it was recognized that the triple-low function of certain vessel level instrumentation may not function under specific DBA conditions (high Drywell temperatures), due to differential heating effects on the instruments' variable and reference legs. During follow-up investigation on level instruments affected by high Drywell temperature, it was determined that high

Drywell temperatures could cause indicated level to be less than actual level due to the configuration of the reference and variable legs for the narrow range vessel instrumentation (unequal lengths of each in the Drywell).

The variable legs were previously rerouted to reduce the vertical drop in the Drywell and in turn reduce the calculated error in the system. DCP-1531, summarized in this report, describes this modification.

Summary of Safety Analysis

SAR Section 15.1 describes events that result in a Reactor water temperature decrease and includes a discussion of feedwater controller-maximum demand and an inadvertent HPCI pump start. Revising the setpoints for the HPCI and RCIC high level trip instruments (and leaving the Main Turbine trip/Main Feed Pump trips as-is) cannot affect the probability of these events. The setpoints only affect at what level the Turbines trip.

The NSOA identifies the transients and accidents for which HPCI and RCIC are designed to mitigate. In those events in which Drywell temperatures are not elevated, HPCI and RCIC will trip prior to 211" above Top of Active Fuel (TAF) with the revised setpoints. However, this is not a concern because both of these systems will continue to reset at 119.5" TAF and maintain vessel level in the desired level band. The narrowing of the automatic level band from 119.5"-211" to 119.5"-207" is not significant.

At maximum Drywell temperatures, the revised setpoints may result in the HPCI and RCIC systems tripping at actual levels greater than 211" TAF but no greater than 227" TAF under worst case conditions. However, the function of this trip is maintained in that the HPCI and RCIC Turbines will still trip in time to prevent water from entering into the main steam lines which could potentially cause increased piping or SRV stresses.

The consequences of feedwater controller failure-maximum demand are also not increased. In the unlikely event that this transient occurs coincident with extreme Drywell temperatures (>320°F), the vessel level would reach the high level setpoint long before the variable leg would equalize with the high Drywell temperatures and therefore, the high level trip (for both the main feed pumps and Turbine) would occur as designed.

The consequences of an inadvertent HPCI pump start are also not affected. This transient is mitigated by the feedwater controller which will compensate by decreasing feedwater flow. This transient is extremely unlikely to occur coincident with high Drywell temperatures. In the event it did, all mitigating actions would have occurred prior to variable leg heat up.

Equipment important to safety previously evaluated in the SAR will continue to initiate on all the required signals and will continue to cycle on/off at or around the appropriate setpoints. While revising the setpoints results in one of the instruments (for each system) being tripped during normal operating conditions, both systems will continue to trip on high water level and prevent water from entering the main steam lines. The affected instruments are reliable and not expected to fail.

One of the instruments adversely affected by high Drywell temperatures is an input to feedwater level control. At extreme Drywell temperatures, the Feedwater level control instrument will sense (erroneously) that vessel level is decreasing and will increase feedwater flow to compensate. Analysis of this scenario has determined that the consequences of this transient (conservatively assuming no feedwater high level trip) are bounded by the feedwater controller failure-maximum demand scenario.

The basis for the HPCI and RCIC 211" trip setpoint is to ensure that the Turbines are tripped in time to prevent vessel level from spilling over into the main steam lines and potentially causing Turbine, main steam line or SRV damage. With the revised setpoints, this trip will still occur less than or equal to 211" indicated level and prevent water from entering the main steam lines. The margin of safety for the HPCI and RCIC systems is also maintained in that all initiation signals are operable and the system will cycle on and off as required.

NCR 91-092 <u>Missing Backdraft Dampers D73-0001 and D73-0004</u>

<u>Description and Basis for Change</u>

A system walkdown revealed that two backdraft dampers which should have been installed in the Standby Filter Unit (SFU) exhaust ductwork were not installed as shown on Plant drawings. An engineering evaluation was performed to address the consequences, and resulted in the decision to "use as is". Design documents were changed to reflect the "as built" conditions.

The dampers were intended to prevent backflow from the Battery Room exhaust trunk into the SFU System during times when the SFU exhaust fans are idle.

Summary of Safety Evaluation

The need to install backdraft dampers is unnecessary because of the normally closed, air operated (fail closed) butterfly valves presently installed in the lines. The air operated valves provide a positive seal, thereby preventing backflow into the SFU System.

SAR Chapter 15 (Accident Analysis) was reviewed in order to identify and evaluate the impact on previously evaluated accidents. The backdraft dampers are not associated with, or related to, pressure boundary, control rod operation/logic, refueling operations, or steam line components. Consequently, the missing dampers do not increase the probability of occurrence of any DBA, or impact any of the Abnormal Operational Transients.

The capability of the SFU system to perform its design function is maintained. Technical Specifications Bases (Section 3.10.A) discuss flow rates and filter efficiencies. Neither are impacted by the absence of the backdraft dampers. The consequences of an accident previously evaluated in the SAR and in the NSOA are not increased.

The absence of backdraft dampers in the SFU exhaust ductwork eliminates one of the two barriers which prevent backflow into the filter trains. However, the consequence of this is insignificant considering the reliability and positive seal provided by the air operated isolation valves.

The backdraft dampers were not intended to protect equipment or components in the SFU system. Therefore, the absence of the dampers does not create the possibility of a malfunction of those components.

During 1992, various procedures as described in the Safety Analysis Report (SAR) were revised and updated. All changes were reviewed against 10 CFR 50.59 by the Operations Committee. Summaries of these procedure changes and their safety evaluations are provided below. No procedure changes were made that involved unreviewed safety questions.

Thirteen (13) Special Test Procedures (SpTPs) were performed in 1992. Each was reviewed by the Operations Committee. No unreviewed safety questions were found to exist. Summaries of these special tests and their safety evaluations are found below.

TEST/PROCEDURE	TITLE/DESCRIPTION
SpTPs:	
174R0	Cardox Demonstration Test for the Cable Spreading Room

Description and Basis for Test

This SpTP performed a functional discharge test of the Cable Spreading Room CO_2 System. The purpose of this test was to assure Control Room habitability and to verify that the Cable Spreading Room CO_2 System can provide a design concentration of 50% carbon dioxide.

Summary of Safety Evaluation

The accidents previously evaluated are described in the SAR and NSOA. The probability of occurrence of these accidents is based on initial conditions and assumptions which do not depend on the end use of or interactions with fire protection systems. Therefore, the functional test of the Cable Spreading Room CO_2 System conducted by this SpTP did not alter any of the inputs or assumptions for the probabilities of accidents previously evaluated.

A review of the SAR and the NSOA matrices confirmed that the Cable Spreading Room CO_2 System is not required to perform a safety function or to support a system which is needed to perform a safety function. It was determined that the functional testing of the Cable Spreading Room CO_2 System would not affect safe shutdown capability because cables in the Cable Spreading Room would not be affected with a CO_2 concentration of 65% or less (substantiated by testing conducted for EPRI, published in EPRI Report NP 7253). Because maximum CO_2 concentration was 53% or less, Control Room instrumentation would not be affected, and the probability of occurrence or the consequence of any accident or malfunction of equipment important to safety previously evaluated in the SAR would not be decreased by the testing.

The Cable Spreading Room was analyzed as a single fire zone. As a result of this analysis, the entire Cable Spreading Room could be lost due to a Design Basis Fire while the Control Room would still have the capability to bring the Plant to a safe shutdown condition. Additionally, the Plant was in cold shutdown during the performance of this test. Thus, it was determined that the test would not degrade or prevent any action assumed to occur in the SAR to mitigate the consequences of a malfunction of equipment important to safety.

During testing, the Cable Spreading Room was inert with carbon dioxide, so no fire growth was possible. The Control Room Operators had SCBA available, each with a one hour supply of air (which includes two spare bottles for each SCBA) and back up capability for six hours of supply air. The Health Physics Department continuously monitored the Control Room for carbon dioxide and oxygen deficiency. After purging of the Cable Spreading Room, a continuous fire watch was posted until the Cardox CO_2 tank had been refilled to a minimum of 90% as required by the Technical Specifications. No margin of safety as discussed in any Technical Specifications Bases was reduced by this test.

174R1 Cardox Demonstration Test for the Cable Spreading Room

Description and Basis for Test

Following the test described in the preceding section, SpTP 174R0 was modified.

Revision 1 of SpTP-174 tested the Cable Spreading Room CO_2 system with an additional vent path installed and sized to maintain the room pressure to 1-2" W.G.. The Cable Spreading Room exhaust damper was set to open approximately 4 minutes and 45 seconds after CO_2 initiation.

Summary of Safety Evaluation

The Safety Evaluation for SpTP 174R0, summarized above, is also applicable for this test.

175R0
175R1
176R0
176R1
177R0
178R0
179R0
180R0
181R0
182R0
183R0

Motor-Operated Valve Testing

Tests performed under the SpTPs listed to the left are discussed below.

Description and Basis for Testing

To fulfill the dynamic testing requirements of NRC Generic Letter 89-10, a series of performance tests were completed. The purpose of the tests was to demonstrate operability of RHR motor-operated valves and to collect performance data under the following conditions:

a) opening against the differential pressure associated with the RHR System Pumps operating near shutoff head

b) closing against the flow associated with the RHR System Pumps operating at the maximum achievable flow through the motor-operated valve being tested.

Each test was performed while operating the RHR System in its design configuration, i.e., within normal operating parameters (flow, pressure, temperature).

Summary of Safety Evaluation

Since the performance tests did not affect the ability of the RHR System to perform its design basis functions, the specified system functional requirements and restrictions that must be observed within each operating state, as defined in the NSOA, were not affected.

In some cases the tested valve was considered inoperable (but functional) while the calibration device was installed. Consequently, one Containment Cooling Subsystem was declared inoperable (but functional) during the short period of time that the calibration device was installed. This conservative rationale ensured that Technical Specifications LCO requirements were satisfied.

Although the RHR System does contain equipment important to safety, the RHR System remained fully functional during the performance test. The performance test did not affect any other system. The activity did not have any negative impact on equipment important to safety.

AOP-301.1 STATION BLACKOUT

Description and Basis for Change

A new Station Blackout (SBO) procedure was developed to provide better guidance to the Control Room Operators.

Under SBO conditions, either the HPCI or RCIC systems are required as water makeup sources to the RPV for significant periods of time independent of station (Vital and Nonvital) AC buses. During HPCI or RCIC operation, HPCI and RCIC room heatup will occur from both system operation and the lack of room cooling. To preclude loss of these systems (the only systems available for RPV water inventory makeup), the procedure includes steps to defeat the HPCI and RCIC rooms Steam Leak Detection isolations and to block open the room doors to establish natural ventilation pathways.

During a SBO, Control Room heatup will occur from the lack of room cooling. To maintain Control Room habitability and system control operability, the procedure includes steps to block open doors and establish forced ventilation.

Restoration of AC power is a top priority, so the procedure allows the Operator to defeat the initiation signal in order to take actions to restore the Standby Diesel Generators (SBDGs) to service. To restore offsite power, the procedure authorizes the bypassing of the essential buses degraded voltage interlocks.

The new procedure bypasses degraded voltage interlocks to ensure that once power has been restored, it does not trip during the starting of loads. The power from offsite sources may not be of sufficient quality to keep the degraded voltage relays picked up. The Operator is directed to place the SBDG in lockout. He is also required to recharge the starting air flasks if needed.

Summary of Safety Evaluation

The new procedure provides assurance of the continuation of adequate core cooling during a total loss of all AC power by ensuring that the HPCI and RCIC Systems are available. Although the procedure defeats the Steam Leak Detection System for the HPCI and RCIC rooms to ensure these systems are available for injection, it also directs the Operators, in accordance with the EOPs, to start depressurizing the RPV. This limits the amount of thermal and pressure stress on the RPV and piping systems to less than design. Therefore, the probability of occurrence of a steam leak in the

HPCI or RCIC rooms is reduced. This depressurization also maximizes core coolant flow and assures the availability of HPCI and/or RCIC to ensure adequate core cooling. In this manner, fuel clad temperatures will remain within acceptable (10CFR 50.46) limits, resulting in no increase in radiological consequences and thereby avoiding other undesirable consequences such as hydrogen generation from metal-water reactions.

Defeating the Steam Leak Detection isolations and blocking open doors to establish ventilation flow paths provides assurance that the HPCI and/or RCIC systems will be available to provide core cooling. Blocking open doors and providing forced ventilation exhaust assures that habitability of the Control Room and availability of equipment necessary to cope with a SBO event is maintained.

The SAR provides evaluations of the consequences of accidents from the perspective of maintaining core cooling, minimizing radiological consequences, and avoiding undesirable system interactions. The new procedure for Station Blackout has no adverse effect on DBAs discussed in the SAR. Under design basis conditions, sustained loss of AC power is not a condition originally designed for or evaluated.

ATWS EOP, RPV Control; EOP 1, RPV Control; EOP 2, Primary Containment Control; ED, Emergency RPV Depressurization (Contingency 2)

Description and Basis for Change

Emergency Procedure Guidelines (EPGs), Appendix B, specified how to develop a caution regarding the use of RPV water level instrumentation. It also provided an analysis of the basis for each portion of the caution. NRC Generic Letter 84-23 identified concerns with the effects of Drywell temperature on water level instrumentation, while Regulatory Guide 1.97 detailed range and redundancy requirements. In order to meet commitments to these concerns and requirements, the majority of the reference leg piping for the narrow range and fuel zone range instrumentation was relocated to outside the Drywell. The caution was modified to take advantage of the improved design.

In order to increase availability and flexibility, Caution 1 of the procedures listed above was modified regarding the use of fuel zone water level instrumentation. The change permits the use of the fuel zone level instrumentation when the average Drywell temperature exceeds Reactor Pressure Vessel (RPV) saturation temperature, as long as the Operators closely monitor the instrumentation for erratic indications. This allows continued use of an indication that would otherwise be deemed unreliable. Thus, the margin of safety as described in the Technical Specifications is not reduced.

Summary of Safety Evaluation

The procedure change will not alter the manner in which any Plant equipment operates, and does not change any system design bases. Therefore, the accident analyses and the assumptions that were made for each scenario as described in SAR Chapter 15 are unaffected. Per SAR section 7.6.4.2.2 Safety Design Basis, "Reactor Vessel instrumentation is designed to provide the Operator with sufficient indication of Reactor Vessel water level during planned operations to determine that the core is adequately covered by the coolant inventory." The availability and reliability of Reactor water level instruments during planned operations is not affected. The redundancy of installed equipment is not affected. In addition, the procedure change requires Operators to more closely monitor water level instrumentation, making early detection of any event that could influence level more likely.

The procedure change allows the instrumentation to be used until erratic behavior, which indicates reference leg flashing, is observed. This is to account for the time required to heat the reference leg piping to saturation temperature, which can be 17 minutes or more, depending on the size and location of the break, and the associated Drywell heatup rate. No instrumentation is used when its indication may be unreliable. Instrumentation is still required to be operated in accordance with approved procedures, except over a wider range of environmental conditions. No instrumentation is used in or subjected to any conditions for which it is not designed.

Technical Specifications discuss that the core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Fuel zone water level instrumentation provides this trip function to containment spray. All other trip functions are provided by instrumentation unaffected by the proposed change. Additionally, fuel zone water level instruments are calibrated cold, and are not density-compensated. This results in actual level being greater than or equal to indicated level for normal, transient, and accident conditions. Because the setpoint is unchanged, the instrument indicates conservatively, and the increased monitoring by the Operators will result in timely detection should any automatic action fail to occur.

EOP ED Emergency RPV Depressurization (Contingency 2)

Description and Basis for Change

In accordance with 10CFR50.63, Loss of all alternating current power, DAEC was required to demonstrate the ability to withstand and recover from a Station Blackout (loss of all AC power). Analysis has been conducted and reviewed by the NRC in a Safety Evaluation Report (SER). This SER states that "...the licensee will depressurize the Reactor within the first hour of the event to limit the Drywell temperature to less than 300°F." Under Station Blackout (SBO) conditions, either the HPCI or RCIC Systems can be used for significant periods of time independent of station (Vital or Non-Vital) AC buses. Under these circumstances, depressurization of the RPV should be terminated such that the RCIC and/or HPCI System can continue to provide steam driven coolant to the Reactor Vessel.

A continuous recheck statement was added to the Emergency Operating Procedures (EOPs) to incorporate this guidance.

Summary of Safety Evaluation

The SAR provides evaluation of the consequences of accidents from the perspective of maintaining core cooling, minimizing radiological consequences, and avoiding undesirable system interactions. The change has no effect upon the course of Design Basis Accidents discussed in the SAR. The change was approved by the NRC prior to implementation.

Adequate core cooling is assured by maintaining HPCI and RCIC above isolation pressure when only these systems are available for core cooling (regardless of the entry condition into the ED procedure). Although the SBO sequence is the only dominant (credible) sequence that could lead to these circumstances, it would be equally appropriate action if no low pressure makeup were available for other reasons. For circumstances in which no low pressure systems are available, but the high pressure Turbine driven HPCI and/or RCIC Systems are available to provide core cooling, terminating vessel depressurization prior to low RPV pressure isolation of these systems will provide assurance of continuation of adequate Core cooling, and will ensure that fuel clad temperatures will remain within acceptable (10CFR50.46) limits, resulting in minimal radiological consequences and avoiding other undesirable consequences such as hydrogen generation from metal-water reactions. In addition, invoking the depressurization in the manner described in this procedure change also maintains containment parameters within acceptable limits, ensuring containment integrity and environmental qualification of equipment important-to-safety.

The modification of the EOPs ensures that the SBO event is accommodated by the Plant's design without exceeding important limits and safety analysis acceptance criteria. This is consistent with the purpose of EOPs to not be limited only to DBAs or events.

COLR

Core Operating Limits Report (COLR)

Description and Basis for Change

The COLR provides cycle-specific operating limits for the current operating reload cycle. It is revised each reload cycle using NRC approved methodologies and analysis. Prior to issuance of the COLR, a 10 CFR 50.59 review is performed to ensure that the revised operating limits or changes to the fuel do not result in an unreviewed safety question.

During the Cycle 11/12 refuel outage, GE8x8NB-3 (GE10) fuel was loaded into the core. As new bundle types are added to the Reactor core, the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) operating limit for the new fuel was calculated. This ensures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10 CFR 50.46 and that the fuel design analysis limits specified in GESTAR II will not be exceeded. As part of the reload licensing process for Cycle 12, GE has reanalyzed the most limiting abnormal operating transients as specified in Table 15.0-1 in the SAR. This analysis was performed using GE transient analysis methodologies. The results of these analyses are reported in the Supplemental Reload Licensing Submittal for the DAEC, Reload 11, Cycle 12.

Summary of Safety Evaluation

The new Cycle 12 operating limits and fuel do not increase the probability of a previously evaluated accident because no changes were made to the facility or its equipment other than the introduction of the additional (GE10) fuel. This fuel design evolved from the GE8x8NB and incorporates two unique features: interactive channel design and an offset lower tie plate. This new design has been analyzed using NRC approved methodologies. These analyses have shown that this new fuel meets all the required fuel performance criteria. The NRC has approved the use of the GE10 fuel in their SER to Amendment 21 to GESTAR II. 104

bundles of this fuel type were also loaded during the Cycle 10/11 refueling outage.

Transient and accident analyses for the new core have shown that the results of the transients and accidents are within the bounds of previously accepted analyses including the requirements of 10 CFR 50.46 and the fuel cladding integrity safety limits. The changes to the operating limits only reflect the results of the Cycle 12 analysis.

Setpoint Change

Intermediate Range Monitor (IRM) High Flux Alarm and Rod Block

Description and Basis of Change

As part of the DAEC Scram Frequency Reduction Plan, the IRM high flux alarm and rod block setpoint was changed (lowered). Since the alarm now comes earlier, Operators have more time to uprange IRM instrumentation prior to receiving an upscale RPS trip, and the likelihood of an inadvertent Reactor scram is reduced.

Summary of Safety Evaluation

The setpoint change alters the manner in which Plant equipment operates to a more conservative position, because the high flux alarm and rod block will occur earlier than previously. However, the required protective action (high flux trip) remains unaltered. The IRM safety design basis is unaffected, and the power generation design basis is modified to a more conservative value (SAR section 7.6.1.5.1 and 7.6.1.5.2). The setpoint change did not alter the initial conditions assumed for any of the accidents analyzed, nor did it alter Operator or automatic actions of any safety systems that are relied upon for accident mitigation.

All IRM instrumentation will continue to function as before, except that the high flux alarm and rod block setpoint will occur at lower values of neutron flux. Failure of any one instrument to alarm or trip at the proper point will not prevent a required protective action. Adjustment of a convenience feature for Operators does not increase the consequences of a malfunction of equipment important to safety.

SAR section 15 discusses accident analyses and the assumptions that were made for each scenario. The setpoint change will not affect any of the DBA analysis nor allow any credible, non-evaluated accident to occur, because of designed redundancy in the IRM system, and the presence and adequacy of the IRM 120% trip setpoint.

Lowering the IRM high flux alarm and rod block setpoint does not violate any design basis, design safety standards, or specification. All actions for protection of personnel and equipment are unchanged.

SECTION C - EXPERIMENTS

This section has been prepared in accordance with the requirements of 10 CFR Part 50.59(b). No experiments were conducted during calendar year 1992.

SECTION D - SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES

This section has been prepared in accordance with the requirements of Technical Specifications 6.11.1.e., "A report documenting safety/relief valve challenges shall be submitted within 60 days of January 1 each year."

DATE	EVENT DESCRIPTION
11/13/92	A failure of a Circulating Water Pump caused the Turbine to trip because of high pressure in the Condenser. The Turbine trip caused four safety-relief valves (PSV4400, 4401, 4406, and 4407) to momentarily lift as designed when Reactor pressure increased to approximately 1138 psig. LER 92-018 discusses this event.

SECTION E - FIRE PLAN CHANGES

The information contained in this section identifies, briefly describes and provides assurance that changes made to the DAEC Fire Plan during the calendar year 1992 did not alter our commitment to the NRC guidelines contained in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

Revision No.	Description of Change
27	This revision was editorial in nature. It only involved changes in titles and responsibilities.