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October 19, 1995
NG-95-2971

Mr. William T. Russell, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

10 CFR 50.59

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
1995 Cyclic Report of Facility Changes, Tests and Experiments and Fire Plan Changes
File: A-118e

Dear Mr. Russell:

In accordance with the requirements of Appendix A to Operating License DPR-49, 10 CFR Section 50.59(b), and NUREG-0737 (Item II.K.3.3), please find enclosed the subject report covering the period from January 1, 1994 through October 1, 1995. A summary of changes to the Duane Arnold Energy Center Fire Plan implemented during the same period is included.

Should you have any questions regarding this matter, please contact this office.

Sincerely,

John F. Franz
Vice President, Nuclear

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Attachment

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**Table of Contents**

## Section A - Plant Design Changes

This section contains brief descriptions of plant design changes completed during the period beginning January 1, 1994 and ending October 1, 1995, 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 involve unreviewed safety questions.

The basis for inclusion of a modification in this report is operational release of the associated modification at the DAEC during the period beginning January 1, 1994 through October 1, 1995. Portions of some of the Plant Modification Packages (PMPs) and Design Change Packages (DCPs) which are listed were partially closed or partially operationally released in previous years.

PMP 0049 Panel 1C-186 Expansion

Description and Basis for Change

This modification replaced obsolete Johnson Controls pneumatic controllers with the manufacturer's suggested direct replacement part. The function and operating specifications were unchanged. Due to a size difference, a new cabinet was added to house the new controllers. This change had no effect on the operation of the Radwaste Building Heating, Ventilation and Air Conditioning (HVAC) system and improved the reliability and the availability of spare parts.

Summary of Safety Evaluation

The activity does not increase the probability of occurrence nor the consequences of an accident previously evaluated in the Safety Analysis Report (SAR). System performance is unchanged; therefore, all previously performed analyses are still valid. The activity does not increase the probability of occurrence nor the consequences of a malfunction of equipment important to safety. Radwaste HVAC is not safety related. Replacement controllers meet all original design requirements. The new controllers are a direct replacement with respect to form and function and do not perform a safety function.

The activity does not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. System performance is unchanged; therefore, all previous analyses are valid. The new equipment meets all original design specifications. The margin of safety at DAEC is not reduced by this modification. TSs are not affected.

PMP 0082 Turbine Building Exhaust

Description and Basis for Change

The purpose of the PMP was to permanently implement a temporary modification to allow the Turbine Building (Roof) Exhaust fans to continue to run with no interlock to Main Exhaust Plenum negative pressure. The intent of this change was to prevent the Turbine Building from being without ventilation following a Group 3 isolation and subsequent Standby Gas Treatment System initiation.

Summary of Safety Evaluation

Allowing the Turbine Exhaust Fans to continue to operate following a Group 3 isolation will provide improved equipment cooling and better monitoring of Turbine Building gaseous releases. The Turbine Building Exhaust Fans are not nuclear safety related and do not perform a function important to safety.

The normal air flow path in the Turbine Building will be reversed. This may result in airborne contamination levels in the upper Turbine Building elevations increasing. This is far outweighed by the improved equipment cooling provided by having ventilation operable.

PMP 0084 Offgas Air Purge

Description and Basis for Change

In 1992, it was noted that hydrogen/oxygen recombination was not occurring within the offgas catalytic recombiner; rather recombination was taking place upstream. This caused premature recombination upstream of the offgas catalytic recombiner. Attempts to halt the premature recombination were unsuccessful. The plant was shut down and the Offgas system was taken out of service. When the Offgas system was placed back in operation and plant power was increased, the premature recombination recurred as the Hydrogen Water Chemistry (HWC) system was placed in service. It is believed that catalytic dust may have migrated within the piping upstream of the offgas jet compressor. When HWC was placed in service, the stoichiometric mixture of the process stream favored recombination in the presence of the catalytic dust. The problem was solved via a temporary modification which injected air downstream of the jet compressor in lieu of oxygen normally injected upstream.

The Instrument Air supply to the Offgas Air Purge system was modified to support controlling and monitoring continuous air injection during normal Offgas system operation. No physical changes to the HWC system or control logic were made. Oxygen to the Offgas system was procedurally valved out. Normally, much of the oxygen needed for recombination is provided from main condenser air inleakage. Adjustments to air injection will ensure that sufficient residual oxygen (10-15%) is present to recombine all hydrogen present in the Offgas system.

The added piping and valves tie into Instrument Air system piping (not Offgas system piping), and meet the codes and standards applicable to the Instrument Air system. The Instrument Air system at the point of tie-in for this modification is not safety related, and is not discussed in any of the DAEC's top level Design Basis Documents.

<u>Summary of Safety Evaluation</u>

None of the systems affected by this modification have safety related functions that will be impacted by this modification nor are any of the systems identified as the causes of accidents which have been evaluated in the Updated Final Safety Analysis Report (UFSAR) and the Nuclear Safety Operational Analysis (NSOA). By discontinuing the injection of oxygen upstream of the offgas jet compressor, the probability of a plant transient caused by premature recombination is reduced.

Since the air injection system was constructed to the same codes and standards applicable to the present Instrument Air System, a mechanical break or failure is no more likely. In the event of an unexpected loss of air injection, a low offgas residual oxygen condition will occur which will be alarmed in the Control Room at 10%, and will trip hydrogen injection at 5%. This is the same result as would occur with an unexpected loss of oxygen injection. A malfunction resulting in excessive air injection would result in increased offgas flow and is alarmed in the Control Room.

Replacing oxygen injection with air injection will increase activity release rates out of the Offgas system, as illustrated in Section 11.3.1.2 of the UFSAR. This is due to the increased offgas flow rates caused by the nitrogen which makes up 80% of the air injected. An increased flow rate reduces the holdup time of the gases in the Offgas system, and consequently raises the activity level of the gases released. The total holdup time is still far in excess of that required. The Offgas post-treatment radiation monitors are set to automatically isolate the Offgas system if offgas activity reaches the predetermined level which would cause site boundary dose limits to be exceeded. This modification does not affect the automatic isolation feature. Dose to plant personnel is not significantly affected by this modification since dose rates are already extremely small, and access to the Offgas system retention equipment (holdup pipe and charcoal adsorbers) is controlled. The limits for plant personnel exposure are not challenged due to this modification.

PMP 0087 Zinc Injection System

Description and Basis for Change

This modification installed a GE Zinc Injection Passive System (GEZIP) to inject small amounts (≤10 ppb in reactor water) of depleted zinc oxide (DZO) to incorporate a thinner, more protective oxide layer on stainless steel piping and minimize primary system radiation dose build up rates. The zinc addition skid is presently in use at twelve (12) operating BWRs, of which 8 are using DZO.

The passive zinc addition skid is designed to continuously inject small amounts of DZO into the feedwater during normal plant operation. The system consists of a simple recirculation loop off the feedwater system. The zinc solution is obtained by tapping off 20-120 gpm from both feedwater pump discharge headers. This feedwater then goes through a dissolution vessel containing pelletized depleted zinc oxide. The feedwater dissolves the pellets as it passes through the zinc vessel carrying the dissolved DZO into the feedwater pumps' suction header. The supply and return taps are 2 inches in diameter, each with double manual isolation valves.

Summary of Safety Evaluation

The zinc injection system cannot initiate any analyzed accidents. The zinc injection system has no adverse effect on the ability of the feedwater system to perform its safety design basis. Components added by the modification do not compromise the reactor coolant pressure boundary. The system has double manual isolation capability and design quality is maintained to that point.

The addition of zinc injection does not increase the probability of a feedwater system failure. The postulated zinc injection system line failures are all bounded under partial or full loss of feedwater system which is already evaluated in the Final Safety Analysis Report (FSAR). IE calculation M94-009 has evaluated the dead weight and thermal effects of the small bore zinc piping and concluded that the resulting stresses are negligible and acceptable. The feedwater and zinc systems do not mitigate any accidents and do not affect equipment important to safety.

Due to a relatively low recirculation flow rate of 20-120 gpm and double isolation capability, no new failure modes are created. High zinc injection flow rates are easily detected and will be limited by the manual flow control valve so as not to affect reactor water chemistry or feedwater chemistry. The presence of DZO in BWR piping and alloys is not detrimental to the BWR.

The zinc injection system does not adversely affect the feedwater system. The zinc system is of simple design and is a passive recirculation loop around the reactor feed pumps. Addition of the zinc injection system is non-safety related. The double isolation valves located off the feedwater suction and discharge header are of equal design quality as the feedwater system. The feedwater system and the zinc system are not defined in any basis in the TSs. Therefore, the modification does not reduce the margin of safety.

DCP 1516 <u>Security Computer Upgrade</u>

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

This modification replaced existing security hardware which was approaching end of life, and software which was no longer supported by the vendor.

The security facility modifications addressed in this Design Change Package do not affect any plant operating systems. The equipment associated with this DCP has no seismic or environmental qualifications requirements.

<u>Summary of Safety Evaluation</u>

The probability of occurrence of an accident previously evaluated in the UFSAR is not increased by this DCP. These changes are limited in scope to the plant security system which by itself can not cause a design basis accident. Since the security system is not relied upon to mitigate any radiological or nonradiological accidents, the consequences of an accident are not increased.

The security system is not discussed in the TSs. This modification enhanced the security system as defined in the Security Plan. The security computer and access control system is not part of an essential system and will not change the environment of safety related equipment.

DCP 1527 <u>Early Warning Fire Detection System</u>

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

Early warning fire detection was added to both nonessential switchgear rooms and the hot chemistry lab area including the count room, the sample room, and the post accident sample lab. The modifications will enhance property protection for these areas. Temporary Modification 92-429, for monitoring the Thermolag cable tray/conduit fire retardant wrap, in response to NRC Bulletin 92-01, was made permanent by this DCP.

Due to increased combustible loading in the Dry Active Waste (DAW) and Resin Storage Vaults in the Low Level Radwaste Processing and Storage Facility (LLRPSF), the automatic actuation of the deluge systems #24 and 25 has been made manual to minimize the possibility of inadvertent operation.

Summary of Safety Evaluation

No Unreviewed Safety Question will result and no TS revision will be involved because the accidents previously evaluated for the DAEC are described in the UFSAR 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. Because all modifications performed by this DCP are to a fire protection system, it will not alter any of the inputs or assumptions for the probabilities of accidents previously evaluated.

Modifications performed by DCP 1527 do not constitute an unreviewed safety question based on the following:

Design and installation meets design basis and safety parameters.

The function and method of operation of DCP 1527 comply with the original design intent.

No other equipment or systems are affected by these modifications.

DCP 1530 Drywell Sump Isolation Valve Replacement

Description and Basis for Change

DCP 1530 replaced four drywell sump containment isolation valves (two for the floor drain sump and two for the equipment drain sump) in order to eliminate two problems with the existing valves: 1) local leakage rate test (LLRT) failures, and 2) operability concerns because they tended to stick closed and had a slow stroke time. The new valves have double discs and are equipped with larger actuators to provide better sealing capability and faster stroke times. This change will improve the performance and LLRT response of these valves.

Replacement of the piping and valves was in accordance with the appropriate code. The double disc valves are larger and heavier than the original valves. Some lines were rerouted to accommodate the larger size of the replacement valves. The applicable stress and support calculations for these lines were revised to reflect the new configuration and the heavier weight of the replacement valves and ensure that all lines are adequately supported.

Summary of Safety Evaluation

The replacement valves meet the required design bases and closure criteria. No change is made to the isolation control logic. The replacement valves are air operated and fail closed like the original valves. The replacement valves and their limit switches were purchased as seismically qualified and are seismically supported, as is any rerouted piping.

Based on review of DAEC UFSAR Section 5.2.5, flow nozzles are required to continually monitor the flow from the drain sump to the collection tank. This flow measurement is integrated over time to determine volume of leakage inside the drywell prior to containment isolation. The limits for containment leak detection are given in Section 3.6.C of the DAEC TSs. Because the piping reroute put an additional bend upstream of each flow nozzle, an evaluation was made to determine that the reroute does not degrade the flow measurement performance of the nozzles.

The drywell sump and drywell equipment drain sump isolation valves are not identified as initiators for any of the accidents described in the DAEC UFSAR or NSOA. DCP 1530 replaced these valves with larger, faster acting valves. The ability of the valves to close to isolate containment in response to an accident improved.

DCP 1533 RCIC Turbine Steam Inlet Support

Description and Basis for Change

This activity was limited to mechanical modifications to the Reactor Core Isolation Cooling (RCIC) Steam Supply Inlet Line (SSIL) support system, fire penetrations for High Pressure Coolant Injection (HPCI) SSIL, and Reactor Water Cleanup (RWCU) Water Return Line (WRL). No SSIL or WRL pressure boundary was opened, tied into, or modified. RCIC water side instrumentation, control, turbine trip, and isolation logic were not affected. The RCIC system continues to function as per design with no adverse impact to the considerations of the system for planned operation, abnormal operational transients, accidents or other special events.

The RCIC SSIL, HPCI SSIL and the RWCU WRL are classified as Seismic Category I. These items have been reanalyzed and each continues to meet all license based and code criteria for normal system operating conditions, abnormal transients and accidents with deadweight/thermal/seismic considerations.

All RCIC, HPCI, and Primary Containment Isolation System (PCIS) safety design bases continue to be met. Piping and equipment support structures continue to withstand earthquake effects without a failure that could lead to a release of radioactivity in excess of published regulatory values. The modification eliminates from further consideration the effects of a High Energy Line Break (HELB) scenario and ensures continued ASME code compliance for piping stress. There were no changes to the pressure boundary, RCIC , HPCI, or RWCU system water side, instrumentation, control, turbine trip, or isolation logic.

Summary of Safety Evaluation

There is no contribution to overall accident probability from this change. HPCI, RCIC, and PCIS continue to perform their functions when challenged during the NSOA events within their individual design bases. Since the HELB assumption in the torus room and reactor building first floor is eliminated, there is enhanced environmental qualification (EQ) safety margin and reduced probability of malfunction of equipment important to safety on the reactor building first floor.

There is no increase in the radiological consequences of any previously analyzed SAR accident. The RCIC SSIL support modification has no influence on accident consequences, nor does the alteration cause any detriment to the operation of the RCIC, HPCI, or PCIS, systems because the seismic attributes remain unchanged and piping stress remains within Code allowables. RCIC and HPCI, in combination with PCIS continue to perform core cooling safety functions.

The SSIL continues to deliver the steam potential as required to RCIC or HPCI without decreased reliability because the SSIL support system retains the ability to withstand seismic events. No additional loading and stresses are imposed on terminal end interface piping or equipment which have not been previously analyzed. The HPCI, RCIC, and RWCU piping stress analyses conform to the requirements or ASME Section III, Division I, Class 1 or 2 (in conjunction with ASME Code Case N-411).

Accidents of a different type than those of UFSAR Chapter 15 are not influenced by this modification. The Chapter 15 accident analysis envelopes a single failure in the HPCI (RCIC) System and a HPCI (RCIC) Room HELB has been previously considered. The piping remains seismic in nature and HPCI and RCIC, in combination with PCIS, retain the ability to cool down the reactor from outside the Control Room during a loss of Control Room habitability.

There is no adverse effect upon any nearby equipment or piping important to safety because the piping continues to be supported in a manner which minimizes displacements during thermal/seismic operating conditions. The RCIC and HPCI systems themselves contain equipment important to safety; however, there will be no adverse consequences to internal system equipment because SSIL interface reactions have been analyzed and found to be acceptable.

There is no reduction in the margin of safety for any TS basis due to the limited scope and passive nature of this modification.

DCP 1535 Fuel Zone Level Compensation

Description and Basis for Change

This modification installed a compensation system to modify the Fuel Zone Water Level System (Fuel Zone Level) indication to compensate for the inaccuracy induced from the differences in the Reactor Vessel density from the calibrated cold condition to Normal Operating Pressure (NOP) and all points in between. Previously, this compensation was performed by manual calculations. Two Class 1E microprocessors are utilized to calculate the compensated level signal. The level signal is modified in an algorithm to compensate for changes in the variable leg density as a function of Reactor Pressure Vessel (RPV) pressure.

The previous RPV Fuel Zone Level consisted of four channels of instrumentation with a 2/3 core covered interlock. The 2/3 core covered interlock prevents spraying the containment unless adequate core coverage is provided. This modification will not affect the 2/3 core covered interlock. This interlock will continue to utilize the uncompensated Fuel Zone Level signals.

Two indicating lights have been installed to inform the operator when pressure compensation is either not working or is not being applied due to loss of pressure signal or RPV pressure approximately zero. During off-normal operation, Reactor Vessel Water Level Control is performed by Operators based upon indication. Fuel Zone Level is an integral part of the overlapping system of RPV Water Level indication and thus is an input for manual control.

## Summary of Safety Evaluation

Fuel Zone Water Level instruments provide Control Room indication and inputs to the 2/3 core covered interlock and are not initiators to any accident evaluated in the SAR. The modification to this system was designed, procured, and installed in accordance with class 1E requirements.

New equipment installed by this DCP meets the criteria for Equipment Qualifications, Environmental Qualifications, Seismic Qualifications, Class 1E, and electrical separation requirements. The modules used in this application have been tested for Radio Frequency Interference (RFI) and Electromagnetic Interference (EMI) and meet the requirements of IEEE-7432. Thus, the new components are extremely reliable and provide the necessary redundancy.

This modification minimized the time required to determine Fuel Zone Water Level and increased accuracy and reliability by reducing the dependency on the manual calculation. Several Emergency Operating Procedure (EOP) decisions are based on Fuel Zone Level.

The probability of failure of the installed components to calculate Fuel Zone Water Level is much less than the probability of human error in the previously required calculation. The compensation modules and associated hardware/software do not affect any previously analyzed automatically-initiated safety actions, and provide post-accident level monitoring capabilities with enhanced reliability and accuracy.

Extensive testing was performed to provide assurance that no latent errors or failures appear in service. The software configuration associated with this modification is simple, and allowed a very comprehensive verification and validation check.

Fuel Zone Water Level indication does not directly control any safety system, although it does provide input to the 2/3 core covered interlock as part of the containment spray permissive. No failure mode can be postulated to create an accident of a different type than evaluated in the SAR.

The configuration software is treated as an IES Utilities Inc. calculation and has been reviewed and verified in accordance with Engineering procedures. The configuration software has been verified, controlled, and will be revised using applicable IES Utilities Inc. Software Control procedures. This extensive design, review, and control of the configuration software ensures the highest standards of quality. The possibility of unauthorized changes to the configuration is precluded.

DCP 1538 Spent Fuel Storage Pool Rerack

Description and Basis for Change

The spent fuel pool at the DAEC had a capacity of 1898 fuel assemblies. This capacity was projected to be filled by the year 2001, with a loss of the capability of completely off-loading the reactor in 1998.

This change replaced the existing spent fuel racks with racks of higher storage capacity. The end result was increased storage capacity of the spent fuel pool from 1898 fuel assemblies to 2411 fuel assemblies with the option of increasing capacity to 3152 fuel assemblies in future modifications.

Summary of Safety Evaluation

Only proven technology was utilized in the construction process and in the analytical techniques necessary to justify the planned fuel storage expansion. The basic re-racking technology was developed and demonstrated in over 80 applications for fuel pool capacity increases previously approved by the NRC.

The new fuel racks have been adequately designed for seismic events, for heat input into the fuel pool cooling and cleanup system, and for a fuel assembly drop within the spent fuel pool. The results of the seismic analysis state that the fuel pool storage racks will maintain their spacing and arrangement to ensure that the fuel remains subcritical with a $K_{eff}$ not greater than 0.95 and that the spent fuel remains adequately cooled.

The effect of a fuel assembly drop upon the new fuel storage racks was analyzed. An assembly dropped from a height of 18 inches above the rack will not affect the criticality of the fuel storage rack. Although some deformation of the rack will occur, the boron containing material will not be damaged and there will be no change in the spacing between fuel storage cells.

Both the existing and the new racks were analyzed to prevent criticality with a $K_{eff}$ not greater than 0.95 and to maintain fuel spacing in the event of a fuel assembly drop accident. The consequences of a loss of spent fuel pool cooling were not increased by the installation of the higher density storage racks.

The safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all normal or abnormal loadings. Abnormal loadings which have been considered are the effect of an earthquake and the drop of a spent fuel assembly. The mechanical, material, and structural design of the new spent fuel pool racks is in accordance with the applicable portions of NRC Position Paper "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978, as modified January 18, 1979; Standard Review Plan 3.8.4 and other applicable NRC guidance and industry codes. The rack materials used are compatible with the spent fuel pool and the spent fuel assemblies. The structural considerations of the new racks address margins of safety against tilting and deflection or movement, such that the racks will not impact each other during the postulated seismic events. In addition, the spent fuel assemblies remain intact and no criticality concerns exist. Thus the margins of safety are not reduced by the fuel pool re-rack.

DCP 1541 MOV Project Modification

Description and Basis for Change

IES Utilities Inc., per NRC Generic Letter 89-10, committed to develop a program to evaluate and test safety related Motor Operated Valve (MOV) operability under design bases conditions. As a result of this evaluation and testing, it was determined that two MOVs (MO-1909 and MO-2238) would perform their intended safety functions but might not meet the torque requirements during degraded voltage conditions. Excessive voltage drop in the motor feeder cables was a significant contributor to the reduced torque.

This modification increased the cable size to ensure that the MOVs will be able to perform their intended design basis and operational function during normal operation and during design basis accidents. These changes allow the motors to produce adequate torque during degraded voltage conditions and do not alter the design or function of the affected MOVs.

Summary of Safety Evaluation

Changing the feeder cables to MO-1909 and MO-2238 to a larger size to minimize voltage drop, thereby increasing motor torque, merely improved the existing design. The new cables meet or exceed nuclear safety related cable requirements. Cables that needed to be spliced in EQ areas were spliced in accordance with DAEC EQ splicing procedures.

The consequences of a malfunction of MO-1909 and/or MO-2238 do not change. No system, equipment interface, logic, function or operation was changed. Increasing the motor torque does not create any new credible accidents. The valve has been analyzed to withstand the existing torque switch setting.

MO-1909 and MO-2238 receive close and open signals from their respective control logic. This modification does not impact the control logic and does not adversely impact the ability of the valves to open or close. It only changes the effective cross-sectional area of the motor feeder cables.

DCP 1544 LPCI Swing Bus Isolation Logic

Description and Basis for Change

In the event of a main control room fire, the Alternate Shutdown Capability System (ASCS) is enabled. Transfer switches shift control of the reactor from the main control room to the remote shutdown panel. These transfer switches also provide double ended isolation for portions of the control circuit that may be affected by the control room fire, thus protecting the ASCS from the effects of the fire.

It was identified that the double-ended isolation did not exist in the control logic circuitry for the Low Pressure Coolant Injection (LPCI) swing bus feeder breakers. The identified discrepancy allowed the possibility that a fire in the control room could cause another wire to short against the auto trip circuit for either feeder breaker. This would cause the emergency fuse for the swing bus feeder breaker to blow, tripping the feeder breaker. This would effectively prevent the operators from progressing to cold shutdown via Residual Heat Removal (RHR) shutdown cooling because the swing bus feeder breaker would remain tripped and unable to energize the swing bus.

This modification restored the original design intent to the swing bus feeder breaker control circuitry. That is, when the transfer switches are taken to the emergency position, the auto trip circuit of the feeder breakers will be isolated allowing manual control of these breakers as needed.

Summary of Safety Evaluation

This change does not alter the function of the control circuit and restores the original design intent by installing a double ended isolation to the LPCI swing bus control circuitry. This change does not introduce any unwanted or previously unreviewed system interactions and does not affect the seismic or environmental qualifications. The RHR system is unaffected by this change.

The activity meets the original design intent for isolation of the ASCS circuits by providing double ended isolation as required by 10 CFR 50

Appendix R. This change will improve system performance. Therefore, the probability of occurrence of an accident previously evaluated in the SAR or NSOA is not increased by this change.

The modification will ensure that the emergency control circuits will be isolated from the normal control circuits so that in the event of a fire, the control circuit will be unaffected and remain functional. Therefore, the probability of occurrence of a malfunction of equipment important to safety is not increased by this modification.

No new failure modes are introduced by this modification. This change represents an improvement over the existing condition of the circuit without introducing any new malfunctions.

This change does not affect the function of the feeder breakers or any mode of RHR. This modification ensures the capability of proceeding to shutdown cooling. No accidents of a different type than previously evaluated in the SAR are created by this change.

This modification does not reduce the margin of safety for the LPCI swing bus. The LPCI swing bus will operate as originally designed, however, the potential failure due to a control room fire is eliminated.

DCP 1545 Alternate Shutdown Cable Reroute

Description and Basis for Change

This modification corrected a deficiency in the safe shutdown separation requirements for 10 CFR 50 Appendix R and Branch Technical Position 9.5-1, Appendix A. Due to cable overfill concerns, the qualification of fire seals between the Cable Spreading Room (CSR) and the Main Control Room (MCR) was indeterminate. To resolve this deficiency, modifications were performed to allow the MCR and CSR to be classified as a single fire area. This DCP rerouted certain cables and modified instrument loops and circuits associated with the CSR and the Remote Shutdown Panel (RSP).

The Alternate Shutdown Capability System (ASCS) was originally designed for a fire in the MCR, control building HVAC room or in the control building chiller area. This modification implemented the changes required for use of the ASCS in the event of a CSR fire.

The ASCS and fire protection systems are not currently addressed in the NSOA or any DAEC Design Basis Document (DBD). The fire protection safe shutdown design bases are discussed in the DAEC UFSAR Section 9.5 and the methodology section of the Fire Hazards Analysis, FHA-200.

The design of the ASCS is discussed in UFSAR Section 7.4.2. In the event of a fire anywhere in the plant, the DAEC is able to achieve and maintain safe shutdown preventing fuel clad damage and loss of containment integrity. Therefore, there are no offsite consequences for design basis fires. The plant can achieve safe shutdown through several paths relying on either Division I or Division II trains or systems. For a fire in any given area, at least one division is protected or evaluated to be free from fire damage. The ASCS is designed to allow shutdown from outside the MCR using the RSP and several local panels. The DAEC ASCS uses primarily Division II systems.

Summary of Safety Evaluation

The instrument loops and circuits modified by this Design Change Package are not identified as initiators for any of the accidents described in the UFSAR or the NSOA. The cable reroutes and circuit modifications meet the design, material and construction standards applicable to these systems. The rerouted cables maintain required divisional separation and operate at the same voltage or current.

This modification does not prevent or degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents. Only one channel of one division will be out of service at any given time, thus there is no possibility for any system to affect its safe shutdown capability.

The modified instrument loops and rerouted control cables are part of systems that are required to mitigate several malfunctions currently described in the UFSAR and NSOA. The bases for these systems are not affected by this modification. The change maintains the design integrity of the primary containment system.

Final terminations of rerouted cables are the same as they were before the implementation of this modification. Rerouting of cables required for ASCS operation enhances the availability of the systems to perform shutdown from outside the control room in the event of a MCR/CSR fire.

DCP 1546 Main Steam Isolation Valve (MSIV) Solenoid Valve Replacement

Description and Basis for Change

MSIV solenoid valves are replaced every refueling outage to ensure their capability to function in their environment. Valves are no longer available to replace these valves on a like for like basis. Solenoid valves from a different vendor and in a different configuration have been used. These solenoid valves are in the DAEC Environmental Qualification (EQ) Program.

The design basis of the MSIVs is to close automatically to: prevent damage to the fuel barrier by limiting the loss of reactor coolant in case of a steam piping leak outside the primary containment; limit the release of radioactive materials by closing the nuclear system process barrier in case of a gross release of reactor fuel to the reactor cooling water and steam; and limit the release of radioactive materials by closing the primary containment barrier in case of a major leak from the nuclear system inside the primary containment.

The operational basis of the MSIVs is to open to allow steam to be transmitted from the reactor vessel to the turbine to allow for electrical power generation by turbine-generator systems.

Summary of Safety Evaluation

The accidents evaluated in the SAR are initiated by events that are independent of the MSIVs (except pressure boundary components). MSIV closure is required to mitigate the consequences of a pipe break either inside or outside of containment and a control rod drop accident. However, the MSIV performance remains unchanged with the solenoid valve replacement. Also, the replacement equipment is qualified to perform its safety function post accident.

The MSIV safety function is to close to isolate. The consequences of a malfunction is an increase in reactor coolant loss or radioactive release to the public. The solenoid valve repositions the pneumatic supply to the actuator prior to MSIV movement and therefore can not cause the MSIV to fail midposition. MSIV seating is dependent on the pneumatic supply to the actuator and occurs after solenoid valve repositioning.

The possibility of a different type of MSIV failure exists only if a different type of solenoid valve failure is possible. The design is improved with the solenoid valve replacement and functional and qualification testing have not identified a different type of solenoid valve failure.

The MSIV logic is unchanged by the solenoid valve change. The MSIV seating force and thus leakage are also unchanged.

DCP 1547 <u>Containment Isolation Valve Replacement</u>

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

This modification replaced the following containment isolation valves (CIVs) and respective operators:

1. CV-5704A: Drywell Cooling Loop "A" Well Water Return Isolation
2. CV-5718A: Drywell Cooling Loop "A" Well Water Supply Isolation
3. CV-1804A: "A" Recirculation Pump Mini-Purge Supply Isolation
4. CV-1804B: "B" Recirculation Pump Mini-Purge Supply Isolation
5. CV-4639: Reactor Recirculation System Sample Line Inboard Isolation.
6. CV-4640: Reactor Recirculation System Sample Line Outboard Isolation.

The air filters and pressure control valves on the instrument air supply to the control valves listed above were, in some cases, replaced as part of this modification.

The six CIVs were replaced with Anchor Darling double-disc gate valves that meet or exceed the original design requirements of the existing valves. Since the primary safety related function of the valves is to provide Primary Containment isolation, the Anchor Darling double-disc gate valves were chosen for their excellent isolation capabilities with minimal seat leakage under peak accident pressure.

The Well Water system isolation valve replacements were performed in accordance with ASME Section XI, 1980 Edition through Winter 1981 Addendum. The Well Water CIVs were installed in the same location as the existing valves. The 4" Anchor Darling double-disc gate valves are larger and heavier than the existing CIVs. The applicable stress and support calculations for these lines were revised to reflect the new configuration and the additional weight of the replacement valves to ensure that these Well Water lines remain seismically supported. Some support modifications were required to maintain the qualification of the piping supports.

The four small bore valves are CIVs for the "A" and "B" Recirculation Pump Mini-Purge Supply and the Reactor Recirculation Sample Line. The original design code for these valves is ASME Section III, 1971 Edition. The CIVs were installed in the same general location as the existing valves. A 3/4" test/vent line with two isolation valves and a pipe cap was also installed upstream of CV-4639. The Anchor Darling double-disc gate valves are heavier than the existing CIV globe valves. As a result, the applicable stress and support calculations for these lines were revised to reflect the heavier weight of the replacement valves as well as the LLRT test line to ensure that the lines remain seismically supported

Summary of Safety Evaluation

The CIVs replaced by this modification are not identified as initiators for any of the accidents described in the UFSAR or NSOA. The CIVs meet the original design codes. The materials, installation and testing requirements were in accordance with the original construction codes, thus assuring that the modified systems will be maintained as previously evaluated in the SAR.

The new Anchor Darling double-disc gate valves provide better seating capability and reliability than the existing globe valves. They will also meet or exceed the existing stroke time requirements of five seconds to close upon receipt of the initiating isolation signal. The isolation logic controlling these valves was not affected by this modification.

Review of the NSOA matrices and Chapters 3, 5, 6, 7, 9, 11 and 15 of the UFSAR demonstrate that the replaced CIVs are not initiators for any of the events previously analyzed. The equipment important to safety that could be impacted by the replacement of these CIVs are the valves themselves since they provide primary containment isolation and reactor vessel isolation functions, and the piping and supports where the valves are located since they provide a pressure boundary for the containment and the reactor vessel. The new CIVs use the existing limit switches for valve position indication and solenoid valves for valve position control. The new valves operate in the same manner as the previous valves and do not introduce any new failure modes.

The replacement valves meet or exceed all original design requirements and construction codes. The new valves are air operated and fail open or closed just as the respective original valves. The valves, piping and supports were installed in the same general area on the existing systems and maintain the original seismic requirements. The new valves are designed as containment isolation valves and are more than capable of providing the required level of containment integrity and leak tightness during normal plant operation, abnormal operational transients and accidents described in the DAEC UFSAR.

DCP 1548 Recirculation Speed Control Modification

Description and Basis for Change

This change removed all the GEMAC components used for the recirculation speed control system and replaced them with a digital controller. The change did not affect any of the interlocks for pump start, motor breaker trips or generator lockouts. The Recirculation Pump Trip (RPT) was not affected.

Summary of Safety Evaluation

The existing GEMAC control loops were replaced with digital controls as a result of the high failure rate of the GEMAC components. The digital controllers are more reliable and less susceptible to failures such as inadvertent runbacks or runaways. The digital controllers are designed such the controller will initiate a scoop tube lock and prevent an inadvertent runback upon a loss of power.

The new recirculation speed control circuitry interfaces with existing plant equipment in the same manner as before. The existing logic (e.g., runbacks and start-up) are performed by the new digital controller. Scoop tube locks are incorporated into the circuitry to prevent inadvertent runbacks or runaway events.

Installing a digital controller did not create an accident or event of a different type since the control loop can only fail such that a rapid recirculation pump speed increase occurs or a rapid runback occurs. These events are limited by the response time of the Bailey Positioner, which was not changed.

Interference from EMI and RFI has the potential to cause a simultaneous speed increase event. This is extremely unlikely and is not considered credible.

The digital controllers were procured and the software designed under the DAEC software control program even though these controls are classified as non-safety related. This was done to provide further assurance that the control loop will function properly and no common mode software problem exists.

TS 3.6.F requires that when both recirculation pumps are in steady state operation, their speed does not vary between them within certain limits. This is presently controlled by procedure. The installation of the new recirculation speed control circuitry does not change the operator's ability to control recirculation speed.

UFSAR Section 7.7.5.3 also adds a safety design basis ". . . no abnormal operational transient resulting from a malfunction in the recirculation flow control system can result in damaging the fuel or exceeding the nuclear system pressure limits." The modification did not involve any of the MG set drive motor trips, the RPT breaker trips or the primary system pressure boundary. These functions remain the same.

DCP 1549 Condenser Bay Fire Protection System

Description and Basis for Change

The purpose of this modification was to install the necessary plant equipment to enhance DAEC's ability to combat a Turbine Generator lubrication oil fire or an Electro Hydraulic Control (EHC) hydraulic oil fire in the Condenser Bay.

The sprinkler system installed in the Condenser Bay is a wet pipe sprinkler system consisting of non-seismic, non-safety related piping, supports, distribution network and sprinkler heads. It meets the density requirements specified and covers the entire Condenser Bay. Curbing was installed to prevent an oil pool fire on the Condenser Bay floor from spreading to other areas of the plant as well as to contain such a fire, thus minimizing the number of sprinklers involved in combating such a fire.

The Pilot Operated Relief Valve (PORV) drain piping was rerouted to an existing drain funnel on Deluge Systems 7, 12, 13, 14, and 15 to prevent back pressure. A check valve was installed under the drain funnel on Preaction System 1 to prevent overflow, while a drain funnel and check valve was installed downstream of the PORV and solenoid valve on Deluge System 18. A new Post Indicating Valve was installed to protect half the Turbine Building fire protection systems in the event of a leak in the main Fire Protection System Yard Loop. An isolation valve and a relief valve were installed on the pressure sensing line that is piped to PI-3307A. The relief valve is set at the system design pressure of 175 psig and discharges to the existing closed drain piping for Sprinkler System No. 4.

The design of the activities in this modification meet the applicable requirements of 10 CFR 50.48, General Design Criterion (GDC) 3 of Appendix A to 10 CFR 50 and Appendix R to 10 CFR 50. Appendix A to Branch Technical Position Auxiliary Power Conversion System Branch BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants prior to July 1, 1976," provides the basis for the design and implementation of fire protection systems at the DAEC.

Summary of Safety Evaluation

The operation of the Fire Protection System does not affect any of the inputs considered in the accidents analyzed in Chapter 15 of the UFSAR or the NSOA. The changes do not alter the interface between the Fire

Protection System and the plant and cannot cause an accident. The probability of occurrence of the accidents discussed in the UFSAR and NSOA is based on initial conditions and assumptions which do not depend on the end use of, or interactions with, the Fire Protection Systems.

This modification does not alter any assumptions previously made in evaluating the radiological consequences of an accident, nor will it play a direct role in mitigating the radiological consequences of an accident described in the UFSAR. The modification had no impact on systems, structures, or components important to safety. The Condenser Bay sprinkler system is not considered as a Fire Protection System required to protect safety related equipment

New accidents of a different type than those of UFSAR Chapter 15 have not been influenced by this modification. A malfunction of the sprinkler system, Post Indicating Valve or other activities performed by this modification do not introduce a new accident with respect to plant safety.

DCP 1550 <u>Regulatory Guide 1.97 Modifications</u>

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

IES Utilities Inc. committed to perform instrument modifications to bring several instruments into compliance with Reg. Guide 1.97. The modification provided electrical separation and reliable power to the Reg. Guide 1.97 instrumentation.

The Design Change Package provided Instrument AC power to the Emergency Service Water (ESW) flow indicators; added a physical barrier between the two Yarway reactor water level indication circuits; and removed non-safety related loads from Instrument AC to improve margin on the backup battery.

<u>Summary of Safety Evaluation</u>

The modification meets the original design, material, and construction standards applicable to the ESW flow indicators and reactor water level indication systems. Overall system performance is not affected because the same instrumentation is used for indication of the reactor water level and ESW flow. The only change involved physical separation and power sources.

The change establishes electrical separation between components in accordance with acceptable DAEC guidelines and ensures that power sources to these components are more reliable. This DCF reduces the probability of occurrence of a single failure disabling the associated flow or level indication.

Changes made by this modification do not change the operation of any system and therefore ensure the availability of equipment which is important to safety.

The loss of ESW flow indication does not affect the accident analysis. Similarly, the loss of level indication does not affect the accident analysis.

DCP 1551 <u>Head Spray Pipe Removal</u>

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

This DCP removed the head spray piping from the RHR header in the torus room. The activity consisted of cutting the head spray piping just above the header in the torus room. A flange was then welded to the cut pipe and a blank flange bolted to it. The piping was removed up to penetration X-17 and the penetration was capped. The head spray mode of RHR has been previously decommissioned; thus, it does not perform any design basis function.

<u>Summary of Safety Evaluation</u>

The consequences of an accident previously evaluated in the FSAR are not increased because the removal of the head spray piping, containment isolation valve, miscellaneous components, and the addition of the pipe cap and flange do not have adverse effect on the isolation of primary containment or the function of the RHR System. Thus, they will continue to fulfill their safety functions to mitigate the consequences of an accident. Additionally, the Head Spray mode of the RHR System has been previously decommissioned and its ability to perform any safety actions per the NSOA are not affected by this DCP.

DCP 1554 <u>Increasing MSIV Leakage Rate Limits and Elimination of LCS</u>

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

<u>Increasing MSIV Leakage Rate Limits</u>

This change increased the allowable leak rate specified in TS 4.7.A.2.c.3 from 11.5 standard cubic feet per hour (scfh) for any one Main Steam Isolation Valve (MSIV) to 100 scfh for any one MSIV with a total maximum pathway leakage rate of 200 scfh through all four main steam lines. If an MSIV exceeds 100 scfh, it will be restored to less than or equal to 11.5 scfh.

The previously allowed MSIV leakage rate was extremely limiting and could result in unnecessary repair and retest of the MSIVs. The increase in the allowable MSIV leakage rate should reduce the need for repair and, thereby, reduce dose to maintenance personnel consistent with as low as reasonably achievable (ALARA) principles.

The design basis of the MSIVs is to close automatically to prevent fuel damage and limit the release of radioactive materials. The operational basis of the MSIVs is to open to allow steam to be transmitted from the reactor vessel to the turbine to allow for electrical power generation by turbine-generator systems. The MSIVs are required to remain operable (i.e., remain capable of closing without sticking open) up to one hour after initiation of a Loss Of Coolant Accident (LOCA). The MSIVs are then required to close and remain closed without spuriously opening for up to 100 days following the accident in the design basis earthquake (DBE) environment.

Elimination of MSIV-LCS

This change eliminated the MSIV leakage control system (LCS) requirements from the TS. The main steam drain lines and the isolated main condenser are used as an alternate method for MSIV leakage treatment. This results in significant operational and maintenance benefits. LCS equipment was located in a high temperature, high radiation area, and was required to be environmentally qualified, necessitating extensive preventive maintenance.

The BWROG recommended the isolated condenser for MSIV leakage treatment. This leakage treatment method takes advantage of the large volume in the isolated main condenser to hold up the release of any fission products potentially leaking from the closed MSIVs. The main steam drain lines are employed to convey leakage to the condenser. Since simpler and less equipment is employed, the alternate method is more reliable than the LCS.

Summary of Safety Evaluation

The amendment does not involve a change to structures, components, or systems which would affect the probability of an accident previously evaluated in the DAEC UFSAR. The accidents evaluated in the UFSAR are initiated by events that are independent of the MSIVs.

The activity does increase the consequences of an accident evaluated previously in the SAR; however, the change does not involve a significant increase in the consequences of an accident previously evaluated and a TS change has been submitted and approved by the NRC. This change results in acceptable radiological consequences for the design basis LOCA which was previously evaluated in the UFSAR.

MSIV closure is required to mitigate the consequences of a pipe break either inside or outside containment and a control rod drop accident. The ability to close is unaffected by the change. The consequences of an operational transient are similarly unchanged.

The MSIV-LCS system has been replaced by a system with no logic or instrumentation. The equipment that is required for the system has been evaluated to be seismically rugged. The equipment required to be repositioned is limited and has been included in the inservice inspection and testing programs, as appropriate, to ensure its reliability. An alternate treatment path has been provided to ensure that a single failure will not disable the treatment system.

The consequences of a malfunction of equipment important to safety is increased; however, the change does not involve a significant increase in the consequences and a TS change has been submitted and has received NRC approval. The MSIV safety function is to isolate. The consequences of a malfunction is an increase in reactor coolant loss or radioactive release to the public. The radiological analysis has been performed with a single MSIV in each main steam line. The radiological consequences have been determined considering a single valve failure (malfunction of equipment). The change results in acceptable radiological consequences for the design basis LOCA which was previously evaluated in the UFSAR.

The BWROG evaluated MSIV leakage performance and concluded that MSIV leakage rates up to 100 scfh will not inhibit the capability and isolation performance of the valves. There is no new modification to the MSIVs which could impact their operability. The LOCA has been analyzed using the main steam piping and condenser as a treatment method to process MSIV leakage at the maximum rate of 200 scfh through all main steam lines. The MSIV-LCS lines connected to the main steam lines will be permanently closed to assure the primary containment integrity, isolation, and leak testing capability are not compromised.

Operation of the DAEC in accordance with the change will not involve a significant reduction in the margin of safety. The allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of bypass leakage assumed in the LOCA radiological analysis. Results of the analysis are evaluated against the dose requirements contained in 10 CFR 100 for the onsite doses and 10 CFR 50 Appendix A (GDC 19) for the Control Room and Technical Support Center doses. The margins of safety are not significantly affected because the dose levels remain well below the limits of 10 CFR 100 and GDC 19.

The treatment of MSIV leakage is performed by using the more effective alternate path via the main steam drain lines and condenser. This treatment method is effective for treatment of MSIV leakage over an expanded leakage range. Except for the requirement to assure that certain valves are opened to establish a proper flow path from the MSIVs to the condenser and that certain valves are closed to establish the seismic boundary, this method is passive and does not require any logic controls or interlocks. The safety significance of the MSIV leakage treatment system in terms of public risk was addressed in NUREG/CR-4330 which contains the evaluation for eliminating the MSIV-LCS. The conclusion was that the increased public risk is less than 1 percent. Therefore, this change does not involve a significant reduction in the margin of safety at the DAEC.

DCP 1555 Refuel Bridge Enhancements

Description and Basis for Change

This modification is being performed to improve refueling bridge performance and reliability. Refueling bridge failures and impact on future refueling outage duration will be reduced by this modification.

The improvements that were made include:

1. Replacement of the mechanical load cell on the fuel handling mast with a digital load cell.
2. Replacement of the refueling bridge air compressor with a new compressor and dryer.
3. Installation of isolation valves and filters on the air supply lines to each of the hoists.

The refueling platform and the refuel platform fuel handling hoist are classified as safety related components. The refuel platform air compressor and the refuel platform bridge drive are classified as non-safety related components.

This modification replaced or upgraded several non-safety related components on the refuel bridge. The replaced components do not affect the safety functions of the refuel bridge and DCP 1555 had no safety significance.

Summary of Safety Evaluation

The refueling accident that is contained in Section 15.7.1 of the UFSAR occurs when an irradiated fuel bundle is dropped from the refuel bridge onto the top of the reactor core. Section 15.7.1 states that this accident can only occur if the fuel assembly handle, the fuel grapple, or the hoist cable breaks. The changes made by this modification do not affect the load bearing capability of the refuel bridge or hoist.

After the fuel bundle is dropped, the consequences of the accident are in no way affected by the configuration of the refuel bridge or its components. The damage to the fuel when it is dropped is determined by the type of fuel and the energy which the fuel bundle possesses when it impacts the top of the core. Since the height at which the fuel is moved and the speed at which the fuel is moved are not affected by this modification, the energy of the fuel bundle, if dropped, is not affected and the consequences of the accident are not affected.

DCP 1555 replaced several non-safety related components on the refuel bridge. Each of these components perform the same functions as the original components. None of the components affect the safety functions of the refuel bridge and none of the bridge protection features or testing requirements are affected. The small amount of additional weight does not affect the seismic qualification of the refuel bridge. The changes increase the operational reliability of the bridge and will not increase the probability of a malfunction.

The changes do not reduce the margin of safety as defined in the basis for any TS. The load handling capabilities of the hoists and the load cell setpoints were not changed.

DCP 1556 Nitrogen Override Switch Logic Change

Description and Basis for Change

CV-4371A provides nitrogen ($N_2$) from the drywell supply accumulator located outside the drywell to the MSIV and Safety Relief Valve (SRV) accumulators, as well as CV-4639 located inside the containment. The valve is a containment isolation valve and will receive a closure signal on a Group 3 isolation. The override switch for CV-4371A was added to comply with our response to NUREG-0578. It was installed in 1980 and discussion was documented at that time to administratively control the use of the bypass switch, such that the override switch would not be taken to bypass unless a Group 3 isolation was verified.

A Group 3 isolation is received any time one of the following parameters is present: reactor water level less than 170", reactor building vent shaft high radiation (or equipment failure), fuel pool exhaust radiation high-high, or offgas vent pipe radiation high-high. The Group 3 isolation signals are initiated to protect the public at the site boundary from radiation.

This modification was performed to correct a problem in the existing logic which inadvertently allowed CV-4378B to remain open if the override switch for CV-4371A was placed in bypass prior to receiving a Group 3 isolation. The original design was not intended to override the logic to CV-4378B.

Summary of Safety Evaluation

This modification meets the current design, material, and construction standards applicable to the PCIS logic. This does not affect overall system performance in a manner which could lead to an accident or increase the probability of an accident previously evaluated in the SAR.

The modification ensures that a situation is not created in which CV-4378B would be opened or prevented from closing if the associated bypass switch for CV-4371A was taken to the bypass position. Thus the probability of having a containment isolation valve open when it is not required to be open is reduced. This reduces the probability of occurrence of a release path through CV-4378A and B, assuming a failure of CV-4378A. The activity meets the original design specification for material and construction practices.

The UFSAR already indicates that these valves have the ability to have their PCIS signals bypassed. The use of the bypass is administratively controlled and is to be used to insure that the plant has a controlled shutdown (i.e., cool down less than 100°F per hour).

The design change eliminates one method of bypassing a CV-4378B isolation signal. This does not affect any TS requirement or margin of safety.

DCP 1557 Utility Penetrations in Power Block

Description and Basis for Change

Routing of utility connections between plant walls required fire doors to remain open thus impairing fire protection systems. Addition of the utility penetrations on selected walls eliminates this problem.

Summary of Safety Evaluation

The drilling of core bores in safety related walls was performed in accordance with the applicable Design Guide requirements to ensure structural integrity of the walls in the proximity of the sensitive equipment. The fire plugs for the walls were procured as quality level II and were installed in accordance with the applicable Design Standards. The integrity of the walls in the areas of structural, separation, and fire protection was not degraded by this modification.

The addition of utility penetrations does not degrade the intended functions of the walls. Routing of cables and other utility connections through the penetrations were performed by the maintenance group in accordance with approved plant procedures. This prevents interaction of the utility connections with the safety related systems, components, or equipment located in close proximity.

Utility penetrations installed in accordance with design guidelines/qualifications and design standards do not reduce the margin of safety as defined in the basis for any TS.

## Section B - Procedure/Miscellaneous Changes

SpTPs:

193 Vessel Bottom Head Coolant Temperature Measurement

Description and Basis for Change

Special Test Procedure (SpTP) 193 was written to determine the flowrate of the coolant through the reactor bottom head drain pipe. Information gathered from this test was used to throttle valve V27-0015 to a position where the flowrate through the reactor bottom head drain line is 30 gpm or greater when the Reactor Water Cleanup (RWCU) system flow is 150 gpm, recirculation pumps not operating and recirculation pump suction and discharge valves fully open. This ensures the temperature indication as provided by TE2750 for reactor bottom head coolant water is accurate.

Summary of Safety Evaluation

This test was intended to verify and adjust, as necessary, the flowrate of reactor coolant through the reactor bottom head drain pipe to ensure accurate measurement of reactor bottom head coolant water temperature. The verification and adjustment of this flowrate does not affect the system performance in such a way as to increase the probability of occurrence of an accident.

This special test did not add additional stresses or loads to the bottom head drain pipe and did not compromise reactor vessel integrity (the bottom head drain pipe specifically). This special test was performed when the reactor was shut down and depressurized during a refueling outage with all fuel removed from the reactor vessel. This special test did not result in any system vibration, water hammer, thermal cycling, or erosion/corrosion of the reactor drain pipe.

This test had no impact upon the assumptions made previously in evaluating the radiological consequences of an accident described in the UFSAR. The test did not affect the mitigation of radiological consequences described in the UFSAR. It did not introduce the potential for any radiological release to the environment that bypasses the standby gas treatment or standby filter unit systems. The test did not have any effect on any fission product release as it merely ensured the limitations of reactor vessel stresses due to thermal stresses were not exceeded.

This special test did not change any of the equipment associated with the reactor temperature indication. The test verified that the flowrate through the reactor bottom head drain pipe is adequate to ensure that TE2750 is providing accurate and reliable indication. There were no separation criteria associated with the reactor temperature indication instruments. The temperature indication system is not designed or maintained as environmentally qualified instrumentation.

SpTP 193 did not impose additional loads on the reactor vessel system, did not delete or modify any reactor temperature sensing equipment, and did not reduce the temperature sensing redundancy or independence. The test did not change the required frequency of operation or impose severe testing requirements on the reactor temperature sensing equipment.

The probability of a malfunction of the reactor vessel temperature sensing equipment or a failure of TE2750 or companion temperature sensor used in comparison of differential temperatures was not affected by this test. The reactor bottom head coolant temperature is required to ensure that over stressing of the reactor vessel material does not occur during plant heat up and cooldown, or during the starting of an idle recirculation pump. The verification of accurate and reliable temperature at the location sensed by TE2750 would not make the failure of TE2750 more severe.

SE 93-22 UFSAR Change: Emergency Service Water Flow Requirements

Description and Basis for Change

The Emergency Service Water (ESW) flow requirements contained in UFSAR Chapter 9 have been revised. IES Utilities Inc. contracted a complete analysis of ESW heat loads versus room cooler performance. The analysis for the room coolers, including credit for the insulation added to the HPCI and RCIC turbine revised RHR and Core Spray (CS) motor heat loads, considered river temperatures from 80°F to 100°F with ESW side fouling factors from 0.0 to 0.001. This UFSAR change incorporated the results of that analysis.

UFSAR Table 9.2-1 has been revised to include ESW flow requirements based on an ESW side fouling factor of 0.001 instead of 0.0005. The effects of temperature stratification in the HPCI, RCIC, and RHR and CS rooms were also included by increasing the flow requirements by an amount required for an increase of 5°F in the river water temperatures from those shown in the design calculations. A column has been added to list the flows desired to maintain a 3 ft/sec tube velocity. The text in UFSAR Chapter 9 has been revised to show the new flow values.

Summary of Safety Evaluation

The ESW system is a support system for major safety systems. Revising the required flow rates does not affect the integrity of the reactor coolant pressure boundary. Therefore, it cannot cause a LOCA, steam line break, control rod drop or refueling accident.

The ability of the ESW system to support safety systems has been evaluated at the new ESW flowrates. The ability of the Emergency Core Cooling System (ECCS) and safety systems to mitigate the consequences of an accident are not reduced.

The safety systems will be fully operational at the new ESW flow rates so the probability of malfunction is not increased. The change to the SAR does not change the consequences of a failure of the ESW system. The change in the required ESW flow rates to the safety systems has been determined to not adversely affect their operation.

The margin of safety, as defined in the bases of the DAEC TS is not reduced because the safety systems supported by ESW remain fully operational when ESW is operated to provide the new cooling water flowrates.

SE 94-01 UFSAR Change: MSIV leak rate

Description and Basis for Change

UFSAR Section 5.4.5.4 contained a description of a test method that could be used to determine the MSIV seat leakage rate. This methodology used reactor pressure to determine the leakage rate across the MSIVs. This method was developed prior to the present requirements concerning leakage rate testing and did not meet the current requirements. This method has not been and cannot be used to determine the MSIV leakage rates.

10 CFR 50, Appendix J contains the current regulatory requirements for leakage rate testing of containment isolation valves. The DAEC TSs and UFSAR Section 6.2.6 incorporate the requirements of Appendix J. DAEC has been granted an exemption to the requirements of Appendix J to allow testing of the MSIVs at a reduced pressure and this has been incorporated into the TSs.

Therefore, Section 5.4.5.4 of the UFSAR was changed to remove the description of the test method and to reference the testing requirements contained in Section 6.2.6 and the TSs.

Summary of Safety Evaluation

The UFSAR change only removed a description of MSIV test method in Section 5.4.5.4 and referenced the testing requirements contained in

Section 6.2.6 and the TSs. The MSIVs will continue to be tested in the same manner and the allowable leakage rate through the MSIVs will remain the same. Removing the test method currently contained in Section 5.4.5.4 had no effect on any plant equipment and did not increase the consequences of an accident. The probability of a previously evaluated accident occurring did not increase.

Changing UFSAR Section 5.4.5.4 did not increase the probability of occurrence of a malfunction of equipment important to safety. The consequences of a malfunction of equipment important to safety were unaffected since there were no changes in plant configuration, operation or testing.

The UFSAR change did not create the possibility of an accident of a different type than previously evaluated. The change did not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated because the MSIVs will continue to be tested in the same manner.

The test method, which was described in UFSAR Section 5.4.5.4, was not described or referenced in the TSs or Bases. UFSAR Section 5.4.5.4 was changed to reference the testing requirements contained in the TSs. The allowable leakage limits and margins of safety were not affected.

SE 94-04 GL 89-10: Motor Operated Valve Program Change

Description and Basis for Change

Each of the Motor Operated Valves (MOVs) included within the scope of the GL 89-10 program was re-evaluated using, primarily, the guidance provided by the Boiling Water Reactors Owners' Group (BWROG) reports on the operational design basis of selected safety related valves. This guidance was used to determine if each MOV has an active safety function. For those MOVs that were determined to have no active safety function, a further review of DAEC specific design and licensing bases considerations was undertaken.

There were originally 107 MOVs within the scope of the GL 89-10 program. The 107 valves are contained in the Containment Atmosphere Dilution (CAD) system, the Emergency Service Water/Residual Heat Removal Service Water (ESW/RHRSW) system, the High Pressure Coolant Injection (HPCI) system, the Core Spray (CS) system, the Main Steam Line Drain (MSLD) system, the Main Steam Isolation Valve Leakage Control System (MSIV-LCS), the Reactor Building Closed Cooling Water (RBCCW) system, the Reactor Core Isolation Cooling (RCIC) system, the Reactor Recirculating system, the RHR system, and the Reactor Water Cleanup (RWCU) system. A common feature of the subject MOVs is that they are normally in the position (open or closed) required to allow their respective system to perform its safety related function, and are only infrequently moved from those positions to support other plant requirements such as testing, maintenance, or pressure control. An exception is the RHR heat exchanger inlet valves, which the BWROG design basis review indicated are normally open and are required to stay open during LPCI operation.

The HPCI, RHR (LPCI mode) and CS systems, along with the Automatic Depressurization (ADS) system, comprise the Emergency Core Cooling System (ECCS) network. (RCIC is not part of the ECCS, but is also included in the network.) In combination, these systems act to ensure that any break, across the whole spectrum of Reactor Coolant Pressure Boundary (RCPB) pipe sizes, can be mitigated to the extent necessary to ensure maintenance of Peak Clad Temperatures (PCTs) and offsite release rates less than acceptable limits. This network is designed to accomplish its design objective, given a postulated Design Basis Accident (DBA), i.e., the largest RCPB break with a concurrent Loss of Offsite Power (LOOP) and the failure of any active safety component (a single active failure).

The ECCS network is designed so that, taken together, no single active failure will result in preventing the network from performing its intended function for the whole range of pipe breaks, up to and including the DBA. The network is also designed to allow the subsystems within the network to be removed from operation for activities such as maintenance and testing. When a subsystem is so removed, it is not counted on to contribute to the overall network objective. In other words, the removal of a single subsystem for such purposes is equivalent to a single active failure that would render a subsystem unavailable, and a single active component failure in addition to a subsystem being removed from service is not required to be postulated, and is beyond the intent of the single failure criteria.

Summary of Safety Evaluation

Of the ECCS network subsystems in question (i.e., HPCI, RCIC, RHR, and CS), only the HPCI system serves as the initiator of an accident or transient that is evaluated in Chapter 15 of the UFSAR. The ECCS network subsystems are designed to respond to a whole variety of accidents and transients described in the SAR. The most limiting of these transients and accidents is the design basis LOCA, concurrent with a LOOP and any single active component failure. The adequacy of the ECCS network to respond to and mitigate any of the postulated events is not diminished since LOCA events are not evaluated assuming ECCS subsystems are initially in modes other than standby readiness. In fact, existing LOCA analyses assume the subject MOVs are initially in positions corresponding to standby readiness, and the MOVs are not required to reposition upon system actuation.

Some of the MOVs that are the subject of this evaluation contribute to safety related functions in addition to the ECCS function (i.e., containment isolation and the post LOCA containment heat removal function of the RHR system). However, these functions are only needed if a LOCA is postulated to occur and an acceptable level of public safety is maintained since the time that ECCS subsystems are in modes other than standby readiness is minimal.

Because the MOVs are not required to change position to support any safety functions, the MOVs are not considered active safety components. Therefore, the subcomponents of the MOV assembly required to change a valve's position (i.e., the motors and operators) can be classified as non-safety related, and postulated failures of these components will not affect the ability of safety related equipment to perform its safety function.

No physical changes to the plant are being made as a result of this evaluation; only a redefining of the MOVs to be included in the GL 89-10 program. Since all of the MOVs will be in the positions assumed during any accident analyses and will be operated in the same manner as is assumed in the accident analyses, there is no opportunity for creating a new accident.

SE 94-06 <u>ODR 94-041: Main Steam /Safety/Relief Valve Tailpipe Pressure Switches not EQ Qualified</u>

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

At the DAEC, there are a total of twenty-four (24) tailpipe safety/safety relief pressure switches, three pressure switches on each of the six (6) SRV discharge lines and three on each of the two (2) short safety valve tailpipes. The tailpipe pressure switches are arranged in a two-out-of-three logic, which means that any two out of three switches located on a single tailpipe provide an input into the Low-Low-Set (LLS) arming circuitry, which is safety related. Any of the 6 SRVs opening (PSV4400, 4401, 4402, 4405, 4406, 4407) will satisfy part of the LLS arming logic. It also requires a reactor high pressure (1055 psig) signal and it is then that LLS is "armed" and begins to control reactor pressure.

The SRV valve position indication for all eight SRVs (6 safety relief valves and 2 safety valves-PSV4403, 4404) are designated as post-accident monitoring instrumentation displayed at Control Room panel

1C-003. This function is considered important to safety and requires the instruments to be environmentally qualified.

In 1992, 6 of 24 Main Steam Safety/Relief Valve (SRV) Tailpipe pressures switches were replaced and their electrical connector seal assemblies were assembled using an unqualified thread sealant.

Summary of Safety Evaluation

The accidents that have been evaluated in which the SRVs would be required to function include: Control Rod Drop, LOCA inside primary containment and LOCA outside containment. As stated in the UFSAR, the safety objective of the nuclear system pressure relief systems is to prevent overpressurization of the nuclear system to prevent a failure that could result in the uncontrolled release of fission products. This safety objective is unaffected by the current configuration. Use of an unqualified thread sealant on the subject pressure switches does not affect the probability of the occurrence of an accident.

The specific condition that can potentially affect the subject pressure switches is the small break LOCA. This is the only condition under which the pressure switches are required to operate and create an environment of high temperatures which may break down the unqualified thread sealant and allow moisture intrusion into the connector assembly.

It has been determined that the function of the pressure switches, even if moisture intrusion were to occur, would not be affected. The pressure switch manufacturer concurred with this conclusion. The pressure relief mode of the SRVs remains unaffected as well as the LLS and ADS functions. The LLS valves, PSV4401 and PSV4407 have fully qualified pressure switch connections and are unaffected by the unqualified thread sealant concern. Therefore, LLS is expected to operate as designed if called upon in an accident condition.

SE 94-07 UFSAR Change: Possible Permanent Installation of the Residual Heat Removal (RHR) to Spent Fuel Pool Cooling (SFPC) Cross-connect Spool Piece

Description and Basis for Change

This Safety Evaluation was written to support an UFSAR change. The UFSAR change consists of two topics:

1. To clarify the UFSAR description of the possible permanent installation of the RHR to SFPC cross-connect spool piece. The spool piece and interfacing piping are designed to allow the spool piece to be

installed or removed. When removed, the interfacing piping has blank flanges installed on the flange ends.

2. To address when the cross-connect spool piece may be used to allow the RHR System to supplement fuel pool cooling.

The SFPC System performs a safety action for various planned operations and does not perform any safety functions for transients, accidents or special events. The "cleanup" function of the SFPC System is not a safety action. The RHR System performs actions for various planned operations, transients, accidents and one special event - Loss of Control Room habitability. There are no safety actions which are performed which require the RHR System to supplement the SFPC System

The impact on the ECCS design bases concerning the installation of the spool-piece can be limited to the impact on the ability of the RHR System to perform its various functions while connected to the SFPC System. Since the other ECCS remain fully available (dependent on plant conditions and the TS out-of-service times) to fulfill the ECCS design basis in accordance with TS, the installation of the spool piece, which only affects the RHR System portion of the ECCS, does not prevent the ECCS from fulfilling its design basis function.

Summary of Safety Evaluation

Per the NSOA and the SAR, the RHR System performs functions to mitigate the consequences of transients and accidents. When the spool piece is installed but the isolation valves are closed, there is no effect on the RHR System. Supplementing spent fuel pool cooling can be initiated when the spool piece is installed, the isolation valves are open and only when the LPCI mode of RHR is not required to be operable.

Per the NSOA and the SAR, the SFPC System does not perform a function to mitigate the consequences of an accident. However, per the SAR, the SFPC System does perform functions to mitigate the consequences of transients (i.e., a loss of cooling and/or inventory). These events are evaluated previously in the SAR.

The SFPC System is not considered to contain equipment important to safety since it does not mitigate any accidents per the NSOA or the SAR. However, fuel pool level requirements are specified in TS 3.9.C. The requirement is that fuel pool level must be maintained greater than 36 ft. The use of the RHR System to supplement fuel pool cooling does not affect the total amount of water in the SFPC System and/or RHR System and the water volume remains constant; thus, the fuel pool level is not lowered when the RHR System is being used to supplement fuel pool cooling and this margin of safety is maintained. Additionally, other margins with respect to SFPC cooling capabilities (i.e., flow, temperature, pressure, etc.) are not adversely affected.

SE 94-10 <u>UFSAR Change: Shift Technical Advisor (STA) Reporting</u>

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

The Nuclear Generation Division was reorganized to enhance organizational effectiveness, and one of the changes was to move the responsibility for the STAs from the Engineering Department to Operations. The STAs now report to the Operations Supervisor instead of the Manager of Engineering.

<u>Summary of Safety Evaluation</u>

Changing the responsibility of supervising the STAs from the Manager of Engineering to the Operations Supervisor does not affect any malfunction of equipment or any accident sequences. The change is merely organizational and does not affect the ability of the STAs to perform their duties or affect any plant equipment or its operation.

The responsibilities of the STA position as defined in NUREG-0737 to provide technical assistance to the operating crew during normal and abnormal operating conditions are not affected by this UFSAR change. There are appropriate procedures in place that define the roles and responsibilities of the STA.

SE 94-19 <u>RTS-276: Shutdown Cooling Safety Limit</u>

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

This evaluation was written to support a TS Bases change regarding the 135 psig reactor pressure isolation of the RHR Shutdown Cooling (SDC) suction valves. The reason for the change is to clarify the bases for the RHR shutdown cooling reactor vessel pressure safety limit and for the RHR shutdown cooling isolation actuation setpoint.

The scope of the activity involved the clarification that, even though the design rating of the RHR SDC piping is 175 psig, the ASME Code allows brief pressures above the design rating. This did not involve any hardware changes to plant equipment, changes to the setpoints of the isolation instrumentation or changes to any operational characteristic of the RHR SDC isolation valves or the RHR System in the SDC mode.

The existing TS safety limit which specified that the "reactor vessel dome pressure shall not exceed 135 psig at any time when operating the

Residual Heat Removal pump in the shutdown cooling mode" was not affected. The safety limit is designed to prevent overpressurizing the RHR SDC suction piping. This is accomplished by reactor pressure instrumentation causing the RHR SDC suction isolation valves to close when a reactor steam dome pressure equivalent to 135 psig is sensed.

Although the isolation is considered to be a safety limit per TS, it is not a safety function with respect to the design basis of the RHR System as noted in Section 3.0.2 of NEDO-10139. With reactor dome pressure at 135 psig and reactor water level at normal level, the maximum RHR SDC suction piping pressure will be approximately 178 psig.

The key process variables for which limits must be observed in each operating state for the RHR System are not affected. The RHR SDC system still operates with reactor pressure less than 135 psig and flow rates greater than 4,000 gpm. The key process variable to keep piping pressure below the design rating of the piping is not met; however, pressures in excess of the piping design rating are acceptable as specified by the ASME Code within certain time restrictions as mentioned in the Code and as specified in the TS Bases change. The TS Bases change does not impact the actual pressure which the shutdown cooling piping will experience (i.e., no reactor pressure setpoints or valve isolation logics are modified). The change allows the ASME Code to be used as the basis for determining what pressures will not damage the piping rather than using the design rating of the piping. This maintains the design intent of the system isolation.

Summary of Safety Evaluation

The system will be operated within the safety limit. The ability to isolate the SDC suction piping from the reactor vessel (a non-safety, non-primary containment isolation function for the 135 psig signal) to avoid damage to the RHR SDC suction piping is not adversely affected. Thus, RHR SDC suction piping will not be damaged and will continue to fulfill its safety function to mitigate the consequences of an accident.

The RHR System has TS requirements to prohibit RHR SDC operation or opening of the SDC suction isolation valves whenever reactor pressure is greater than 135 psig. This activity does not affect the 135 psig safety limit nor cause the RHR System, SDC mode, to operate at a reactor pressure closer to the 135 psig limit than before the activity. The activity only clarifies that the ASME Code allows brief pressure above the design rating of the RHR SDC piping. Thus, the margin of safety is not reduced.

SE 94-23 SEP 305, ECCS Suction Strainer Blockage

Description and Basis for Change

This safety evaluation was written to implement Emergency Operating Procedure (EOP) Support Procedure (SEP) 305. This procedure was developed to provide supplemental guidance to the Control Room operator on detecting symptoms of clogged Emergency Core Cooling System (ECCS) strainers and unclogging ECCS strainers. The procedure will be used in conjunction with EOPs to ensure that plant parameters are restored and/or maintained within acceptable limits

SEP 305, "ECCS Suction Strainer Blockage," utilizes one loop of the RHR System to backflush the suction strainers in the other loop of the RHR System. The backflush pressure and flow in the RHR loop which is providing the backflush of the suction strainers in the other RHR loop is provided from either the RHR or RHRSW pumps.

Once backflush pressure is established, several manual valves (i.e., the valves in the RHR flush lines which are designed to flush the RHR loops to the radwaste system) are opened to establish a flowpath via a 4" line around the RHR pump and its discharge check valve and then to the torus suction strainer. SEP 305 then provides for a one-to-two minute backflush.

The RHRSW System is designed to provide water to the RHR System for injection into the reactor vessel; thus, there is no effect of this backflush procedure on the RHRSW System.

The torus is affected due to water being discharged into the torus via the RHR suction strainers. When one RHR System loop is backflushing the other RHR System loops' suction strainers, there is no effect on the torus water level because the RHR System is taking a suction on the torus and backflushing an RHR System loop's suction strainers, a water source external to the primary containment is being added to the water inventory in the primary containment (i.e., the torus). This could cause torus water level to increase to unacceptable levels. However, EOPs currently address this situation and provide for corrective actions to reduce torus level within acceptable limits. Additionally, SEP 305 includes a caution to "Secure backflushing if torus level increases above 16 feet."

Summary of Safety Evaluation

The RHR System does not contribute to the probability of occurrence of an accident as evaluated in the SAR. Per the NSOA and the SAR, the RHR System performs functions to mitigate the consequences of transients and accidents. The backflush procedure occurs during the use of EOPs for a "beyond-design-basis" event. SEP 305 prescribes that the RHR System will already have performed its safety functions with the plant parameters being controlled within specified limits prior to initiating the backflush; thus, the consequences of an accident are being adequately mitigated.

Per the NSOA and the SAR, the RHR System contains equipment important to safety. Backflushing will decrease the probability of occurrence of a malfunction of the RHR pump suction strainer and RHR pumps. SEP 305 prescribes that the RHR System will already have performed its safety functions with plant parameters being controlled within specified limits prior to initiating the backflush. During the backflush the RHR System will not be permanently adversely affected and after the backflush the RHR System will be fully restored such that it will be able to perform its various safety functions. This ensures that the margin of safety for proper operation of the RHR System and thus for the overall safety of the plant is not reduced.

Additionally, an accident where the RHR pump suction strainers are completely plugged (or in some other way the RHR pump suction is blocked) is not evaluated in the SAR. Therefore, backflushing will decrease the possibility for an accident of a different type than any evaluated previously in the SAR.

The possibility of damage to the torus or an adverse increase of torus water level is not created by this backflush procedure. An overpressurization of the RHR pump suction piping is not credible.

SE 95-02 <u>Intake Structure Recoating</u>

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

During the 1995 refuel outage, the River Water Supply (RWS) pit was de-watered. The metal and concrete surfaces were prepared for painting (with the exception of the traveling screen). The surfaces were then primed and painted per manufacturer's procedures. This reduces the risk of zebra mussels attaching to the intake structure pit.

UFSAR Section 17.2.9.5, Special Protective Coatings (Paint), requires that the application of a special protective coating be controlled as a special process when the failure (i.e., peeling or spalling) of the coating to adhere to the substrate can cause the malfunction of a Quality Level I structure or component. Priming and painting of the concrete intake structure pit walls, piping and grating below the water line in the intake pit was performed in accordance with IES Utilities Inc. approved vendor procedures. In addition, coating thickness was verified by an IES Utilities

Inc. representative to ensure acceptability of the coating process. Therefore, application of the zebra mussel primer and coating were controlled as a special process and met the UFSAR requirements.

If a coating is to fulfill its function of protecting a metallic or concrete substrate, it must adhere to this substrate for the expected service life. The adhesion of the paint was tested at an independent laboratory per appropriate ASTM Standards.

The application of this coating in the intake structure does not affect the operation of any equipment described in the UFSAR.

Summary of Safety Evaluation

Coating the substrate in the RWS pit impacts, at most, the RWS System's ability to support accident mitigation. The failure of the RWS system cannot cause any accident evaluated in the SAR (rod drop, LOCA, fuel handling).

Application of the coating was controlled as a special process per UFSAR Section 17.2.9.5. Therefore, the probability of a major coating failure is reduced to an insignificant level. The positive effects of the coating to reduce future fouling due to zebra mussels helps ensure the historic excellent performance of the service water systems is maintained.

The reliability of the service water systems is not adversely affected by the addition of the coating. The Emergency Service Water (ESW) and RHR service water strainers are capable of removing any coating that might fail. The service water systems are support systems. Their failure cannot cause an accident and the most severe consequence of a coating failure would be the loss of these systems.

The application of coating in the intake structure pits would, at worst, be expected to introduce paint chips into the system that the ESW and RHR service water strainers will have to remove.

SE 95-04 Core Operating Limits Report for Cycle 14

Description and Basis for Change

In accordance with DAEC TSs, a Core Operating Limits Report (COLR) was prepared to support the addition of new fuel to the core and the relocation, i.e., shuffling, of the existing fuel that remains in the core for the next operating cycle. The COLR contains the thermal limits for the fuel, which are derived from the results of the analysis of the limiting operating transients and accident analyses in the UFSAR. The cycle-specific analysis of these limiting transients and accidents is performed using NRC-approved methods, as described in GE's Standard Application for Reactor Fuel (GESTAR: NEDE-24011-P-A), and the results are presented in the Cycle 14 Supplemental Reload Licensing Submittal (SRLS) for the DAEC. The fuel design used for Cycle 14 was of the same fuel type (GE10) as that loaded in previous reloads at the DAEC (i.e., Cycles 11, 12, and 13). These fuel designs are licensed by the NRC via GE's topical report, NEDE-31152P, GE Fuel Bundle Designs.

Summary of Safety Evaluation

The GE10 fuel design met al! requirements for fuel designs and was essentially a like-for-like replacement of the fuel previously loaded in the DAEC core. Compliance with the thermal limits for this core design ensures that the fuel design requirements are satisfied during reactor operation in all applicable Operating States in the NSOA. These thermal limits were derived using NRC-accepted methods, which demonstrate, in the SRLS, that the consequences of the limiting events in UFSAR Chapter 15 are within the acceptance criteria for such events.

Given that the GE10 fuel types loaded in this reload met all acceptance criteria for fuel designs and were manufactured/constructed under an NRC-approved Quality Assurance program, the probability of a failure of the fuel cladding (the equipment important to safety), when operated in accordance with the thermal limits provided in the COLR was not increased from that previously evaluated. Also, the ASME Vessel Overpressure analysis in the SRLS demonstrated that the peak RPV pressure was well within the design allowable value.

SE 95-05 Fire Water System Modification (DDC 2791)

Description and Basis for Change

The Control Building HVAC room changes were installed to address a problem identified in a flooding study performed in support of the DAEC Individual Plant Examination (Generic Letter 88-20). This report identified a sequence where rupture of fire system piping causes water to flow into both essential switchgear rooms. Since a large volume of water was calculated to enter the rooms, likelihood of damage to electrical equipment required for safe shutdown was estimated to be relatively high. This scenario was the largest contributor to core damage frequency as calculated using the latest revision of the DAEC Probabilistic Safety Analysis (PSA, Rev. 3). This modification eliminated the core damage sequence through closure of existing fire system isolation valves. In the event of a fire system pipe rupture within the Control Building HVAC room, the volume of water released is greatly reduced.

The Standby Gas Treatment (SBGT) room change was performed to prevent inadvertent actuation of the carbon bed deluge system. Damage to the SBGT system from this malfunction was estimated to be substantial, and might have required a plant shutdown to rectify the problem. The modification to the Standby Filter Unit (SFU) deluge system in the Control Building HVAC room prevents damage from this type of malfunction.

Summary of Safety Evaluation

The activity involved closure of three manually operated valves that supply fire water to five sprinkler and deluge fire suppression systems. Procedural guidance was provided for opening the appropriate valve in the event of a fire in an area served by these systems. This change did not increase the frequency of fire initiation events, nor did it increase the frequency of large fires with capability to disable safe shutdown equipment.

The frequency of fire related events currently expected is essentially the same as the frequency of fire events expected prior to the change. By eliminating the possibility of an HVAC room fire line rupture causing damage to essential switchgear room equipment necessary for safe shutdown, protection of the public from undue releases of radiation is enhanced.

The severity of sprinkler and deluge water line rupture was reduced by this activity. The amount of water released by rupture of either line in the Control Building HVAC room with their respective isolation valves closed is inadequate to reach the essential switchgear room via the HVAC duct chase.

Severity of design basis accidents described in the SAR (i.e., LOCA, Rod Drop Accident, Main Steam Line Break Outside Containment, Fuel Bundle Drop Accident) were not impacted by this change. The fire water system is not credited as a source of reactor vessel water inventory. Fire water system penetration seals that serve as a barrier to radioactive material release are not affected by this change. No physical change was made other than to close existing valves that were previously open.

The capability to detect and suppress fires initiated in the Control Building HVAC room and the SFU and SBGT carbon bed filters is unchanged.

SE 95-06 Emergency Service Water (ESW) Flow Measured at Refuel Floor Less Than Required

Description and Basis for Change

This evaluation was performed to determine if flow rates from ESW to the Spent Fuel Pool (SFP) less than 43.11 gpm constituted an unreviewed safety question.

UFSAR Section 9.1.3.3 states: "The makeup flow rate to maintain a pool level of 36 ft would be 38.8 gpm. In addition to the makeup capabilities of the fuel pool cooling and RHR systems, makeup is available from the emergency service water system." (Note: As a result of the installation of high density fuel racks into the SFP per DCP 1538, the amount of flow required to makeup for evaporative losses under assumed worst case SFP heat load conditions has increased from 38.8 gpm to 43.11 gpm.)

The SFP itself is designed to withstand seismic loads, and the Spent Fuel Pool Cooling (SFPC) system is designed so its connections to the SFP are located in such a way that if the nonseismic SFPC system piping were to fail, the fuel in the pool would not be uncovered. This is accomplished by locating the SFPC system piping connections high in the pool, and by providing vacuum breakers on that piping so that water can not be siphoned out of the pool. The spent fuel residing in the SFP is considered adequately cooled if it is fully covered. Therefore, the function of removing spent fuel decay heat is primarily performed by the non-safety related SFPC system, but is supplemented by the safety related RHR system. The SFP cooling assist mode of the RHR system, however, is not considered a safety related function.

Since adequate cooling is accomplished by keeping the spent fuel covered, a source of makeup water from a fully qualified source (i.e., the ESW system) is provided. By definition, the function of providing makeup water to the SFP is a Safety Class 3 function. In the event of a failure of the non-safety related SFPC system and non-safety related sources of makeup water, the SFP would heat up until boiling occurred. The spent fuel is kept adequately cooled by providing an amount of makeup water from the ESW system equal to the evaporative losses (i.e., 43.11 gpm under worst case spent fuel loading conditions.)

A conservative estimate of the SFP heat up rate after a loss of SFPC can be found in Appendix 3 to Abnormal Operating Procedure 149 (Loss of Decay Heat Removal). For periods of time greater than 60 days after shutdown, the SFP heat up rate is less than 2.5 °F. per hour. If the SFP water temperature is maintained less than the UFSAR assumed maximum (150°F), then over 24 hours after the total loss of SFPC will elapse before the onset of pool boiling and the time when SFP level would begin to decrease. During this 24 hour period, the top priority would be to establish the SFPC assist mode of RHR as the primary SFP heat removal method. When utilizing the SFPC assist mode of RHR, the Low Pressure Coolant Injection (LPCI) mode would be considered inoperable, requiring the plant to enter a 7 day Limiting Condition for Operation (LCO). This would meet the intent of commitments made to shut down the plant when the SFPC assist mode of RHR is utilized.

The method of establishing the SFPC assist mode of RHR (without shutdown cooling [SDC] in service) is contained in Section 8.2 of Operating Instruction 149 (RHR). Since the RHR to SFP spool piece is currently installed, it was judged that this task could easily be accomplished within the required 24 hour (time to boil) period. During the 7 day LCO period, every effort would be made to restore the SFPC system, or to establish an alternate means of SFP cooling. If at the end of the 7 day period RHR was still being utilized in the SFPC assist mode, then the plant would be required to shut down to the hot condition within 12 hours, and to the cold shutdown condition within the following 24 hours (reference TS 3.5.A.6). In order to reach the cold shutdown condition, the SDC mode of RHR would be required. (Note that this mode of RHR is also not considered to be a safety related mode.) Since it is not possible to "split" the RHR system so that both the SDC and SFPC assist functions can be accomplished simultaneously (with the vessel head installed), the SFPC function would, by necessity, be required to be terminated at that time.

Another option available to the plant would be to utilize the alternate SDC function. This would involve raising vessel level above the steam lines, flowing through an open Safety Relief Valve (SRV) to deposit the decay heat into the suppression pool, and removing the decay heat via the Suppression Pool Cooling mode of RHR. (Note that the Suppression Pool Cooling mode of RHR is a safety related mode, and that it is possible to "split" the RHR system so that both the SFPC assist function and the Suppression Pool Cooling function can be accomplished simultaneously). In any event, it was judged highly likely that some sort of SFP cooling would be reestablished within the 7 day period.

Various other activities could be undertaken in the initial 24 hour period in order to preclude the need for making up with the relatively low quality ESW water. These actions would include:

1. restoring the SFPC system,
2. obtaining and installing a portable pool cooling unit, and

3. providing SFP cooling via innovative methods such as "feed and bleed" with Condensate/Service Water (CSW) while draining from the pool an equal amount via one of the many fuel pool drains.

Although none of the above mentioned methods of SFP cooling or makeup are strictly qualified, conditional use can be made of those methods, given the relatively long length of time available to institute those methods, and the low probability of ever needing them. In addition, the relatively poor quality of the ESW water for SFP cooling purposes provide added incentive for providing some other, more suitable, method of makeup. Therefore, because the use of the SFPC assist mode of RHR is available to preclude the occurrence of a SFP boiling event (and the resultant need for ESW makeup), and because various other methods of SFP cooling and makeup likely would be available, it was concluded that an adequate level of safety existed in the interim time period until full ESW makeup flow to the SFP is demonstrated.

Summary of Safety Evaluation

The fact that ESW was unable to deliver the amount of flow assumed in the UFSAR did not impact the ability of the ESW system to perform its other safety related function of supplying cooling water to support various other systems in transient or accident mitigation efforts. The loss of SFP cooling event that the ESW makeup flow would be required to mitigate is not considered an accident or transient, and the ability of ESW to provide the UFSAR assumed makeup flow did not influence the relative probability of occurrence of the event.

The function of providing SFP cooling is a safety action for planned operations (along with SFP shielding and reactivity control) and loss of SFP cooling is not considered an accident or transient that is evaluated in the SAR. Furthermore, the consequences of the event (loss of SFP cooling) are not increased since other reliable means (i.e., the SFP cooling assist mode of RHR) are available for providing SFPC such that boiling subsequent to the loss of the SFPC system would be reliably prevented prior to the need for mitigation via the ESW makeup line. In addition, the time between the postulated loss of the SFPC system and the onset of pool boiling (i.e., at least 24 hrs) is sufficient to reliably allow establishing either the SFP cooling assist mode of RHR or some alternate SFP cooling method, or to provide an alternate (and more desirable) means of providing makeup to replenish evaporative losses from the boiling SFP.

The consequences of a malfunction of the SFPC system are not increased because other reliable means of mitigation are available. Also, since the ESW to SFP makeup line does not interface with any systems, structures, or components that can act as initiators of any accidents or transients, the relative condition of the makeup line can not create the possibility of an accident or transient.

SE 95-07 <u>EDS Computer Contract</u>

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

The responsibilities for computer hardware and software use at the DAEC have been transferred to an outside contractor. That contractor, EDS, specializes in performing this service for other companies throughout the world. EDS already performs the same function for the non-nuclear portions of IES Industries. This change allows a company, whose specialty is computer resources, to manage the Division's computing functions so that IES Utilities Inc. does not need to maintain that specialized staff. This change should result in overall cost savings and gains in efficiency because of the reduced burden attracting and retaining a specially trained staff to perform those functions. EDS assumed the responsibility for supplying and maintaining all computer hardware and software to support the DAEC.

IES Utilities Inc. maintains oversight of EDS. EDS is obligated to comply with all procedures associated with the work they perform. All aspects of software control previously implemented at the DAEC are still utilized by EDS. The EDS organization in support of the DAEC reports through the IES Utilities Inc. management of the DAEC. In that way, EDS is also obligated to comply with IES Utilities Inc. Quality Assurance Program as it applies to any computer-related activities.

This represents a change to the organizational structure only. The process for modifying the plant is not altered, only the company making changes to software and hardware is changed. EDS, while designing and possibly implementing the changes, must work under controls established to support the DAEC Quality Assurance Program. All changes to plant software and hardware are reviewed in accordance with 10 CFR 50.59.

<u>Summary of Safety Evaluation</u>

This activity is an organizational change only and has no impact on any accident analyses, malfunctions of equipment, or margins of safety.

## Section C - Experiments

This section has been prepared in accordance with the requirements of 10 CFR Section 50.59(b). No experiments were conducted during the period beginning January 1, 1994 and ending October 1, 1995.

## Section D - 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 period beginning January 1, 1994 and ending October 1, 1995 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 |
| --- | --- |
| 29 | A reference to a DAEC Abnormal Operating Procedure was added for clarification. |
| 30 | Revisions were made to reflect DAEC organizational changes. |