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Cyclic Report of Facility Changes, Tests and Experiments, Fire Plan Changes and Commitment Changes
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In accordance with the requirements of 10 CFR Section 50.59(b), and NUREG-0737 (Item II.K.3.3), please find enclosed the subject report covering the period from October 1, 1998 through February 29, 2000. A summary of changes to the Duane Arnold Energy Center Fire Plan during the same time period is included, as well as a summary of commitment changes. There are no new commitments made in this letter.

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

Sincerely,

Kenneth E. Peveler

Kenneth E. Peveler
Manager, Regulatory Performance

JE47

Attachment: Cyclic Report of Facility Changes, Tests and Experiments, Fire Plan Changes and Commitment Changes

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April 27, 2000

Cyclic Report of Facility Changes, Tests and Experiments, Fire Plan Changes and Commitment Changes

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Section A - Plant Design Changes

This section contains brief descriptions of plant design changes completed during the period of October 1, 1998 through February 29, 2000, and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR Section 50.59(b). All changes were reviewed against 10 CFR 50.59 by the Duane Arnold Energy Center (DAEC) Operations Committee. None of the changes involved unreviewed safety questions.

The basis for inclusion of an Engineering Change Package (ECP) in this report is operational release of the associated modification at the DAEC during the period of October 1, 1998 through February 29, 2000. The basis for inclusion of an Engineered Maintenance Action (EMA) is completion of all the changes described in the Safety Evaluation, during the period of October 1, 1998 through February 29, 2000. Portions of some of the modifications listed were partially closed or partially operationally released in previous years.

SE 97-048 (Revision 1) ECP 1589 – Replacement of 'C' Well and Pump

Description and Basis of Change

The 'C' Well capacity was reduced due to degradation of the gravel pack formation around the inlet screen, as evidenced by testing. Capacity was limited to 400 GPM (Original design rating is 750GPM). Engineering determined that the 'C' well should not be considered as a long-term reliable means of production. It was considered likely that the twenty-year old well would no longer be functional after a few more years. This change abandoned the 'C' well, in accordance with current laws and drilled a new 'C' well approximately 25 feet northwest of the abandoned well. The old well was approximately 140 feet deep and drew water from the glacial drift above the Devonian/Silurian aquifer. The new well is approximately 350 feet deep and draws water from the Devonian/Silurian aquifer. The deeper well is a more reliable water source. This change also added two check valves to reduce water hammer effects and air in the Well Water System at pump startup. The existing 100 HP vertical turbine pump in the 'C' well was replaced by a 100 HP submersible pump, however, the flow and head capacity are essentially the same. The controls and logic remain the same.

Safety Evaluation Summary

Loss of well water is not an initiator of any of the design accidents in the plant. The moving of the well and replacement of the vertical turbine pump with a submersible pump does not affect any of the inputs considered in the accidents analyzed in the UFSAR (Updated Final Safety Analysis Report) or the NSOA (Nuclear Safety Operational Analysis). This change did not alter the interface between the well water system and the plant and cannot cause an accident or increase the likelihood of an accident. The probability of an 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 Well Water System. Therefore, this activity will not result in a condition which increases the probability of occurrence of an accident previously evaluated in the UFSAR. Well water is not relied on for recovery of an accident previously evaluated in the UFSAR. There will be no increase in the radiological consequences of any previously analyzed UFSAR accident. The changes made by this activity did not change, degrade or prevent actions described or assumed in an accident discussed in the UFSAR. This activity did not alter any assumptions previously made in evaluating the radiological consequences of an accident, nor does it play a role in mitigating the radiological consequences of an accident described in the UFSAR. The modifications to the system made by this activity had no impact on systems, structures or components important to safety. Replacement of the 'C' well with a comparable pump does not degrade the system's ability to perform its design function. The Well Water System performance will not be adversely impacted by this modification. Well Water is not relied on to mitigate an event. No physical or electrical separation criteria are affected by this alteration. The deeper well is a more reliable water source, therefore, it will not have an adverse effect on drywell cooling. The replacement pump is comparable in both pressure and flow and the well water system has no safety significance in the UFSAR. This alteration did not create an accident of a different type than any evaluated previously in the safety analysis report. This alteration restored the reliability of the Well Water System to its original design capacity but did not change the parameters of the system. There are no Technical Specifications associated with Well Water.

SE 97-061 High Pressure Coolant Injection (HPCI) Turbine Exhaust High Pressure Trip Setpoint Change

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

The HPCI turbine exhaust high pressure trip setpoint was changed by an EMA from a nominal value of 150 psig to a nominal value of 140 psig. No physical equipment changes were made. Only the trip setpoint of two pressure switches was changed. The new setpoint 140 psig ± 5 psig, is far enough above the maximum expected exhaust pressure (conservatively estimated to be 84.3 psig), and far enough below the minimum exhaust diaphragm burst pressure (159 psig) and piping design pressure (150 psig), that HPCI System performance is not adversely impacted by this change. In fact, since the setpoint was moved further from the minimum exhaust diaphragm burst pressure, yet still has adequate margin above the maximum expected exhaust pressures, overall HPCI System reliability should be improved.

Safety Evaluation Summary

This change did not involve a change in the design, material or construction standards of the HPCI System. Therefore, this change did not increase the likelihood of an accident occurring. This change did not affect the HPCI initiation or control logic, or the integrity of any piping connected to the reactor vessel. Therefore, the probability of an inadvertent HPCI initiation or of a Loss of Coolant Accident (LOCA) was not increased. The effectiveness of the HPCI System and primary containment in mitigating the consequences of accidents evaluated in the SAR was not reduced, and the consequences of such accidents were not increased. This lower setpoint has been evaluated and determined to not increase the probability of spurious HPCI turbine trips, while actually offering increased protection of the turbine exhaust piping. This change increases the availability of the HPCI System, since, if a high exhaust pressure condition should occur, the likelihood of the turbine tripping before the rupture diaphragms burst will be increased, thus allowing the System to more likely be quickly returned to service if the cause of the high turbine exhaust pressure is identified and corrected. Therefore, the probability of a serious malfunction of the HPCI System will be reduced. The probable failure modes of the HPCI System and the effects of these failures, will not be changed. Since the probability of the HPCI turbine exhaust rupture diaphragms bursting has been reduced, the probability of releasing radioactive steam into the HPCI room (secondary containment) if a malfunction in the turbine exhaust line should occur, will be reduced. Therefore, the consequences of a malfunction of the HPCI System have not been increased. No new challenges to the existing barriers (fuel cladding, reactor coolant pressure boundary or primary containment) were created. This change did not change the design basis of the HPCI System in any way. Therefore, this change did not create the possibility of any new types of accidents. The probability of an inadvertent HPCI System trip due to high turbine exhaust pressure conditions is being maintained acceptably small. This change did not create the possibility of a different type of malfunction of the HPCI System than previously evaluated. Since this change affected a turbine protective function and not the system actuation function, no margin of safety defined in the basis for any Technical Specification was reduced.

SE 97-067 Installation Of Ladder Rack In 'A' Emergency Diesel Generator Room

Description and Basis of Change

The purpose of this EMA was to install a ladder rack in the 'A' Emergency Diesel Generator Room. A ladder is required to isolate local air receiver tanks. The installed rack provides storage for one twelve-foot stepladder, which is suitable for this purpose. The scope of this activity included welding angle iron brackets on to two columns located on the east wall of the room. These brackets were welded directly to the structure, eliminating the need for a base plate. The ladder rack brackets were shortened such that they will only accept one ladder. The shorter moment arm resulting from this change added conservatism to the design.

Safety Evaluation Summary

The addition of brackets to secure ladders to the Turbine Building columns had no effect on the accident evaluation contained in the UFSAR. There were no changes in any of the accident consequences as a result of this activity. The evaluation demonstrates the adequacy of the attachment for all applicable loading conditions. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This addition has no effect on any equipment important to safety. The additional loading placed on the building as a result of attaching ladders to it is inconsequential. The evaluation included with this EMA and calculation demonstrates the racks are adequate to secure ladders during a seismic event, providing assurance the ladder will not fall on sensitive equipment. This change did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. This change did not have any effect on the operation of the associated equipment. The installation of ladder racks allows a ladder to be stored in the room, which will facilitate the performance of the Standby Diesel Generator Operability Testing. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification. This activity had no effect on the System's operation, setpoints, capacity, or any of the operating modes described in the Operating License and Technical Specifications.

SE 97-068 Installation Of Ladder Rack In 'B' Emergency Diesel Generator Room

Description and Basis of Change

The purpose of this EMA was to install a ladder rack in the 'B' Emergency Diesel Generator Room. The practical configuration for ladder storage is for the ladder to hang horizontally from two ladder rack brackets attached to the east wall of the room. A ladder is required to isolate local air receiver tanks. The rack installed under this EMA provides storage for one twelve-foot stepladder, which is suitable for this purpose, and one six foot step ladder for general maintenance. The ladder rack was fastened to the wall with ½ inch Hilti Drop-In anchors and ½ inch bolts. The rack was placed on approximate four foot centers to accept a six foot as well as a twelve foot step ladder. The rack was fabricated and installed using approved plant procedures.

Safety Evaluation Summary

The addition of brackets to secure ladders to the Turbine Building interior wall had no effect on the accident evaluation contained in the UFSAR. There were no changes in any of the accident consequences as a result of this activity. The evaluation demonstrates the adequacy of the attachment for all applicable loading conditions. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This addition had no effect on any equipment important to safety. The additional loading placed on the building as a result of attaching ladders to it is inconsequential. The evaluation included with this EMA and calculation demonstrates the racks are adequate to secure ladders during a seismic event, providing assurance the ladders will not fall on sensitive equipment. This change did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. This change did not have any effect on the operation of the associated equipment. The installation of a ladder rack allows ladders to be stored in the room, which will facilitate the performance of the Standby Diesel Generator Operability Testing as well as general maintenance. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification. This activity had no effect on the System's operation, setpoints, capacity, or any of the operating modes described in the Operating License and Technical Specifications.

SE 97-085 (Revision 1) ECP 1592 - Level 1 Security Modifications

Description and Basis of Change

This modification allows the Security Department to operate in a more efficient manner while still being able to adequately protect the vital areas of the plant. Due to the sensitive nature of security and the need to maintain certain aspects of these modifications confidential, a detailed explanation as to the purpose of each modification is not discussed. Detailed security measures for the physical protection of Nuclear Power Plants are required by 10 CFR 50.34(c) and applicable sections of 10 CFR 73. For a more detailed explanation as to the purpose of this modification, refer to the DAEC Security Plan.

The following changes were made:

- Installed cages with chain link sides and fold away scissors gates on both sides inside and out of the south turbine roll-up door and the north turbine roll-up door.
- Closed off the south turbine walk out door and removed exit signs.
- Installed a man-door with card reader west of the north turbine roll-up with a chain link cage on the inside and flood block guides on the outside.
- Installed magnetic locks on the outside doors of the north administrative corridor vestibule.
- Installed a partition and roll-up gate in the first floor administration building hallway.
- Installed a hinged steel plate in the administration building hallway.
- Installed an activation switch in the Secondary Alarm Station.
- Installed override switches that will lock down both the reactor building railroad airlock doors and the radwaste - reactor building airlock doors.
- Installed a cage made of deck grating with a door around the exterior of the recombiner door and removed exit signs.
- Installed a steel plate outside of the secondary alarm station.
- Installed heaters with thermostats on the radwaste gate and the main gate active vehicle barriers.
- Alarmed doors 224, 239, 239A, 242, 260, 285, and 420.

Safety Evaluation Summary

This modification enhanced the security system by installing additional barriers and locks, none of which are initiators of any of the design accidents in the plant, nor do they affect any of the inputs considered in the accidents analyzed in the UFSAR or the NSOA. The proposed change does not decrease the effectiveness of any of the safety systems in the plant and cannot cause an accident or increase the likelihood of an accident. The probability of an 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 Security System. Therefore, this activity will not result in a condition which increases the probability of occurrence of an accident previously evaluated in the UFSAR. Security barriers are not relied upon for recovery of an accident previously evaluated in the UFSAR. There will be no increase in the radiological consequences of any previously analyzed UFSAR accident. The changes made by this activity will not change, degrade or prevent actions described or assumed in an accident discussed in the UFSAR. This activity will not alter any assumptions previously made in evaluating the radiological consequences of an accident, nor will it play a role in mitigating the radiological consequences of an accident described in the UFSAR. The modifications to the systems made by this activity have no impact on systems, structures or components important to safety, nor do they degrade the systems abilities to perform their design function. The secondary containment airlock function will not change and the turbine building will maintain its original design function. The Security System has no safety significance in the UFSAR. The secondary containment will not be adversely impacted by this modification. The turbine building will retain its original design functions. The Security System is not an initiating event nor is it relied on to mitigate an event. No physical or electrical separation criteria are affected by this alteration. This change did not create an accident of a different type than any evaluated previously in the SAR. There are no Technical Specifications associated with the Security System.

SE 97-104 Installation Of Crack Arrest Verification

Description and Basis of Change

This EMA converted a temporary modification to a permanent modification. ½ inch Crack Arrest Verification (CAV) tubing was NMCA (Noble Metal Chemical Addition) treated during the plant NMCA process. These NMCA tubes were added to the CAV sample so that the NMCA treated surface would be exposed to Recirculation water. Several coupons are removed periodically to assess the durability of the NMCA process. These coupons are the primary means of assessing the durability of NMCA. The CAV System remains configured as two separate sample loops, CAV-A and CAV-B with each loop capable of monitoring on-line stress corrosion cracking (SCC) growth rates and electrochemical corrosion potential (ECP) in the primary reactor coolant sample taken from the B-loop of the Recirculation System. CAV-A continues to monitor SCC and ECP of Recirculation water. CAV-B continues to sample reactor water but is occasionally augmented with oxygenated water to control the ECP. Extended monitoring capability was added to the CAV-B loop to evaluate feasibility and durability of the NMCA process. Since the plant was treated with NMCA in Refuel Outage 14, the equipment has been used to evaluate the durability of plant treated NMCA coupons throughout Fuel Cycle 15 and 16. Such information is crucial to determine the long-term effectiveness of the DAEC NMCA process.

This EMA converted the following CAV-B temporary equipment to permanent plant equipment:

- An oxygen injection skid was permanently connected to the plant Demineralized Water supply. The Demineralized Water is circulated within the skid while being sparged with oxygen gas. The resulting oxygenated water is then injected into the CAV-B sample loop. This process allows direct control of the hydrogen and oxygen molar ratio in the CAV-B sample which in turn controls the ECP of NMCA treated and untreated stainless steel coupons in the CAV-B loop. Oxygen concentration in the CAV-B loop is controlled by a combination of CAV-B sample flow rate and oxygenated water injection rate.

- Chemistry Sample Station and Chemistry Monitoring Station - During periods of oxygen injection, separate chemistry monitoring equipment is required to monitor hydrogen, oxygen and conductivity of the CAV-B loop. The sample station is the "wet" part of the monitoring package which contains the sensors and is piped to the CAV-B sample point. The system is configured so that either CAV-B inlet or outlet sample can be monitored. The CAV-B sample is rough cooled with Reactor Building Closed Cooling Water (RBCCW) and finish cooled with an electric constant temperature bath. The dual function sampler also acts as a calibrator for the hydrogen and oxygen sensors. Calibration gases are sparged through a column of Demineralized water in order to calibrate and QC check the sensors. Two levels of oxygen and hydrogen concentrations, the balance being nitrogen, are needed along with pure nitrogen to calibrate the sensors. These five gas bottles are located near the chemistry sample/calibrator station. The chemistry monitoring station is the "dry" component which contains the oxygen, hydrogen and conductivity analyzers. The resulting signals are input into the CAV data acquisition system.

- NMCA coupons were installed at the beginning of Fuel Cycle 15. These coupons were NMCA treated along with the plant during Refuel

Outage 14. Some of the coupons were exposed to reactor water for several years as part of the CAV system prior to being treated with NMCA. Several coupons are removed every three months and analyzed for residual noble metal still present on the surface.

- ECP High Flow Module - Several of the NMCA coupons have been fabricated into an ECP assembly. Each specimen is electrically isolated and functions as an ECP working electrode. As residual noble metal decreases, the ECP of the corresponding surface can be monitored in the ECP module. The reference ECP electrode for this module is cooled with RBCCW.

- The CAV-B sample is also routed to the plant installed on-line ion chromatograph.

Safety Evaluation Summary

Connecting the plant Demineralized Water System to the CAV system sample line, via high pressure injection pumps, and adding additional monitoring equipment has no safety significance or impact on any accidents postulated in the SAR. The permanent addition of this equipment did not affect any of the inputs considered in the accidents analyzed in the UFSAR or NSOA. This modification did not alter the interface between the CAV System and the plant, and it did not cause an accident or increase the likelihood of an accident. The probability of occurrence of the accidents discussed in the SAR and NSOA is based on initial conditions and assumptions that do not depend on the end use of or any interactions with the CAV System. Therefore, this activity did not result in a condition that increased the probability of occurrence of an accident previously evaluated in the SAR. This activity did not increase the consequences of an accident evaluated previously in the SAR. There was no increase in the radiological consequences of any previously analyzed accident. The changes made by this activity did not change, degrade or prevent actions described or assumed in an accident discussed in the SAR. This activity did not alter any assumptions previously made in evaluating the radiological consequences of an accident, nor did it play a role in mitigating the radiological consequences of an accident described in the SAR. All the equipment made permanent is located downstream of the Recirculation sample valves, which act as primary containment isolation. This activity did not impact 10 CFR 100 limits. This activity did not increase the possibility of associated equipment malfunction that was not previously evaluated in the SAR. The design basis was not changed. This change did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This activity did not alter any assumptions previously made in evaluating the consequences of an equipment malfunction. All activities were non-safety related. The CAV System and its associated equipment are installed at a safe distance from the safety related structures, systems, and components. The conduit is designed and installed as seismic II over I in the vicinity of the safety related instrument lines. There are no other seismic concerns. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No new failure modes were created. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification. The margin of safety was not affected by the installation of the CAV System. The CAV System is not mentioned in the Operating License or Technical Specifications.

SE 97-105 Copper-trol Injection To The Well Water System

Description and Basis of Change

This EMA installed a system to inject a Copper Corrosion inhibitor (BetzDearborn-Copper-trol) into the Plant Well Water System to ensure adequate protection for the copper alloys in the various coolers and heat exchangers serviced by the Well Water System. In particular, the Drywell System coolers are installed with copper tubing. The addition of BetzDearborn Copper-trol should lower the corrosion rate, lower the chance of attack on mild steel, reduce copper in the Well Water System outfall/discharge, and reduce the probability that the coolers will have to be replaced before the end of the projected plant life in the year 2014. Copper-trol is shot-fed for a short period, and the resulting film is tenacious enough to remain for 3-4 weeks. Injection of Copper-trol is controlled via a Plant Chemistry Procedure. The non-safety related Copper-trol skid is mounted a significant distance from safety related equipment onto the seismically qualified Reactor Building second floor (in close proximity to the sodium hexametaphosphate skid). If the mounting were to fail during a seismic event or other failure mechanism were to occur, safety-related equipment operating on the same floor will not be affected due to their remote locations in reference to the skid. In reference to corrosion failures, all connections involved with this modification are stainless steel to stainless steel, minimizing and/or eliminating galvanic corrosion concerns. The Copper-trol injection skid operates within the pressure design limits of the Well Water piping.

Safety Evaluation Summary

The probability of an 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 Well Water System.

Therefore, this activity will not result in a condition that increases the probability of occurrence of an accident previously evaluated in the SAR. The addition of Copper-trol will have no impact on Drywell cooler efficiency or effectiveness, since the Copper-trol layer is not thick enough to impede heat transfer. Copper-trol is compatible with the materials in the primary containment. The changes made by this activity will not change, degrade or prevent actions described or assumed in an accident discussed in the UFSAR. There will be no increase in the radiological consequences of any previously analyzed accident. This change will not alter any assumptions previously made in evaluating the radiological consequences of an accident, nor will it play a role in mitigating the radiological consequences of an accident described in the SAR. Therefore, this modification will not increase the consequenses of an accident evaluated previously in the SAR. This change does not adversely affect equipment that has a safety function. This modification has no adverse impact on systems, structures, or components important to safety. This activity will increase the reliability of the Well Water System coolers due to reduced corrosion rate. The Well Water System has no safety significance in the UFSAR. The UFSAR has evaluated the potential for equipment damage due to a cooling water leak in containment. The existence of Copper-trol in the water does not challenge the bounds of this evaluation. Therefore, the proposed activity will not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. There are no credible scenarios where the failure of this system in a non-conservative direction could increase the consequences of a malfunction of equipment important to safety evaluated previously in the UFSAR. This activity will not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This modification will not create an accident of a different type than any evaluated previously in the SAR and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. There are no Technical Specifications associated with the Well Water System. The margin of safety as defined in the basis for any Technical Specification is not reduced.

SE 97-113 Reactor Building Equipment Drain Sump Flush Lines

Description and Basis of Change

This EMA installed flush connections on the piping to and from the Reactor Building Equipment Drain Sump Heat Exchanger and on an isolation valve on the drain line from this unit. These connections aid in the decontamination of this unit by allowing this unit to be chemically cleaned and/or flushed with clean condensate water without damaging or taxing the existing Radwaste equipment. This change added a branch connection with an isolation valve on the inlet line to the heat exchanger, a branch connection with an isolation valve on the outlet line from this unit and an isolation valve after this connection to prevent unwanted flush water from entering the drain sump. This modification also disconnected a portion of the equipment drain line on the Reactor Core Isolation Cooling (RCIC) System skid from piping that flows into closed radwaste and reconnected it to an open radwaste connection located in close proximity. The relocation of this portion of drain line eliminates the effect of back pressure caused by water draining into the system from higher elevations in the reactor building. With the back pressure concerns eliminated the funnel/vent on the closed radwaste line located in the Torus basement can be replaced with closed piping.

Safety Evaluation Summary

The addition of flush connections with isolation valves to the Reactor Building Equipment Drain Sump Heat Exchanger inlet and outlet piping will not increase the probability of an accident evaluated in the SAR. The Reactor Building Equipment Drain Sump System is considered part of the Radwaste System. Relocation of the RCIC drain lines within the Radwaste System will not adversely affect the system operation. All of the system components affected by this change are located outside of primary containment but are inside the concrete reactor building such that any possible release will not contribute to any accident evaluated in the SAR. The addition of valves and connections using items that meet or exceed plant design requirements will not have any effect on any of the evaluated accidents. Relocation of the RCIC drain lines and the notation of a more conservative pipe class will not affect any accident analysis. There will be no increase in dose consequences as a result of these activities. The Liquid Radwaste System does not contribute to safe Reactor shutdown. The heat exchanger feed and return lines as well as the RCIC drain lines are seismic class II and have no nuclear seismic requirements. The components used meet the system design requirements and are installed and tested using approved plant procedures. This activity did not challenge any of the installed equipment. This change will not create the possibility of an accident of any type. The valves used are consistent with plant design and will provide isolation for the flush connections and serve as system pressure boundary. The heat exchanger is normally isolated and not required for safe shutdown. The piping used for the drain line modification meets plant design. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. Based on a review of the SAR and the Technical Specifications, no margin of safety is defined which would be affected by this activity. Therefore, this change will not reduce the margin of safety.

SE 98-014 Removal Of Personal Monitoring Stations And Curb From Turbine Operating Deck

Description and Basis of Change

This EMA removed the personal monitoring stations on the Turbine Operating Deck on the north and south ends of the turbine building to free floor space. This activity removed cinder block structures, including electrical outlets and light, from the Turbine Operating Deck. The curb on the northeast corner of the Turbine Operating Deck was also removed to remove the floor obstruction from this area.

Safety Evaluation Summary

The personal monitoring stations are non-safety related and house no safety related items. These stations, as well as the curb used to contain runoff from the no longer used turbine wash, are not essential to plant operation. The removal of these structures does not increase the probability of an accident, or reduce any safety margins below the levels specified in the UFSAR. The structures do not increase the consequences of an accident, or reduce any safety margins below the levels specified in the UFSAR. The shielding provided by the monitoring stations is to reduce background radiation while using a hand held radiation monitor and has no effect on any dose consequences. Interface with plant systems is structural and removal of the stations did not affect any safety related electric, air or water services. Removing the monitoring stations and floor curb does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the UFSAR, and the consequences of a malfunction of equipment important to safety evaluated previously is the SAR is not increased. The possibility of an accident of any type that has not been evaluated previously in the SAR was not created. The electric loads associated with the removed outlets and light are not listed or considered in the Power System Analysis. This activity will not create the possibility of a malfunction of equipment important to safety different than those evaluated previously in the UFSAR. The NSOA does not address buildings. A review of the SAR concluded no margins of safety are affected, therefore, none can be reduced below the levels specified in Technical Specifications.

SE 98-060 Containment Atmosphere Control System Modification

Description and Basis of Change

It was determined the installed configuration for the Nitrogen Dryer Outlet Filter was not properly depicted on the Containment Atmosphere Control System Piping and Instrument Diagram (P&ID). The unit does not have an automatic drain, such that manual manipulation of the bowl drain was required for moisture removal. The drain valve line up was changed to effectively abandon in place a drain trap and Y-strainer so that the drain line from the filter/moisture separator does not become pressurized. This EMA installed an automatic drain in the Nitrogen Dryer Outlet Filter filter bowl, opened the drain trap bypass valve, and closed the drain trap inlet valve. This valve manipulation bypassed and abandoned the drain trap and inlet strainer. The P&ID was also revised and updated to indicate the filter/moisture separator is on the outlet of the refrigeration dryer, the drain trap bypass valve is normally open, and the drain trap inlet valve is normally closed. The Operating Instuction (OI) were also revised in conjunction with this EMA to show this valve line-up.

Safety Evaluation Summary

The addition of an automatic drain to this filter/moisture separator allows the refrigerator dryer to function as designed without manual intervention. The elimination of the strainer and drain trap from the filter/moisture separator drain line to closed radwaste assures this drain line will not become pressurized and rupture the tygon tubing used to make this connection. This change to the drain configuration will not increase the probability of an accident evaluated in the SAR. This modification did not impact operation of the Drywell Pneumatic Nitrogen Supply Compressor. This activity did not increase the consequences of an accident evaluated previously in the SAR. The change to the drain configuration will have no affect on the cleanliness of nitrogen delivered to this system. The drain to closed radwaste is the normal flow path for the removed moisture and this change will have no additional radiological consequences. This modification will assure moisture is removed from the nitrogen pneumatic supply. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The equipment affected by this change is not considered important to safety and there is no adverse impact on the system operation. The addition of an automatic drain to the filter/moisture separator may improve the nitrogen quality by assuring moisture is removed as a source of contamination. This drain line is not identified with a line designation and has no nuclear seismic requirements. There was no change in the drain line components or mounting. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This change did not affect the equipment function or flow path. The automatic drain trap serves the same purpose as the drain trap, and the prefilters on the compressor unit will suffice for the y-strainer eliminated by this valve lineup. This change did not create the possibility of an accident of any type, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The margin of safety as defined in the basis for any Technical Specification was not reduced. This activity had no effect on system operation, set points, capacity, or any of the operating modes described in the Operating License and Technical Specifications.

SE 98-061 Change Overcurrent Settings On Low Level Radwaste (LLRW) Load Center Breaker 1BR725

Description and Basis of Change

This EMA activity adjusted the overcurrent settings of LLRW load center breaker 1BR725 to offer the maximum protection from nuisance tripping due to down stream circuit breakers not responding quick enough, resulting in faults migrating to the load center breaker. Breaker 1BR725 receives its power from LLRW bus 1BR7. Bus 1BR7 supplies several non-safety related loads in LLRW, including room and area heaters, fans, pumps and motor control centers. All loads supplied from 1BR7 are balance of plant system and are not safety related. However, breaker 1BR725 supplies Radwaste Processing Motor Control Center (MCC) 1B63. The Radwaste Processing System ensures that wastes are processed in a timely manner. Breaker 1BR725 has the capability to be adjusted to a slightly longer fault response time, therefore, allowing the downstream molded case breakers the needed time to provide fault tripping for the individual branch circuit. The short time delay setting, was adjusted from 0.18 seconds to 0.33 seconds, which allows ample time for the slower responding branch circuit breakers to trip and isolate the fault. This adjustment does not affect electrical coordination with upstream breakers and will not exceed amperage interrupt ratings of the LLRW system electrical distribution system.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. MCC 1B63 is the only load supplied by load center breaker 1BR725. MCC 1B63 has no safety related loads. The Radwaste System was designed to process wastes in a timely manner, ensuring that reactor operations and maintenance activities do not impose restrictive impairment upon the operation of the plant. This activity offers better electrical system coordination and enhances the performance of the Radwaste System. This activity did not increase the consequences of an accident evaluated previously in the SAR. This activity did not change the capability of the Radwaste System to maintain the radioactivity concentrations in the discharge system as low as reasonably achievable and well within the guideline limits of 10 CFR 20. The Radwaste System is sized to collect and process the radwaste generated from the reactor under normal power operation and expected operational occurrences. This activity did not change the design features or inhibit the ability of the Radwaste System to perform its intended functions. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased. This change minimizes unplanned equipment unavailability within the Radwaste System. Proper breaker coordination provides better reliability of the Radwaste System. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. This activity did not cause or create a failure mode which has not been previously evaluated in the UFSAR. No new type of failure mode or additional system malfunction was introduced by this activity. A review of the SAR indicated there is no margin of safety involved with this activity, therefore, there is no reduction in margin of safety as defined in the basis for any Technical Specification.

SE 98-062 Change Overcurrent Settings Of LLRW Load Center Breaker 1BR727

Description and Basis of Change

This EMA activity adjusted the overcurrent settings of LLRW load center breaker 1BR727 to the values required within the National Electrical Code, NFPA section 70, to offer practical safeguarding of this Electrical System and for protection from overcurrent within the limitations of the electrical components being supplied by this load center breaker. Breaker 1BR727 receives its power from LLRW bus 1BR7. Bus 1BR7 supplies several non-safety related loads in LLRW, including room and area heaters, fans, pumps and motor control centers. All loads supplied from 1BR727 are balance of plant systems and are not for safe shutdown of the plant. Breaker 1B727 is the primary source of power for security loads. A transfer switching scheme is used to provide alternate power to security should normal power be lost from 1B727 or LLRW main bus 1BR7. This activity did not change the capacity of breaker 1BR727 to provide power to security load as designed. The operation and design of the security transfer scheme was not affected or changed in anyway. This work activity provides electrical system coordination for the non-essential LLRW electrical distribution system and proper overcurrent protection for the security electrical system. Loss of breaker 1BR727, primary source of power to the security building will not affect plant operation or any system required for the safe shutdown of the plant.

The frame size of the 1BR727 load center breaker was listed incorrectly as 400 amperes in an UFSAR figure. The frame size or rating was revised to 800 amperes, the actual rating of this breaker.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The breaker is not used for any safety related loads or balance of plant loads, other than the security building. Security is not an initiator of any accident evaluated in the SAR. Loads supplied by this breaker are not required for safe shut down of the plant. This activity did not increase the consequences of an accident evaluated previously in the SAR. The capability of the Security Electrical Distribution System was not changed. This modification did not change the design of the Security System. The probability of a malfunction has been reduced somewhat, and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. Proper breaker coordination provides better reliability of the Security System and the LLRW supply bus. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the probability of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. No new type of failure mode or additional system malfunction was introduced by this modification. The function and design of the Electrical Distribution System remain the same and are not changed. No margin of safety was involved, therefore, this change did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-064 Removal Of Chemical Waste Filter And Detergent Drain Filter High Differential Pressure Annunciators

Description and Basis of Change

The Chemical Waste Filter and Detergent Drain Filter are no longer used due to personnel exposure. Replacement of the filter element exposed personnel to higher doses of radiation. The filter elements were previously removed from the Chemical Waste Filter and the Detergent Drain Filter. The use of the Floor Drain Demineralizer to process the Chemical Waste Tank and Detergent Drain Tank contents allows processing from the

Radwaste Control Room. If the contents of these tanks are beyond the capability of the Floor Demineralizer, the waste is solidified and processed as solid waste. Both of these functions can be performed from the Radwaste Control Room. This reduces personnel exposure and radioactive waste. This EMA disabled the annunciators for the high differential pressure alarms for both the Chemical Waste Filter and the Detergent Drain Filter since water circulated through the filter housings caused spurious alarms to be received. The differential pressure transmitter for both filters were isolated at the root isolation valves and the instruments in the loop de-energized. The annunciator card was then removed and the annunciator window was blanked. An UFSAR drawing has also been updated.

Safety Evaluation Summary

The changes described above do not affect the inputs into any accident analysis performed for DAEC. Therefore, the removal of the high differential pressure annunciators for the Chemical Waste Filter and Detergent Drain Filter did not increase the probability of occurrence of any accident evaluated previously in the SAR, and the consequences of any accident evaluated previously in the SAR were not increased. The affected equipment is not classified as nuclear safety related. This change did not create any situation where equipment important to safety would be compromised, and any regulatory commitments regarding malfunction of equipment were not affected. The Radwaste System is classified by DAEC as Quality Level 2, which invokes quality assurance requirements for the pressure boundary components of the Radwaste System only. This change does not affect the pressure boundary of the Radwaste System. Therefore, the probability of occurrence of a malfunction of any equipment that has an important safety function is not affected. This change does not result in increased radiological exposure to plant personnel or the public. Therefore, this change does not increase the consequences of a malfunction of any equipment important to safety evaluated previously in the SAR. The operating procedures for chemical waste and detergent waste is to process them via the Floor Drain System or solidify and dispose of the waste as solid waste. Since the UFSAR already addresses the current operating practices of not using the Detergent Drain Filter or the Chemical Waste Filter, the removal of the high differential pressure annuciators for these filters does not create the possibility of an accident of a different type than any evaluated previously in the SAR. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The bases for the Technical Specifications associated with the Radwaste System are not affected by this change. The margin of safety is not affected by this change.

SE 98-066 Changes To Control Building Computer Room Air Conditioning Unit Condensing Unit

Description and Basis of Change

The purpose of this EMA was to improve the reliability of the Plant Air Conditioning System. The Plant Air Conditioning System is designed to control the plant air temperatures and the flow of airborne radioactive contaminants to ensure the operability of plant equipment and the accessibility and habitability of plant buildings and compartments. In order to better meet this design basis the Control Building Computer Room Air Conditioning Unit Condensing Unit required a hot gas bypass valve, that would unload the compressor on low load, and a low pressure bypass trip with a time delay. The unit operated with the hot gas bypass valve open continuously. This short cycled the refrigeration cycle and reduced the efficiency of the compressor lubrication system. During winter operation the refrigerant temperature was very low with the unit secured. During startup the unit frequently tripped on low suction pressure as a result of the low ambient temperature. A new two stage temperature control system was installed to cycle the compressor and hot gas bypass independently. Also, a time delay was installed to allow the refrigerant to warm (refrigerant pressure increase) prior to startup to avoid tripping the unit during low ambient temperature conditions.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The affected equipment is not required to mitigate any accident or transient evaluated in the SAR. The overall function of the Computer Room HVAC System before and after the modification remains the same. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. The computer room air conditioning (AC) unit is quality level (QL) 4 and non-seismic. Loss of this unit will not affect operation of any equipment required to be operable in any event or transient discussed in the SAR. The failure modes of the new switch and time delay, though highly unlikely, are limited to loss of the cooling unit only, and loss of this unit and the components cooled by this unit does not affect safe operation of the plant. The changes made by this activity are limited to the control circuit of the cooling unit, and do not affect any other Structure, System, or Component (SSC) of the plant. No equipment important to safety is adversely affected by this change. This modification will not increase the probability of occurrence of a malfunction of equipment important to safety evaluated in the SAR. For accident conditions the AC unit is not required for safety. The computer equipment cooled by the unit is QL 4 with the exception of the Safety Parameter Display System (SPDS) equipment which is QL2. None of this equipment is required to be operable in the event of an accident. This activity does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. No failure to cause an accident of different type than previously evaluated can be postulated by this modification. The operation or loss of this equipment does not affect habitability and radiation control in the control room. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The overall system is made more reliable by eliminating unnecessary compressor trips. This activity did not reduce margin of safety as defined in the basis of any Technical Specification Section.

SE 98-067 Cable Spreading Room Temperature Controller

Description and Basis of Change

This EMA restored automatic temperature control to the cable spreading room. The Plant Heating, Ventilating and Air Conditioning System is designed to control the plant air temperatures and the flow of airborne radioactive contaminants to ensure the operability of plant equipment and the accessibility and habitability of plant buildings and compartments. In order to better meet this design basis, the cable spreading room temperature controller was replaced and its action changed to reverse acting. The direct acting temperature controller did not control temperature in automatic mode, it did not calibrate, and the model of the controller was obsolete.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. This activity modified a temperature controller which is quality level 4 and is not required to safely shutdown the plant. The consequences of an accident evaluated previously in the SAR were not increased. The affected equipment is not required to mitigate any accident or transient evaluated in the SAR. This modification allows the temperature to be automatically controlled, as designed. The overall function of the cable spreading room HVAC system before and after the modification remains the same. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The failure modes of the controller are to fail to cool or to overcool the cable spreading room. Neither of these failures adversely affects any equipment whose malfunction is evaluated in the SAR. No equipment important to safety is adversely affected by this change. Since the changes made by this activity meet the original system bases, electrical separation, physical separation, Appendix R, seismic and environmental requirements, this modification did not increase the probability of a malfunction of equipment important to safety evaluated in the SAR. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. The operation or loss of this equipment does not affect habitability and radiation control in the control room. Loss of this unit will not affect operation of any equipment required to be operable in any event or transient discussed in the SAR. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The overall system is made more reliable by allowing for automatic operation. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-077 Removal Of Filter Elements From Radwaste Tanks' Vent Lines

Description and Basis of Change

Various EMAs removed filters and housings, as needed, from vent lines of the Condensate Phase Separator Tanks, the Chemical Waste Tank, the Chemical Waste Sample Tank, and the Radwaste Collector Tank. These filters were ineffective due to deterioration. The filter elements trapped and held moisture, causing the filters to become ineffective and the trapped moisture caused excessive oxidation and corrosion of the filter housing.

Safety Evaluation Summary

The vent lines from these tanks are connected into the Reactor Building main air exhaust duct, exhausting through the building exhaust stacks. The removal of these filters did not alter this vent path nor interfere with the automatic actions initiated from the radiation monitors in this system. The filters and the associated housing and duct work are not considered important to safety. These filters are not an initiator of any analyzed accident. The removal of these filters had no effect on the radiation protection features of the Reactor Building Ventilation System's control of radiation release to the environs. The vent path meets General Electric's Design Recommendations. This activity did not increase the consequences of an accident evaluated previously in the SAR. These filters do not have any safety function to mitigate the consequences of an accident. The removal of these filters did not affect the operation of the Reactor Building Ventilation System. The likelihood of an unacceptable Radiological Release is not changed by this modification. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR, and this activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. Radiological surveys performed on the interior of the filter housings and the removed filters during maintenance activities found low levels of radiation present. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The path through Standby Gas Treatment in the event of the detection of high radiation levels was not affected. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. These changes did not impact any safety related equipment and did not introduce any new plausible equipment failure modes. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification. The setpoints and the operation of the Secondary Containment Isolation Dampers were unchanged by this modification.

SE 98-079 Removal Of Offgas Dilution Dampers' Flow Control Equipment

Description and Basis of Change

An EMA removed the Offgas dilution dampers' flow control equipment. The purpose of this equipment was to automatically control the Offgas dilution flow. This automatic function has not been used since 1993. The air supply used to perform this function was capped off in 1997. The dampers are failed open, which is the most conservative position. The failure of these components, could have caused a failure of equipment required to be in service in the Offgas system. All other controls and operation of the Offgas System were not affected.

Safety Evaluation Summary

This activity meets the design, material, and construction standards of the Offgas System. This modification did not change the flow through the Offgas or Standby Gas Treatment (SBGT) Systems. The probability of occurrence of an accident evaluated previously in the SAR was not increased because the dampers are in the failed open position. During an auto start of SBGT these dampers should be open. The safety function of the Offgas Dilution Flow Loop is to ensure there is Offgas dilution flow when the SBGT System is in operation. The controls for the dilution fans in the Offgas System remain the same. This modification ensures there is Offgas dilution flow when the SBGT System is in operation by failing the flow dampers in the open position. There is no control function for these dampers. Therefore, removal of the control equipment for the dampers reduces the likelihood of having a control circuitry failure, which may cause a failure of the fully open dampers. This activity did not increase the consequences of an accident evaluated previously in the SAR. This activity decreased the probability of occurrence of a malfunction of equipment important to safety. This was accomplished by reducing the number of components that could fail, which could affect equipment required to perform a safety function. The components important to safety were not changed. This activity did not increase the consequences of a malfunction of equipment important to safety, and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. This modification did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. Any margins of safety are maintained.

SE 98-082 Abandon In Place General Service Water (GSW) System Supply To Auxiliary Heating System Drain Attemporator

Description and Basis of Change

This activity abandoned in place the General Service Water (GSW) System cooling supply to the Auxiliary Heating System Drain Attemporator. The drain attemporator receives exhaust steam from the shell side relief valve of the House Heating Heat Exchanger. The GSW cooling supply mixed with the relief valve discharge to quench the plume as it traveled to the floor drain. A temperature control valve controls the GSW supply to the attemporator. GSW cooling supply was provided only when relief valve discharge to the attemporator was detected. The GSW supply line sat in a stagnant condition for long periods of time. The corrosion buildup over time completely blocked cooling supply. Cleaning of the supply line provided temporary correction. The attemporator location, extremely infrequent use, lack of impact on relief valve operation, and lack of personnel and equipment (both safety and non-safety related) impact justified abandoning the cooling water supply to the attemporator. The GSW System P&ID and the Auxiliary Heating System P&ID required markup to indicate the GSW cooling supply line abandonment.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. All of the accidents previously evaluated in the SAR were reviewed with respect to this activity. None of the accidents previously evaluated were affected by this activity. The operation of the House Heating Heat Exchanger Relief Valve was not impacted in any way. This activity had no impact on any safety related system. The GSW and Auxiliary Heating Systems are not initiators of any accident evaluated in the SAR, and they are not relied upon to mitigate an accident or the radiological consequences of an accident. This activity did not increase the consequences of an accident previously evaluated in the SAR. The removal of cooling water to the drain attemporator had no effect on the operation of the house heat exchanger shell side relief valve. The relief discharge path to drain was not affected. The relief valve will lift at its setpoint without any effect. The overall effect of this change was a loss of quenching medium for the steam relief. This will result in steam pluming in the southeast corner of the Condenser Bay while the relief valve is open. Worst case estimate is a total relief volume of 1 % of the condenser bay volume. This volume will not cause any impact on equipment contained in the Condenser Bay and Heater Bay or cause any increases in general area temperatures, and it will not cause any increases in general radiation levels above normal background with the plant operating. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR, and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. There are no Technical Specifications for either the GSW System or the Auxiliary Heating System. There are no components that require GSW or the Auxiliary Heating System to maintain a margin of safety as defined in the basis for any Technical Specification. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-083 Installation Of Zebra Mussel Detection And Monitoring Equipment AT 'B' Side River Water Supply

Description and Basis of Change

This EMA installed a Bio-Box at the 'B' side River Water Supply (RWS) System at the River Water Intake Structure to monitor Zebra Mussels. Flow through the Bio-Box sample is approximately 1-2 gpm and the sample is returned to the RWS pump pit. The sample flow is approximately 0.017% of the required flow for the RWS system and is located upstream of the flow instrumentation for the RWS. Therefore, flow for the sample is not counted as part of RWS flow. This change revised P&ID M-129, adding new equipment identification numbers for three ball valves, a tank, and a Bio-Box. The Bio-Box provides flow through the test chamber for evaluating Zebra Mussel settling rate, growth rate, and adult treatment effectiveness. This detection device assists in monitoring Zebra Mussel infestation at the service water systems. Zebra Mussels in industrial facilities are biofoulers. Zebra Mussels enter a facility through unfiltered, untreated water drawn from an infested water body. They can enter the plant either as microscopic larvae (veligers) or as larger individuals ready for immediate colonization. The mussels can pose a significant threat to a facility where it is critical that flow is not reduced or impeded. If mussels are allowed to remain undisturbed on structures in the plant, they will grow and may reduce flow efficiencies. Zebra Mussels are prevalent in the Great Lakes and lower reaches of the Mississippi River. As a precautionary proactive measure a monitoring program should be established to determine the seasonal trends in abundance of both settled mussels and veligers.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Placement of the Bio-Box has no safety significance. The RWS System still delivers the required flow to the Pump House. The water sample used by the Bio-Box does not increase the probability of occurrence of an accident evaluated previously in the SAR. Installation of the sample point downstream of the manual isolation is outside the seismic boundary for the RWS System. The RWS System was maintained operational during the installation, and therefore the Bio-Box did not result in, or initiate an accident described in the SAR. This activity did not increase the consequences of an accident evaluated previously in the SAR. The input and output functions of the design safety standards were reviewed for the RWS. In all cases, the consequence of each accident evaluated previously is sufficiently extreme so as to envelop the consequence of any potential accident resulting from this activity. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This change did not affect any equipment that has a safety function. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not created. The RWS system still has the same probability of a malfunction as before the installation of the Bio-Box. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The installation of this Zebra Mussel detection device did not increase the possibility of associated equipment malfunction. Although unlikely, the types of events that this activity could create include the failure of the piping system that would develop leakage. The piping system was installed downstream of a Quality Level 1 isolation valve, and outside of a seismic boundary. The new piping system is rated well above the design and operating pressure and temperature of the RWS System. This activity did not create the possibility of an accident of a different type than previously evaluated in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The margin of safety as defined in the basis for any Technical Specification was not reduced.

SE 98-094 Installation Of Zebra Mussel Detection And Monitoring Equipment AT 'A' Side River Water Supply

Description and Basis of Change

This EMA installed a Bio-Box at the 'A' side River Water Supply (RWS) System at the River Water Intake Structure to monitor Zebra Mussels. Flow through the Bio-Box sample is approximately 1-2 gpm and the sample is returned to the RWS pump pit. The sample flow is approximately 0.017% of the required flow for the RWS system and is located upstream of the flow instrumentation for the RWS. Therefore, flow for the sample is not counted as part of RWS flow. This change revised P&ID M-129, adding new equipment identification numbers for three ball valves, a tank, and a Bio-Box. The Bio-Box provides flow through the test chamber for evaluating Zebra Mussel settling rate, growth rate, and adult treatment effectiveness. This detection device assists in monitoring Zebra Mussel infestation at the service water systems. Zebra Mussels in industrial facilities are biofoulers. Zebra Mussels enter a facility through unfiltered, untreated water drawn from an infested water body. They can enter the plant either as microscopic larvae (veligers) or as larger individuals ready for immediate colonization. The mussels can pose a significant threat to a facility where it is critical that flow is not reduced or impeded. If mussels are allowed to remain undisturbed on structures in the plant, they will grow and may reduce flow efficiencies. Zebra Mussels are prevalent in the Great Lakes and lower reaches of the Mississippi River. As a precautionary proactive measure a monitoring program should be established to determine the seasonal trends in abundance of both settled mussels and veligers.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Placement of the Bio-Box has no safety significance. The RWS System still delivers the required flow to the Pump House. The water sample used by the Bio-Box does not increase the probability of occurrence of an accident evaluated previously in the SAR. Installation of the sample point downstream of the manual isolation is outside the seismic boundary for the RWS System. The RWS System was maintained operational during the installation, and therefore the Bio-Box did not result in, or initiate an accident described in the SAR. This activity did not increase the consequences of an accident evaluated previously in the SAR. The input and output functions of the design safety standards were reviewed for the RWS. In all cases, the consequence of each accident evaluated previously is sufficiently extreme so as to envelop the consequence of any potential accident resulting from this activity. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This change did not affect any equipment that has a safety function. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not created. The RWS system still has the same probability of a malfunction as before the installation of the Bio-Box. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The installation of this Zebra Mussel detection device did not increase the possibility of associated equipment malfunction. Although unlikely, the types of events that this activity could create include the failure of the piping system that would develop leakage. The piping system was installed downstream of a Quality Level 1 isolation valve, and outside of a seismic boundary. The new piping system is rated well above the design and operating pressure and temperature of the RWS System. This activity did not create the possibility of an accident of a different type than previously evaluated in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The margin of safety as defined in the basis for any Technical Specification was not reduced.

SE 98-095 Removal Of Circulating Water Blowdown Flow Recorder

Description and Basis Of Change

The purpose of this EMA was to remove the Circulating Water Blowdown Flow recorder. The P&ID is in the UFSAR as figures 10.4-2 and 11.2-7 which have been revised to reflect the removal of recorder. Frequent repairs of the recorder were made due to failure of the chart drive motor. The chart drive motor failure was attributed to pump house environment and lack of lubrication. Since the Circulating Water Blowdown Flow data is also available from the Plant Process Computer, it was determined acceptable to remove the recorder from the system.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The recorder provided local indication only. This activity had no adverse effect on the Plant Process Computer point to provide the necessary data. Since there was no impact on the overall operation of the Circulating Water System, the consequences of an accident evaluated previously in the SAR were not increased. The probability of a malfunction of equipment important to safety evaluated previously in the SAR was not changed. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased since the Circulating Water System is not relied upon to prevent or mitigate an accident or the radiological consequences of an accident. As no new components or failure modes were added by this EMA, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The Circulating Water System function was not adversely affected by this activity. This change did not reduce the margin of safety as defined in the basis for any Technical Specification. There is no requirement per the National Pollutant Discharge Effluent System to have the local recorder for monitoring Circulating Water Blowdown Flow. Additionally, the Iowa Department of Natural Resources allows the use of process knowledge and engineering calculations to determine the flow when associated instruments are out of service. Thus, the removal of this recorder did not have any environmental consequences.

SE 98-097 Liquid Radwaste System Temporary Modifications

Description and Basis of Change

DAEC operating experience has revealed that from time to time it is necessary to employ temporary filtration or processing equipment to supplement the processing capability of the permanent Liquid Radwaste System. Such systems may include the use of equipment designed to address intrusions of liquid waste streams high in the level of organics, conductivity, turbidity or other water-borne chemical agent. In circumstances where temporary equipment is utilized, such equipment is either designed in a manner to be consistent with the pressure rating of the Liquid Radwaste System or pressure regulating devices will be used to ensure that the pressure does not exceed the design pressure of the temporary equipment. The effluents from the temporary equipment are returned to the Radwaste System for final processing prior to transfer to the Condensate Storage System or environmental release.

Safety Evaluation Summary

These modifications enhanced the DAEC's performance in the areas of liquid radwaste storage and treatment in total organic carbon reduction. The equipment introduced by these modifications are not initiators of any accident. These modifications did not increase the probability of occurrence of an accident evaluated previously in the SAR. The temporary equipment was either designed in a manner to be consistent with the pressure rating of the Liquid Radwaste System or pressure regulating devices were used to ensure that the pressure did not exceed the design pressure of the temporary equipment. Therefore, the modification did not increase the consequences of an accident evaluated previously in the SAR. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not increased. The integrity of liquid radwaste is required to prevent an excessive rate of leakage of liquids to the environs. Protection against accidental discharge is provided by instrumentation for the detection and alarm of abnormal conditions and procedural controls. Since these controls were not changed, this modification did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The Radwaste and the LLRPSF buildings are able to handle a major leak in the largest tank without permitting significant quantities of the liquid to escape off the site. The modification did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The Technical Specifications do not specify any margin of safety for the Liquid Radwaste System or its components. There are no changes required to the Offsite Dose Assessment Manual. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-098 Change Of Air Supply To Control Components Of The 'A' Control Building Air Conditioner Unit Cooling Coil Chill Water Control Valve

Description and Basis of Change

It was discovered the 20 psig H&V instrument air supply to the control components for the 'A' Control Building Air Conditioner Unit Cooling Coil Chill Water Control Valve was supplied by the "Common" H&V instrument air supply. This configuration was undesirable because the "Common" H&V instrument air supply automatically isolates to protect the 'A' and the 'B' side air supplies if a low pressure condition should occur. The loss of this air supply would result in a loss of the control air to control components resulting in the closure of the Control Valve, which would prevent chill water flow to the cooling coil of the 'A' Control Building A/C Unit. An EMA was initiated to change the instrument air supply for the Control Valve control components to the 'A' H&V Instrument Air Compressor System. This change is consistent with the design requirements of the Johnson Controls panel and provides a more reliable air supply to the control components for the Control Valve.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR since the Control Room HVAC is not a contributor or initiator of any transients, accidents, or special events as evaluated previously in the SAR. This change did not increase the consequences of an accident evaluated previously in the SAR since it ensures a separate safety related air supply to redundant chill water control components. The probability of a malfunction was not changed and the consequences of a malfunction of equipment important to safety previously evaluated in the SAR was not increased. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. This change did not affect the margin of safety as defined in the basis for any Technical Specification. The operation or method of operation of the chill water control to the Control Building HVAC did not change.

SE 98-099 Lower Overcurrent Setting On Low Level Radwaste Building Breaker

Description and Basis of Change

The load center breaker frame sizes or ratings are shown in UFSAR figures. The frame size was incorrect as shown for the Low Level Radwaste Processing and Storage Facility (LLRPSF) breaker 1BR825. The purpose of this EMA modification was to list the correct breaker size and settings on the proper documents and adjust the breaker to these settings as required. A plant drawing and an UFSAR figure were required to be revised to show the correct breaker frame size. The breaker was also adjusted to settings required to offer proper electrical protection of plant equipment and correct electrical coordination.

Safety Evaluation Summary

The design basis and breaker ratings were not changed as the breakers were not being replaced. The change to the plant drawings and the UFSAR did not impact the ability of the breaker to function. Adjustment of the breaker settings did not affect any safety related plant system or plant operation. This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Loads supplied by this breaker are not required for safe shut down of the plant. This modification offers better electrical system coordination and enhances the performance of the LLRPSF electrical system. This activity did not increase the consequences of an accident evaluated previously in the SAR. The probability of a malfunction has been reduced by improving the reliability of the breaker and the protection of its loads. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR have not been increased. This activity did not affect the operation of the LLRPSF or its capacity to perform its intended function. No possibility of an accident of a different type than any evaluated previously in the SAR was created by this modification. The probability of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not increased. This activity did not cause or create a failure mode not previously evaluated in the UFSAR. The function and design features of the LLRPSF electrical distribution system were not changed. This change did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-100 (Revision 1) ECP 1614 – Low Pressure Coolant Injection (LPCI) Check Valve Differential Pressure (dP) Modification

Description and Basis of Change

The inboard containment isolation valve for the Low Pressure Coolant Injection (LPCI) System inject lines are tilting disk check valves. These valves function to isolate the containment and the reactor coolant pressure boundary, and open to allow flow to the reactor vessel for several modes of the Residual Heat Removal (RHR) System including the LPCI mode. The operational leakage allowed for these valves is established at 2 gpm, and these valves are tested to this criteria every refueling outage. During plant operation, leakage past the check valve can pressurize the volume between the check valve and the normally closed outboard containment isolation valve (the Inboard LPCI Inject Valve) as high as the pressure in the discharge piping of the Reactor Recirculation System. When the pressure is equalized across the check valve, the normal pressure fluctuations in the Reactor Recirculation System can result in the valve disk alternately moving on and off the valve seat, damaging the check valve seat and wearing the hinge pin. This damage can be severe and has resulted in the check valves exceeding the allowable leakage criteria. The purpose of this modification was to create a sufficient differential pressure across the check valves to ensure that the normal pressure fluctuations in the Reactor Recirculation System will not result in movement of the valve disk off its seat, consequently limiting the potential for damage to the check valves. ECP 1614 installed a small cross tie between the LPCI injection lines and the Shutdown Cooling (SDC) line in the RHR System. This cross tie connected existing leakage test connections. A single test connection was installed to allow for hydrostatic and seat leakage testing.

The cross tie creates a dP of up to 190 psid to limit the potential for damage to the check valves from normal pressure fluctuations in the Recirculation System.

<u>Safety Evaluation Summary</u>

The accidents identified in the NSOA that may be affected by this modification are the LOCA inside primary containment or the LOCA outside of primary containment. The plant systems affecting these accidents are the LPCI mode of RHR and the Passive Containment System. The cross tie was designed as part of the reactor coolant pressure boundary, the design requirements for containment isolation are satisfied by the design, and the design did not create a high energy line break concern. Therefore, the design of the cross tie did not increase the probability of occurrence of an accident previously evaluated in the SAR. The radiological consequences of an accident were not increased. This activity did not increase the consequences of an accident evaluated previously in the SAR. The piping meets all the design criteria so that the probability of a pipe rupture was not increased. The primary containment isolation logic is not affected by the cross tie, but the isolation is no longer uniquely provided by the valves in the given line. Flow through the cross tie from a given line is isolated by the outboard isolation valves on the other lines. The pressure isolation function between the high pressure reactor coolant system and the lower pressure RHR system is improved as the leaktightness of the check valves on the LPCI lines are significantly improved without adversely effecting the leaktightness of the other isolation valves. The cross tie will not result in an overpressurization of the RHR System as the relieving capability is greater than the cross tie flow. Therefore, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not increased, and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. The consequences of a malfunction are not changed as the area of the cross tie is small and has a negligible effect on the systems being connected. The possibility of an accident of a different type than any evaluated previously in the SAR was not created, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The margin of safety as defined in the basis for any Technical Specification was not reduced. The margin of safety for the Primary Containment and Emergency Core Cooling Systems (ECCS) was not affected by the cross tie.

SE 98-101 Removal Of Service Platform And Refueling Interlocks

Description and Basis of Change

The Service Platform was provided to the DAEC as part of the basic equipment used to perform certain activities during Refueling Outages. Over the course of many years, the usefulness and the condition of the Service Platform resulted in actions being taken for its removal and eventually its disposal. The Service Platform was disposed of after RFO 14 in 1997. The platform was in a state of disrepair and replacement parts were nearly impossible to obtain. Tooling and techniques for work performed in-vessel demonstrated that the Service Platform was no longer considered a viable tool and should be removed from the Refuel Floor. The Service Platform was retired as part of ECP 1606. The activities performed using the Service Platform were eliminated and addressed in safety evaluation SE 98-18. The activities once performed with the Service Platform are now performed using the Refueling Platform. The disposal of the Service Platform provided justification for removal of the circuitry and wiring associated with the platform and the Refueling Interlocks. Removal of the platform and the circuitry associated with the Refueling Interlocks removed inputs to the Reactor Manual Control System (RMCS). These inputs to RMCS were used to prevent core alterations when the Service Platform was installed. The UFSAR has been updated to reflect the removal of the components from the plant.

Safety Evaluation Summary

This change actually reduced the probability of an accident evaluated previously in the SAR. Core alterations cannot be performed by the Service Platform anymore, hence, there is no way a reactivity excursion can occur due to a control rod withdrawal simultaneously with fuel being handled by the Service Platform. The Service Platform can no longer lift a load over irradiated fuel, hence, eliminating the probability of occurrence of a fuel handling accident. This change did not increase the consequences of an accident already evaluated in the SAR because the activities once performed by the Service Platform are no longer performed by the platform. The inputs to the Refueling Interlocks and the RMCS from the Service Platform no longer exist so they cannot increase the consequences of an accident. The probability of occurrence of a malfunction is reduced because the Service Platform is no longer transported over the reactor vessel or irradiated fuel. The absence of the Service Platform, and its inputs into the Refueling Interlocks and RMCS did not introduce any new techniques or activities that may result in an increase in the possibility of any accident. The removal of the Service Platform and the circuitry associated with the Refueling Interlocks and

RMCS did not reduce the margin of safety as defined in the Technical Specifications. The interlocks associated with the Service Platform were removed from the Technical Specifications when the Technical Specifications were revised to the Improved Technical Specifications. No new techniques, procedures or activities resulted from these changes, therefore, no new accidents or consequences were introduced.

SE 98-103 Condensate Demineralizer Panel Annunciators EMAs

Description and Basis of Change

Changes were made to the annunciators on the Condensate Demineralizer Panel associated with the Condensate Filter/Demineralizer Precoat Recirc Header High Pressure alarm and the Condensate Filter/Demineralizer Precoat/Backwash Header Low Pressure alarm. The Condensate Filter/Demineralizer Precoat Recirc Header High Pressure alarm annunciated window D-7 and received a signal from pressure switch (PS) 1760 and PS1761. The Condensate Filter/Demineralizer Precoat/Backwash Header Low Pressure alarm annunciated window D-6 and received a signal from a second switch PS1761. The Condensate Filter/Demineralizer Precoat/Backwash Header Low Pressure alarm was not part of original plant equipment. This alarm was installed by a Design Change Request, but was never activated. Both these alarms had been disabled. Since PS1760 and PS1761 both provided a signal to annunciator window D-7, it was difficult for the operators to determine which header was actually seeing a high pressure condition. To alleviate this problem the high pressure switch in PS1761 has been connected to annunciator window D-6 and provides a signal on high pressure in the precoat and backwash header. The low pressure switch has been removed from PS1761. Annunciator window D-7 now only receives a signal from PS1760.

Safety Evaluation Summary

None of the accidents previously evaluated in the SAR are affected by the conversion of the low condensate service water pressure alarm on the precoat and backwash headers to a high pressure alarm. This part of Condensate Demineralizer System is not safety related. There are no credible ways of increasing either the probability of occurrence of an accident or the consequences of any of the accidents evaluated in the SAR as a result of this activity. This activity involved the conversion of the Low Condensate Service Water Pressure Alarm on the precoat and backwash header to a high pressure alarm. This alarm is part of the Condensate Demineralizer System, which is non-safety related and can not impact the operation of any safety related equipment. There are no credible ways that this activity could increase either the probability of occurrence or the consequences of a malfunction of equipment important to safety as evaluated in the SAR. The Low Condensate Service Water Pressure Alarm was not original plant equipment. This alarm was installed but never activated. The high pressure alarm already existed in the plant and was connected to a common annunciator that was disabled. There are no credible failures that could create the possibility of an accident not previously evaluated or increase the possibility of malfunction to any equipment important to safety not previously evaluated in the SAR. There is no reference to this alarm or the Condensate Demineralizer System in the basis for any Technical Specification. Therefore, the conversion of the Low Condensate Service Water Pressure Alarm to a high pressure alarm did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-105 Fire Pumps' Test Return Line Valve Replacement

Description and Basis of Change

An EMA replaced the Diesel and the Electric Fire Pump Test Return Valve. This is a normally closed valve and its primary function is to provide isolation. This valve is throttled to provide a pressure breakdown when performing operability testing of the fire pumps. Cavitation occurred when the Vee-Ball valve was required to provide a large pressure breakdown, resulting in excessive line vibration and valve damage. The valve was changed from a Vee-Ball valve to a caged globe valve. The new valve was designed to provide the pressure breakdown required by the operability tests without causing cavitation and excessive line vibration.

Safety Evaluation Summary

The change from a Vee-Ball valve to a cage type globe valve within the Fire Protection System did not increase the probability or consequences of an accident because fire is not an entry condition, basis or assumption for any accident previously evaluated in the SAR. This change did not increase the probability or consequences of a malfunction to safety related equipment already analyzed in the SAR because replacing the valve did not affect the ability of any safety related equipment to perform its safety function. This change did not create the possibility of an accident or a malfunction of safety related equipment of a type not already analyzed in the SAR. The new style cage globe valve has been designed to reduce the cavitation and to provide better throttling capabilities to assist with the flow test performed on the Diesel and Electric Fire Pumps. The Fire Protection System is not part of Technical Specifications, therefore, this change could not impact the margin of safety as defined in the basis for any Technical Specification.

SE 98-108 (Revision 1) **Removal Of River Water Supply (RWS) System Conductivity And pH Instrumentation**

Description and Basis of Change

EMAs removed the river water supply conductivity and pH instrumentation, valves and piping associated with the instruments. The river water supply conductivity instrumentation was no longer used by the Chemistry Department and Operations Department. The pH and conductivity instrumentation was not used in any surveillance tests and they were not used to determine system operability.

Safety Evaluation Summary

This activity did not increase the probability of an accident evaluated previously in the SAR. The RWS conductivity instruments provide indication only. The removed instrumentation did not provide any safety function. This activity increased the amount of water supplied to the plant because the pH and conductivity elements diverted a small amount of river water through the sampling process. This change did not increase the consequences of an accident previously evaluated in the SAR. The removal of this instrumentation did not alter the operation of the RWS System. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not increased and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. This change did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The removal of the instrumentation reduced the location for a potential leakage from the system. The removal of the instrumentation did not create any new failure modes. This change did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR, and the margin of safety as defined in the basis for any Technical Specification was not reduced. The RWS System still meets the Technical Specification requirements of providing sufficient coolant to the Residual Heat Removal Service Water System.

SE 98-110 **Turbine Building Auxiliary Heating Loop Temperature Controller Upgrade**

Description and Basis of Change

This EMA replaced the Turbine Building Auxiliary Heating Loop Temperature Controller since the failed temperature control unit did not calibrate and the model was obsolete. This change required the system P&ID to be revised, which resulted in an UFSAR change.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR, and did not increase the consequences of an accident evaluated previously in the SAR. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. This change did not reduce the margin of safety as defined in the basis for any Technical Specification. The operation of this unit does not directly or indirectly affect the margin of safety of any safety limit or limiting safety system settings. The overall system is made more reliable by allowing for proper automatic operation. The affected equipment is not safety related, and the upgrade had no adverse effect on plant operation. The new unit is equivalent or superior to the old unit. The probability of occurrence and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased.

SE 98-111 ECP-1611 - Ericsson Wireless Telephone System

Description and Basis of Change

This change to the facility involved the installation and operation of a local, pico-cellular, Wireless Telephone System in selected areas of the plant. The purpose of this change was to enhance the communication capabilities within the plant. A need was identified for improved communications within the Power Block, especially during refueling outages when a significant amount of Radiation Protection coverage is required. Although the DAEC telephone system is not a safety system, it is relied upon for effective communications during all plant conditions, including both normal operations and emergencies. The Wireless Telephone System interfaces with structures, systems, and components that are designated as safety-related, including the reactor building structure (secondary containment) and the -48 VDC battery. Through a deliberate process of technical evaluation, DAEC-specific testing, and industry experience research, the DAEC has established high confidence in the electromagnetic compatibility between the wireless telephone equipment and existing plant equipment.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Evaluation, testing, and industry experience has demonstrated general electromagnetic compatibility of the Ericsson Wireless Telephone System and nuclear power plant equipment. There are no credible effects of the Ericsson wireless telephone equipment that would lead to the initiation of an accident previously evaluated in the SAR. The twisted-pair wiring for the system uses existing conduit penetrations through secondary containment. There are no credible effects of the Ericsson wireless telephone equipment which would increase the consequences of an accident previously evaluated in the SAR, or lead to the malfunction of structures, systems, or components as described in the SAR. There are no credible effects of the Ericsson Wireless Telephone System that would challenge a fission product barrier or lead to failure of plant structures, systems, or components. Similarly, the installation and operation of the Ericsson equipment would not cause an increase in the radiological consequences of a malfunction of plant equipment important to safety. There is no credible mechanism for the Ericsson wireless telephone equipment to create or initiate an accident, including any of a different type than those previously evaluated in the SAR. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. This change did not affect the margin of safety as defined in the basis for any Technical Specification.

SE 98-112 Installation Of Evacuation Alarm In The Technical Support Center (TSC)

Description and Basis of Change

A multi-tone generator and evacuation alarm was installed in the TSC. This allows the emergency response personnel in the TSC, when it is operational, to activate the evacuation alarm from the TSC. Previously activation of the evacuation alarm was coordinated between the TSC and the Control Room. This is an enhancement to the TSC and the Emergency Response Organization. The scope of this activity involved installing a multi-tone generator and evacuation button on the east wall of the TSC. This was tied into the existing page system. A statement in the UFSAR required revision as a result of this EMA.

Safety Evaluation Summary

The evacuation alarm is part of the communication system. The communication system has no safety function. The communication system is provided in the plant to ensure reliable communications for startup, operation, shutdown, and maintenance under all normal and emergency conditions. The evacuation alarm portion of the power generation design basis is that the alarm signals can be transmitted over the page system to warn personnel of emergency conditions. The installation of a multi-tone generator and evacuation button in the TSC did not affect any DBD, and none of the accidents previously evaluated in the SAR were affected. There are no credible ways of increasing either the probability of occurrence of an accident or the consequences of any of the accidents evaluated in the SAR as a result of this activity. The evacuation alarm is manually operated and does not affect any safety related, structure, system, or components. There are no credible ways that this activity could increase either the probability of occurrence or the consequences of a malfunction of equipment important to safety as evaluated in the SAR. There are no credible failures that could create the possibility of an accident not previously evaluated or increase the possibility of malfunction of any equipment important to safety not previously evaluated in the SAR. There is no reference to the communication system in Technical Specifications. Therefore, there is no possibility of this activity reducing the margin of safety as defined in the basis for any Technical Specification.

SE 98-115 Temporary Modification To Measure Pressure At Inlet of Feedwater Flow Nozzles

Description and Basis of Change

This activity involved the installation of a temporary modification to measure the pressure at the inlet of the feedwater flow nozzles for the purpose of correcting the alternate flow measurement (ultrasonic flow measurement) for the pressure effect on the fluid and the piping. Two temporary pressure transmitters were installed on the high side calibration/drain valves of the A and B Feedwater Loop Flow Element Differential Pressure Indicators. These instruments are shown on P&ID BECH-M107, which is a Rack/Position drawing considered essential to plant operations. The transmitters were installed only long enough to obtain accurate correlations to existing pressure instrumentation at various flows and load ranges and were removed at the end of that time.

Safety Evaluation Summary

There was no credible way this activity could contribute to the occurrence of any accident previously evaluated in the SAR. No safety significant systems were involved in any way during this activity. The taps are completely separate from the primary taps used for reactor level control and did not affect the ability of the Feedwater System to maintain the core covered. The consequences of an accident evaluated previously in the

SAR were not increased. The Loss of Feedwater Flow accident discussed in the UFSAR was not affected because anything done with the secondary taps cannot affect the signal from the primary taps, which are the ones used for the control system. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR, or the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The possibility of an accident of a different type than any evaluated previously in the SAR was not created. This activity did not have the potential to create any kind of accident, therefore, there was no possibility of creating an accident of a different type than previously evaluated. Since this activity did not involve any equipment important to safety, and no equipment important to safety was added, the possibility of a malfunction of a different type than was previously evaluated was not possible. Since no Technical Specification was involved in this activity and no equipment involved in any Technical Specification was involved in this activity, no margin of safety for any Technical Specification was reduced.

SE 98-117 Temporary Modification To Low Pressure Coolant Injection (LPCI) System Inject Lines

Description and Basis of Change

The purpose of this temporary modification was to remove a pipe cap and install a temporary pressure indicator, vent valve, and associated fittings and tubing to piping downstream of Leakage Test Connection Isolation Valves V20-0118 and V20-0119. This was done in order to obtain pressure readings on, and to bleed off the piping between the LPCI Inject Check Valves and the Inboard LPCI Inject Valves on the 'A' and 'B' LPCI Inject Lines. This created a small steam void in the LPCI Inject Line piping near the LPCI Inject Check Valves to aid in creating a differential pressure across the LPCI Inject Check Valves for longer periods of time such that their discs would be held tightly into their seats in order to prevent the discs from moving and wearing the hinge pins. V20-0118 and V20-0119 are considered to be part of the primary containment and reactor coolant pressure boundary. Although the temporary components were not considered fully qualified for containment and reactor coolant pressure boundary functions, they were qualified to the necessary pressure rating and an operator was stationed at V20-0118 and V20-0119 when venting to maintain the containment pressure boundary function acceptable.

Safety Evaluation Summary

The probability of occurrence of an accident previously evaluated in the SAR was not increased because the temporary components were adequately qualified for pressure rating while V20-0118 and V20-0119 were open. There was no effect on the permanent piping/valves when V20-0118 and V20-0119 were closed. Opening the permanently installed isolation valves to measure pressure and to vent the 'A' and 'B' LPCI Inject Lines was under the direct control of an operator to provide quick re-isolation if necessary. The capability of the Residual Heat Removal (RHR) System to maintain its safety functions was not adversely affected. There were no effects on dose consequences, or fission product barriers. This activity did not increase the consequences of an accident evaluated previously in the SAR. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR was not increased. The temporary components did not adversely affect the LPCI Inject piping's seismic criteria, separation criteria or environmental qualifications. No equipment protection features were deleted or modified. Support system performance necessary for reliable operation of the primary containment and reactor coolant pressure boundaries was not affected. System/equipment redundancy or independence was not affected. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR were not increased because a break of any line at the location between the primary containment and reactor coolant pressure boundary isolation valves was already addressed in the SAR. The possibility for an accident of a different type than any evaluated previously in the SAR was not created, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The margin of safety was not reduced.

SE 99-001 Temporary Modification Of Control Building Envelope

Description and Basis of Change

This activity involved a temporary modification to the exhaust ducting of the 'A' Control Building Standby Filter Unit (SFU) to allow preplanned replacement of the charcoal. This activity required disassembly of the SFU train and removal of ducting that opened the control building envelope. This Temporary Modification installed a blank flange in the duct opening to maintain the control building envelope throughout the replacement of the SFU charcoal.

Safety Evaluation Summary

The SFU and Control Building Ventilation Systems are support systems that are not accident initiators. Temporarily modifying the ducting of one SFU did not have any effect on accident initiating systems. Therefore this modification did not increase the probability of occurrence of an accident evaluated in the SAR, and the consequences of an accident evaluated previously in the SAR were not increased. The probability of a malfunction of equipment important to safety, and the consequences of a malfunction of equipment important to safety were not increased. This temporary modification did not constitute a permanent change to the UFSAR. This temporary modification did not adversely affect the operation of the Standby Filter Units, and the 'B' SFU continued to operate within the design calculations that are the basis of the SAR. The possibility of an accident of a different type than any evaluated previously in the SAR was not created, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. Throughout this activity, the 'B' SFU remained operable per the Technical Specifications, and the 'A' SFU was considered inoperable and the Technical Specification LCO was entered. The margin of safety as defined in the basis for any Technical Specification was not reduced.

SE 99-002 Installation Of Vent Line In Auxiliary Boiler Steam Supply Line

Description and Basis of Change

This EMA activity added a vent line, containing an isolation valve, to the Auxiliary Boiler steam line that supplies steam to the liquid nitrogen vaporizer and the High Pressure Coolant Injection System and the Reactor Core Isolation Cooling System test runs. This modification allows draining of condensate from the Auxiliary Boiler steam lines prior to admitting steam, which prevents water hammer when steam is admitted. The vent line and isolation valve allow the system to be vented without opening Secondary Containment.

Safety Evaluation Summary

The components added by this modification were designed and fabricated to the same specification as the piping to which they were attached. The Auxiliary Heating System Boiler is not included in the NSOA, is not an accident initiator, and is not required to mitigate the consequences of any accident evaluated in the SAR. None of the affected components have any safety significance. This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR, and the consequences of an accident evaluated previously in the SAR were not increased. The probability of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. Equipment important to safety was not impacted by this modification. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR was not increased. No new failure modes were created by this change. The possibility of an accident of a different type than any evaluated previously in the SAR was not created, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The margin of safety as defined in the basis of any Technical Specification was not reduced since the Auxiliary Heating System is not included in the DAEC Technical Specification and no equipment important to safety was impacted.

SE 99-003 Temporary Modification For Main Condenser Air In-leakage Test

Description and Basis of Change

Air in-leakage test equipment was installed to test the gases that travel through the Offgas System to determine the location of the Main Condenser air in-leakage. The sample was taken at the discharge of the Steam Jet Air Ejector (SJAE) and was analyzed for trace amounts of helium gas. The helium gas was sprayed in areas of the plant, where air in-leakage to the Main Condenser may have been occurring. With the tracer gas plus the normal in-leakage into the Main Condenser, the location of the air in-leakage was determined. Once the components that were leaking were located, the components were repaired as soon as possible. The Offgas and Main Condenser Systems were in operation during the in-leakage testing. The helium gas did not increase the amount of non-condensable gases in the Offgas System. This tracer gas replaced a small amount of normal air in-leakage, therefore the gases traveling through the systems remained the same. Therefore, the systems operated the same during the in-leakage test as before the installation of this temporary modification. The Offgas release rate did not change during this test. Additionally helium gas is an inert gas that did not react with the other gases in the Offgas System.

Safety Evaluation Summary

Since operation of the affected systems was not altered, the probability and the consequences of the occurrence of an accident were not increased. The affected sections of these systems were not required to perform any operation during any accident as described in the SAR. Therefore, since this equipment was not required to perform any safety function, the probability and the consequences of a malfunction of equipment important to safety were not changed by the installation of this temporary modification. The helium gas did not react with the other non-condensable gases traveling through the Offgas System. The SAR contains an evaluation for a main steam line break outside of containment, which is a much larger leakage point of radioactive material than all of the normal flow through the Offgas System. Therefore, an accident with a release rate larger or of a different type than previously evaluated in the SAR was not created. The activity release rate was maintained below the limits in the Technical Specification. Therefore, this activity did not reduce the margin of safety as defined in the basis for the Technical Specifications.

SE 99-004 Replacement Of Main Steam Line Low Pressure Switches

Description and Basis of Change

The Bourdon tube operated pressure switches, for Main Steam Line Low Pressure, demonstrated higher set point drift than desired for the function they performed and were being calibrated on a monthly basis. An EMA replaced these pressure switches with diaphragm operated switches which provide improved set point stability, resulting in more reliable performance of the switch function.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The Nuclear Steam Supply Shut-off System reacts to accidents and does not cause accidents. The switch set point was not changed. This activity did not increase the consequences of an accident evaluated previously in the SAR. The ability of the replacement switch to support the Allowable Value given in Technical Specifications was evaluated in a Set Point Calculation. The replacement switch actuates above the Allowable Value. There was no increase in the consequences of an accident evaluated previously in the SAR. This activity did not increase the probability of occurrence of a malfunction of equipment evaluated previously in the SAR. The diaphragm operated pressure switches have less set point drift than the Bourdon tube operated pressure switches. This activity did not increase the consequences of a malfunction of equipment previously evaluated in the SAR. The system response to switch failure was not changed by this activity. The consequences of an equipment malfunction were not increased. This activity did not create the possibility of a different type of accident than previously evaluated in the SAR. A new accident type was not generated by pressure switch failure, since the Bourdon tube and diaphragm operated switches have the same general failure modes. The sharing of general failure modes also eliminates the possibility of a malfunction of equipment that is of any different type than previously evaluated in the SAR. This activity did not reduce the margin of safety as defined in the Technical Specifications.

SE 99-007 Primary Containment H_2O_2 Analyzer Changes

Description and Basis of Change

This EMA installed an isolation valve for the inlet to the sample pump on the Primary Containment H_2O_2 Analyzers and changed the percent concentration of the reagent and calibration gases. Previously, when a sample pump was replaced, a pressure test of the piping connections to the Primary Containment H_2O_2 Analyzer was required. To alleviate the potential for equipment damage and to minimize the LCO time required for pump replacement, a sample pump inlet isolation valve was installed between the accumulator and the sample pump inlet. This allows isolating the sample pump from the rest of the analyzer for maintenance and testing. The sample pump inlet isolation valve was installed on both the 'A' and 'B' Primary Containment H_2O_2 Analyzers.

Changing the percent concentration of the calibration gas and/or the reagent gas did not affect the operation of the analyzer. It may be desirable to change these concentrations in order to assure adequate flow across the pressure control valves or to tighten the calibration span. Calibration of an analyzer sets up the machine to the correct mass flow rates for the respective gas concentrations. This is performed by a Surveillance Test Procedure in accordance with Technical Specifications.

Safety Evaluation Summary

The safety function of the Containment Atmosphere Control System that is specific to the H_2O_2 analyzers is monitoring the oxygen and hydrogen concentrations in the containment atmosphere. None of the criteria specified in the DBDs were affected by this activity. All of the accidents in the UFSAR were reviewed with respect to this modification. The sample pump inlet isolation valve is a manually operated valve, whose position is verified during testing prior to the H_2O_2 analyzer being returned to service. The function of the valve is for maintenance testing only. Changing the percent concentration of the gases does not adversely impact the operation of the analyzers. Therefore, there are no credible ways of increasing either the probability of occurrence of an accident or the consequences of any of the accidents evaluated in the SAR. There are no credible failures that could increase either the probability of occurrence or the consequences of an accident evaluated previously in the SAR. The probability of a malfunction of equipment important to safety as evaluated in the SAR was not increased. There are no credible failures that could create the possibility of an accident not previously evaluated or increase the possibility of malfunction of any equipment important to safety of a different type than any evaluated previously in the SAR. There is no possibility of reducing any margin of safety as defined in the basis of Technical Specifications.

SE 99-008 **ECP 1615 – Feedwater System Ultrasonic Flow Instrumentation**

Description and Basis of Change

Fouling of flow measuring element nozzles in the Feedwater System caused inaccuracy in the measurement of Feedwater System flow. This activity involves the correction of a known inaccuracy in the feedwater flow value that is supplied to the Plant Process Computer (PPC) Core Thermal Power (CTP) calculation in the PPC. This correction factor is applied when fouling is present in the feedwater flow nozzles or to compensate for a non-conservative instrument failure. The correction is determined by an independent measurement of the feedwater flow by a cross-correlation type flow meter. This instrument is not connected to any control circuits. The accuracy, repeatability and dependability of this equipment is high. Existing transient and LOCA analyses bound the results of this activity.

Safety Evaluation Summary

This activity does not increase the probability of occurrence of an accident evaluated previously in the SAR. No change is made to primary system pressure boundaries that would increase the likelihood of a LOCA event. Likewise, no change is made to the Control Rod Drive System hardware, vessel internals, fuel handling equipment, or procedures that would increase the likelihood of a control rod drop accident or of a fuel bundle drop accident. Therefore, the probability of occurrence of an accident evaluated previously in the SAR is not increased. This activity only involves the application of a multiplier to the process computer's feedwater flow equation that allows operation up to the actual rated CTP for the DAEC. Since this change improves assurance that core conditions will be maintained within the range of conditions assumed by the DAEC Accident Analysis, the consequences of an accident evaluated previously in the SAR is not increased. No physical change was made to Feedwater System components or to control devices that would increase the likelihood of a malfunction. In addition, no change was made to other equipment that could cause a plant transient upon failure or which provide a protective function for plant transients. Therefore, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased. Since expected conditions resulting from this change are within the range of initial conditions assumed by DAEC transient analyses, the consequences of a malfunction of equipment important to safety evaluated previously in the SAR is not increased. No new means for bypassing or failing radiological barriers that could result in off-site doses were created. The possibility of an accident of a different type than those described in the SAR is not introduced, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR is not created. The methodology for analyzing LOCA and plant transient events provides conservative margins for feedwater flow and CTP uncertainty. A detailed review of uncertainty analysis for transients concluded that these margins would not be exceeded as a result of application of this correction factor. In addition, thermal limits established from thermal-mechanical design analysis were not impacted by this change because instrument uncertainty is accounted for in these analyses, and CTP is maintained within the bounds of this assumed uncertainty. Therefore, margins of safety as defined in the basis for Technical Specifications are not reduced.

SE 99-015 Installation Of Temporary Covers To Reduce Main Condenser In-Leakage

Description and Basis of Change

This Safety Evaluation was written to install a temporary modification at the Steam Seal Supply Header Bypass Line Pressure Relief Valve in the Turbine Steam Seal System. This temporary modification installed covers on the open vacuum pressure compensating line and the open bonnet plug for the relief valve. These covers were installed on the vacuum side of the system. The temporary covers were not designed as pressure retaining components. These two covers reduced in-leakage by approximately 70 scfm.

Safety Evaluation Summary

The relief valve continued to operate the same as before the installation of these covers. The Turbine Steam Seal System is not safety related and the installation of these covers did not affect the operation of the Turbine Steam Seal System. This activity did not increase the probability or the consequence of an accident evaluated previously in the SAR. The probability and the consequences of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR were not increased. Since the Turbine Steam Seal System does not have any equipment important to safety, a malfunction in this system can not increase the probability or the consequences of any accident. The relief valve is not required to be seismically installed and the addition of the covers did not change this requirement. The possibility of a malfunction of equipment important to safety of different type than any evaluated previously in the SAR was not created. If the cover were to fail, all of the safety systems would be unaffected and the plant would return to air in-leakage levels as before the installation of these covers. The Turbine Steam Seal System and the relief valve do not have any requirements in the Technical Specifications or in the basis for Technical Specifications, therefore no margin of safety was reduced.

SE 99-018 ECP 1612 – Replacement Of Recorders

Description and Basis of Change

Eleven Honeywell recorders were replaced in Control Room Panel 1C029 with Westronics series 1600 digital recorders. These recorders monitor Stilling Basin Level, Torus Water Temperature, Torus Water Level, Torus Air Temperature, Make-up Nitrogen Supply Line Pressure, and Containment Atmosphere including Air Temperature, Pressure and Radiation levels. The Honeywell recorders were becoming obsolete and costly to repair due to frequent failures of the slidewire assembly. Analog signals are converted into digital equivalents by analog to digital (A/D) converter modules, which are part of the Westronics series 1600 recorder. Each recorder has one common module, which processes the signals from as many as four different inputs. The Westronics series 1600 recorder internal modules utilize the digital signal and are not subject to tolerance or drift as are the Honeywell recorders. Therefore, the Westronics series 1600 recorder operates with increased accuracy and provides more reliability when compared to the analog recorders.

Safety Evaluation Summary

This modification did not affect the functions or requirements of the Drywell Radiation Monitoring System, Containment Atmosphere Control System, River Water Supply System and Primary Containment System. This modification did not increase the probability of occurrence of a malfunction of equipment important to safety. Software common cause failure is not a significant failure as the new recorders are only used for data gathering and do not perform any automatic function required for safe shutdown. This activity did not increase the probability or consequences of an accident previously evaluated in the SAR. This activity increased the reliability of the instrument loops to provide required parameters. None of the recorders provides a control signal to the logic of any circuit important to safety. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. The replacement Westronics digital recorders are not used as control devices and do not provide input to any system that is important to safety. They are used as indication only instruments. The possibility of a malfunction of equipment important to safety of a different type was not created, and the safety margin as defined in the Technical Specifications was not adversely affected by this activity.

SE 99-022 Temporary Modification On Residual Heat Removal (RHR) Inboard Shutdown Cooling Suction Isolation Valve

Description and Basis of Change

This temporary modification eliminated the influence of a 120 VAC signal in the cables from the RHR Inboard Shutdown Cooling Suction Isolation Valve to relays affecting the rest of the 125 VDC System and blowing 'A' RHR Logic fuses. This modification lifted the leads from the RHR Inboard Shutdown Cooling Suction Isolation Valve to relays K15A and K16A and installed a jumper in the Group IV isolation logic. This jumper maintained the low reactor water level and high drywell pressure isolation functions of the Low Pressure Coolant Injection (LPCI) inboard isolation valve isolation logic operable. The various modes of the RHR System remained operable while this temporary modification was installed. Additional procedural actions were provided to ensure the jumper was removed as part of the manual operator action needed to reset a Group IV isolation and realign RHR/LPCI for injection.

Safety Evaluation Summary

The probability of occurrence of an accident previously evaluated in the SAR was not increased because the "No suction path" pump trip feature and the Inboard LPCI Inject Valve isolation feature are not accident initiators per the SAR. The proposed activity did not increase the consequences of an accident evaluated previously in the SAR. The Inboard LPCI Inject Valve isolation function was maintained by administrative controls. This consisted of installing a jumper in the Group IV isolation logic only when in the Shutdown Cooling mode to ensure functionality of the low reactor water level and high drywell pressure isolation of the LPCI inboard isolation valve. Thus, there was no effect on the fission product barriers or dose consequences. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR was not increased. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR were not increased for the RHR pump since the probability of damage was not increased. For the LPCI inject valve isolation logic, the installation of the jumper in the Group IV valve isolation logic ensured that the isolation function was maintained when the plant was in the Shutdown Cooling mode. 10 CFR 100 limits were not exceeded.

Therefore consequences of a malfunction were not increased. The possibility of an accident of a different type than any evaluated previously in the SAR was not created. No new failure modes were created as a result of this activity since the worst case failure mode is the loss of an RHR pump. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The margin of safety was not reduced. Based on a review of the Technical Specifications, Technical Specification Bases, UFSAR and NSOA, there was no margin of safety defined which could have been affected by the installation of the temporary modification.

SE 99-025 Temporary Modification Of Drywell Cooling High Temperature Switches

Description and Basis of Change

Leads were lifted to remove the 120VAC power to a portion of the 'A' side Drywell high temperature switch circuits that enter/exit the Drywell via 1JX105A header hole No. 23. Troubleshooting had identified short circuits between this circuit and other circuits in header holes 17 and 23. Lifting these leads was necessary to prevent propagation of the problem to other circuits. Temperature switches associated with CRD area temperature and Drywell fan-coil units' air temperatures were affected by this temporary modification. Two of the switches close if CRD area temperature reaches 150°F. Seven of the switches close if the fan-coil units' exhaust air temperature reaches 135°F. One switch closes if a fan-coil unit's intake air temperature reaches 150°F. The switches provide the following functions:

- Open Drywell Cooling Water Loop 'A' Well Water Isolation Valves if the 'A' System mode handswitch is positioned to BACKWASH and a high CRD area or fan-coil unit high air temperature condition is present.
- Open Drywell cooling coil inlet and outlet valves if the Cooling Coil Control Logic handswitch is positioned to OPEN or CLOSE, System 'A' mode handswitch is in BACKWASH or STANDBY and a CRD area or fan-coil unit high air temperature condition is present.
- Start Drywell return air fans in high speed if the System 'A' mode handswitch is in BACKWASH or STANDBY and a fan-coil unit high air temperature condition exists. In addition, CRD area return air fans (CRD Area Cooling Coil Exhaust Air Recirculation Fan and Drywell CRD Area Cooling Coil Exhaust Air Recirculation Fan) will start in

high speed if the System 'A' mode handswitch is in BACKWASH or STANDBY and a CRD area high temperature condition is present.

This temporary modification lifted the leads for the switches so that the switches no longer provide the above-described functions. This change did not adversely impact the requirement for Primary Containment Heat Removal. The Drywell Coolers still provide cooling. No difference in actual system operation occurred during normal reactor power operation. Redundant automatic actions and operator actions in response to alarms established adequate cooling during backwash or standby operation while in shutdown.

Safety Evaluation Summary

The Primary Containment Cooling System is required for normal power operation only. The cooling function is not safety-related and not required for the mitigation of an accident. The probability of occurrence of a malfunction of equipment important to safety was not increased, nor were the consequences of a malfunction of equipment important to safety increased. Lifting the leads for the high temperature switches and changing the logic did not adversely affect the ability of the containment isolation valves to close, or the seismic qualification of the system. The probability of an accident or the consequences of an accident were not increased. All of the Drywell coolers are normally operated in high speed so the temperature switches do not play a role in their control during reactor power operation. The possibility of an accident of a different type than evaluated in the SAR was not increased. The possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the UFSAR was not increased. The requirements in the Technical Specifications were not impacted. The margin of safety as defined in the basis for the Technical Specification was not reduced. Lifting leads associated with the temperature switches did not adversely affect safety.

SE 99-026 Replacement Of Liquid Radwaste Filter Flow Transmitter

Description and Basis of Change

This EMA replaced the Liquid Radwaste Filter Flow Transmitter. The replacement flow transmitter requires the use of a preamplifier to amplify the signal from the flow element. This new model is a vendor recommended replacement. The previously installed flow transmitter did not require a preamplifier.

Safety Evaluation Summary

This activity did not affect the inputs into any accident analysis performed for DAEC. Therefore, the installation of the Floor Drain Filter Outlet Flow Preamplifier did not increase the probability of occurrence of any accident evaluated previously in the SAR, and the consequences of any accident evaluated previously in the SAR were not increased. The affected equipment is not classified as Nuclear Safety Related. This change did not create any situation where equipment important to safety would be compromised. The Radwaste System is classified as Quality Level 2, which invokes quality assurance requirements for the pressure boundary components of the Radwaste System only. This activity did not affect the pressure boundary of the Radwaste System. Therefore, this change did not affect the probability of occurrence of a malfunction of any equipment important to safety. This change did not result in increased radiological exposure to plant personnel or the public. Therefore, the consequences of a malfunction of any equipment important to safety evaluated previously in the SAR were not increased. The possibility of an accident of a different type than any evaluated previously in the SAR was not created. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The margin of safety is defined for this system by bounding criteria that are contained in the SAR and are not affected by these changes.

SE 99-027 Removal Of Offgas Automatic Isolation On Post-treat Alarms And Revision Of Operability Requirements In Offsite Dose Assessment Manual (ODAM)

Description and Basis of Change

By letter dated May 16, 1974, from General Electric (GE), justification was provided to allow the Offgas Isolation Valve to fail open on loss of air to prevent the possibility of an inadvertent plant shut down. The same arguments could be used to justify making the valve fail open on loss of control power. In fact, closing this valve in most circumstances would be counterproductive because the DAEC Offgas System can retain only a limited amount of gaseous waste. If the Main Steam Isolation Valves (MSIVs) failed to isolate in the event of gross fuel failure, this gaseous waste would otherwise ultimately be released via the Turbine Building exhaust that monitors but does not treat gaseous waste. Therefore, under most scenarios closing the Offgas Isolation Valve would likely result in larger releases than if the valve were allowed to simply remain open.

This change, which eliminated the automatic isolation feature of the

Offgas post-treat radiation monitor HI-HI-HI, DOWNSCALE, or INOP alarm, extended the ODAM LCO for inoperable post-treat radiation monitors and revised the UFSAR to reflect these equipment changes had no effect on the probability of an accident evaluated previously in the SAR. The elimination of the automatic isolation of the Offgas Isolation Valve did not affect the operation of the Standby Gas Treatment System or the Offgas stack fans. Extending the ODAM LCOs for inoperable Offgas post-treat radiation monitors from 72 hours to 30 days, provided the Offgas charcoal beds are not bypassed and the Offgas noble gas radiation monitors are operable, provides an allowable out of service time that is consistent with the restrictions and requirements of other radiation monitors in the ODAM.

Safety Evaluation Summary

The Offgas System is not an initiator of any accidents evaluated in the SAR. The Offgas System does not perform any nuclear safety-related activity and is not used to prevent any accident. Therefore, this activity did not increase the probability of any accident. The removal of the Offgas automatic isolation signal on HI-HI-HI, DOWNSCALE, or INOP post-treat radiation monitor alarms did not increase the consequences of an accident previously evaluated in the SAR. This change did not affect the operation of other plant systems designed to mitigate the consequences of previously evaluated accidents. In most cases, allowing this valve to remain open will actually reduce net releases and provide additional operator flexibility in dealing with plant conditions. Increasing the LCO for both post-treat radiation monitors being inoperable had no impact on the consequences of an accident previously evaluated. The effective LCO for both post-treat radiation monitors being inoperable was changed to a length consistent with other inoperable radiation monitors. The same caveats (charcoal bed not in bypass and the Offgas stack noble gas activity operable) apply while the monitors are inoperable. In addition, other monitors are available to monitor Offgas effluents such as KAMAN 10. Therefore, this activity did not increase the consequences of any accident. The removal of the isolation reduces the probability of a spurious isolation of the Offgas System that could result in the need to initiate a manual scram at power due to the loss of condenser vacuum. The Offgas System does not perform any nuclear safety function; therefore the removal of this automatic isolation function and increasing the allowable LCO for inoperable Offgas post-treat radiation monitors did not increase the probability of a malfunction of equipment important to safety. The removal of the isolation of the Offgas Isolation Valve on HI-HI-HI, DOWNSCALE, or INOP Offgas post-treat radiation monitor alarms did not affect the ability to manually isolate the Offgas System in the event of an indication of a large release (HI-HI-HI, DOWNSCALE, or INOP) alarm. The original GE Offgas System isolation was designed with a 15 minute time delay to allow operators to take compensatory measures in the event of an alarm. The DAEC is designed for a substantially longer holdup time and per GE, this valve is not required to be present. Therefore, it is not possible for these changes to increase the consequences of a malfunction of equipment important safety. All other features and interlocks associated with the Offgas Isolation Valve and the Offgas post-treat radiation monitor remain unchanged. This change did not introduce any new failure modes. The valve remains available for remote operation from the Control Room in the event circumstances make closure of the valve desirable. The changes to the ODAM did not affect any new or previously analyzed accident scenarios. Therefore, this change did not create the possibility of an accident of a different type than evaluated previously, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. All operability requirements associated with the Offgas post-treat radiation monitors are located in the ODAM. Therefore, the margins of safety as defined in the basis for the Technical Specifications were not reduced.

SE 99-028 **ECP 1621 – Second Stage Reheat Subsystem Modification**

Description and Basis of Change

This ECP modified the piping configuration of the second stage reheat subsystem by adding flanges, valves, and modifying the vent system. This configuration changed the plant as shown in UFSAR figures. The basis for this modification is to: maintain the second stage reheat subsystem configuration as there is sufficient potential for beneficial plant use of the subsystem in the future; provide a more positive means of isolation of possible leakage sources that could reduce plant electrical output; improve the vent system to minimize the potential for hydrogen buildup; provide a means to process second stage reheat tube leakage. Portions of the piping that will be modified have different functions. The functions performed by portions of the piping include: main steam supply to second stage reheat; main steam drain from second stage reheat; Main Steam Isolation Valve Leakage Treatment System (MSIV-LTS); extraction steam drain from second stage reheat; scavenging steam supply from second stage reheat to first stage reheat; venting of second stage reheat. The piping as modified meets all design and seismic requirements. The NSOA was unaffected by this modification. This modification did not affect the design capability of the MSIV-LTS or power conversion system. Since the second stages of the moisture separator reheaters are not typically inservice, the use of line blinds as a passive pressure boundary does not usurp the active function of any components.

Safety Evaluation Summary

The modification has been evaluated and the probability of occurrence or consequences of an accident previously evaluated are not increased. The equipment important to safety affected by the modification has been evaluated and the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated was not increased. This modification did not create the possibility of a different type of accident or malfunction of equipment important to safety than previously evaluated. The margin of safety for the primary containment was not affected by this modification.

SE 99-029 Temporary Modification Of 'A' Recirculation Pump Discharge Valve

Description and Basis of Change

This temporary modification eliminated a short that existed from the power supply wiring that supplies the 'A' Recirculation Pump Discharge Valve position indication input to the 'A' Recirculation Pump 20% runback logic. A jumper was installed around the contact that supplies one of the two inputs that feed the 'A' Recirculation Pump 20% runback logic, i.e. the 'A' Recirculation Pump Discharge Valve position indication input to the 'A' Recirculation Pump 20% runback logic. The jumper allowed the 20% runback feature of total feedwater flow less than or equal to 20% for 15 seconds to remain functional.

Safety Evaluation Summary

This change did not increase the probability of occurrence of an accident evaluated previously in the SAR because the 20% runback circuitry is a control system not required for safety. The purpose of the 20% runback feature is only for Recirculation Pump protection. It prevents operation of the pump with the discharge valve closed so as not to damage the thrust bearing. The 20% Recirculation Pump runback circuitry is not an accident initiator per the SAR and therefore, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The consequences of an accident evaluated previously in the SAR were not increased. The Reactor Recirculation System is not required to operate after a design basis accident. 10 CFR 100 limits were not exceeded. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR. A warning tag was hung on the handswitch for the 'A' Recirculation Pump Discharge Valve alerting the operators that the position switch contact for the valve input into the Recirculation Pump 20% runback logic was unavailable and if the valve position indication did not indicate full open, to ensure appropriate action was taken to trip the 'A' Recirculation Pump. Also, the associated annunciator procedures were revised to provide the operators a reminder that the position switch input for the 'A' Recirculation Pump Discharge Valve into the 20% runback logic was not functional and to take appropriate action if necessary. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. This activity did not create the possibility of an accident of a different type than any previously evaluated in the SAR. No new failure modes were created that could cause the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. This change did not reduce the margin of safety as defined in the basis for any Technical Specification because the detail of the 20% runback is not mentioned in the Technical Specification. The minimum speed runback is not a required safety action for any analyzed event, therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-032 Temperature Elements Installed Downstream Of Main Steam Bypass Valves

Description and Basis of Change

The DAEC Megawatt output was lower than the original design value by about 3.5%. A possible cause was steam leakage through the Main Steam Bypass Valves (BPVs). The temperature of the BPVs were monitored too close to the valves, and therefore indicated a high temperature, even when not leaking. EMAs were used to install two temperature elements on the main steam lines located down stream of the BPVs. The temperature elements are used to provide additional information to determine position/leakage of the BPVs. The new elements are far enough away from the BPVs to ensure radiant heat from the valves will not cause erroneous temperature indications. By positively identifying the position of the BPVs, the plant can run more efficiently. This modification prevents loss of Megawatts due to unknown leakage past the BPVs.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The new temperature elements are the same manufacturer and type as the other temperature elements already installed. The new temperature elements cannot initiate an accident previously evaluated in the SAR. The consequences of an accident evaluated previously in the SAR were not increased. The installation of the temperature elements had no effect on any fission product barriers.

The new temperature elements do not function to mitigate the consequences of an accident previously evaluated in the SAR. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not increased. This modification did not directly nor indirectly affect equipment important to safety. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased, and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. The temperature elements are used for indication only and there is no association with automatic functions. The temperature elements can not cause or prevent an accident condition. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The margin of safety as defined in the basis for any Technical Specification was not reduced. This modification did not change any set points dealing with safety limits or limiting safety system settings. Also, there was no modification to any fission product barriers such as fuel cladding, Reactor Coolant Pressure Boundary, or Primary or Secondary Containment.

SE 99-034 Control Building Computer Room/Secondary Alarm Station (SAS) HVAC Control Panel Changes

Description and Basis of Change

An EMA changed the Computer Room/SAS HVAC Control Panel by removing nuisance alarms associated with the Computer Room/SAS HVAC System. These alarms included; Computer Room Humidity High alarm set at 60% relative humidity, Computer Room Humidity Low alarm set at 40% relative humidity, Computer Room Air Flow Low alarm set at 1 inch water gauge, Computer Room Air High Temperature alarm set at 80°F, and Computer Room Inlet Air Filter Dirty alarm set at 0.75 inch water gauge. The Computer Room/SAS has undergone many changes in the last few years. The SAS has been moved to a new location, and the majority of the plant process computer has been moved to the Data Acquisition Center. The original design of the Computer Room/SAS HVAC System maintained strict temperature and humidity limits to optimize the Plant Process Computer's performance by removal of the heat dissipated from that equipment. The removal of the Plant Process Computer makes the strict environmental temperature and humidity requirements unnecessary. The panels located in this area presently contain power supplies, terminations and fiber optic equipment. This equipment is not as environmentally sensitive and is rated for temperatures in excess of 120°F and humidity levels up to 95% non-condensing. Presently the Air Conditioning System is operated as necessary based upon personal comfort and not equipment operability. Additionally the former Computer Room/SAS Ventilation System communicates with the Control Room Ventilation System. The control room environmental conditions now control the conditions in the computer room.

Safety Evaluation Summary

These changes did not initiate, or increase the probability of occurrence of any accident or transient evaluated by the SAR. This equipment is powered by a non-vital bus and is not required to safely shutdown the plant. This equipment is not required to mitigate any accident or transient evaluated in the SAR. The Computer Room HVAC System is quality level 4 and non-seismic. For accident conditions the HVAC System is not required for safety. This equipment is not safety related. The switch setpoints altered by this modification cannot in themselves cause an accident. The Computer Room/SAS HVAC System is part of the Control Building Ventilation System. The design basis of this system is for habitability, radiation control and equipment operability. The operation of this equipment does not directly or indirectly affect the margin of safety of any safety limit or limiting safety system settings. The overall system operation is made more reliable by eliminating unnecessary alarms. Therefore, this activity did not reduce margin of safety as defined in the basis of any Technical Specification.

SE 99-043 Tool Crib, Break Area And Office In The Turbine Building

Description and Basis of Change

This EMA provided a structure on the south turbine deck, which contains a tool crib, craft break area, and an office area suitable for supervisory staff, procedure references and drawings. The structure will mainly be used during refueling outages. This structure is anchored to the floor on the southeast corner of the south turbine operation deck. The three-story building is approximately 12 feet-six inches wide, 48 feet in length and 26 feet in height. There were no safety related structures, systems or components affected by this structure.

Safety Evaluation Summary

This modification did not change the probability of an accident evaluated previously in the SAR. The turbine building operating floor is not required to mitigate any credible accident under any plant condition. This modification did not affect the ability of the turbine operating floor to perform its function, nor did the installation impair any of the other systems from performing their function. Therefore, this activity did not increase the consequences of an accident as evaluated in the SAR. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not increased. The turbine operating floor with the structure installed does not impair any systems from performing their intended functions because the turbine operating floor is not seismically designed. Therefore, this activity did not adversely affect the equipment in any systems including the systems related to safety. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The structure is in a non-safety related area. Also, the consequences of an equipment malfunction are not more severe as a result of the installation of the structure because the proposed structure does not affect equipment in any systems related to safety. This modification did not change the way the equipment on the turbine south operating deck is operated. The function of the turbine operating deck is not required to mitigate any credible accident under any plant condition. This activity did not create the possibility of an accident of a different type than any previously evaluated in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR was not created. The installation of this modification had no impact on the margin of safety as defined in the basis for any Technical Specification. The Turbine building is not described in the DAEC Technical Specification.

SE 99-047 Installation Of Temporary Jumper To Override Low Level Trip Of 'D' Well Hypochlorite Injection Pump

Description and Basis of Change

The level instrumentation associated with the Well Water Chlorine Injection Hypochlorite Storage Tank was inoperable. In order to ensure the continued efficient operation of the Drywell coolers, this activity temporarily jumpered the Sodium Hypochlorite Pump low suction trip allowing the pump to run even if a low tank level existed. The probability of a loss of the suction source to the pump during the time the level switch was jumpered out was minimal.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The defeat of the low level trip of the 'D' well chlorination pump did not affect any of the inputs considered in the accidents analyzed in the UFSAR or the NSOA. This activity did not increase the consequences of an accident evaluated previously in the SAR. There was no increase in the radiological consequences of any previously analyzed UFSAR accident. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not increased. This change had no impact on systems, structures, or components important to safety. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. There are no credible scenarios where the failure of this system in a non-conservative direction could increase the consequences of a malfunction of equipment important to safety, evaluated previously in the SAR. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The risk of plant shutdown due to the inadequacy of the Drywell cooler capacity could be reduced by this change. The margin of safety as defined in the basis for any Technical Specification was not reduced. There are no Technical Specifications associated with sodium hypochlorite injection into the Well Water System.

SE 99-048 Creation Of Three New Computer Points

Description and Basis of Change

An EMA created two computer screens to monitor area radiation levels in various areas of the plant. One screen displays area radiation levels required to be monitored during Emergency Operating Procedure (EOP)-3 while the other screen displays all thirty points on the plant process computer (PPC). For this, three new computer points were created to monitor the radiation levels in Jungle Room, RWCU Heat Exchanger Room and RWCU Pump Room. None of the thirty computer inputs provide a signal to any circuit important to safety.

Safety Evaluation Summary

This activity involved only a software and drawing change. No new components are introduced into or removed from the ARM system. The function of the Area Radiation Monitoring (ARM) System before and after the implementation of this modification remained unaffected. Therefore, the probability of occurrence of an accident evaluated in the SAR was not increased. The new and existing PPC displays cannot cause an accident. The ARMs do not provide automatic signals to any circuit, which is required for safe shutdown. Therefore, the consequences of an accident previously evaluated in the SAR were not increased. ARMs do not provide signals to any circuit required for safe shutdown. This activity involved a software and drawing change only and did not introduce any new failure modes. Therefore, the probability of a malfunction of equipment important to safety was not increased. The ARMs involved do not provide signals to control any equipment important to safety to mitigate the consequences of an accident. No new failures can be attributed to the creation of new PPC points for display on the computer. Thus, the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. The display by itself can not create an accident. No accident mitigating actions are based on the information provided by the display. Hence, the possibility of an accident of a different type or the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. There are no Technical Specifications for the ARM System.

SE 99-051 Reload 16/Cycle 17 Core Loading Pattern

Description and Basis of Change

This change involves the replacement of GE 10 fuel with GE 12 fuel. The transition involves two cycles, such that approximately one third of the fuel was replaced during RFO 16. The change was evaluated with respect to shuffling and operation in order to support the activities described in Core Modification Package (CMP)-16. The bases for the change are analyses, performed by General Electric and reviewed for acceptance by DAEC, that include evaluation of the GE 12 design generically, specifically applied to DAEC and specifically focused on the mixture of GE 12 and GE 10 fuel to evaluate the behavior of the first transition core.

Safety Evaluation Summary

The probability of an accident evaluated previously in the SAR is not increased by the shuffle of fuel assemblies or operation of this fuel during Cycle 17. The fuel is licensed by the NRC via GE's topical report (NEDE-24011-P-A) which includes, by reference, GE Fuel Bundle Designs (NEDE 32417P and NEDE-31152P). No changes in fuel handling practices or equipment that would affect the bundle drop accident (i.e., the Refuel Accident in the UFSAR) were made with this core modification. Additionally, the Nuclear Fuel System as described in the NSOA does not perform any safety action for transients, accidents, or special events. The consequences of an accident previously evaluated in the SAR were not increased by the shuffle of fuel assemblies or operation of this fuel during Cycle 17. The core loading pattern has been evaluated in the Supplemental Reload Licensing Report (SRLR) to demonstrate compliance with the licensing basis as described in the SAR. Although the U235 enrichment has been increased in the bundle design for Cycle 17, that enrichment is bounded by the current analysis and does not increase the consequences of an accident. The dropped bundle analysis has been analyzed and is bounded by the existing UFSAR analysis. Although a new fuel type was introduced, DAEC specific analyses have been performed which demonstrate that the consequences of an accident have not been increased both for equilibrium GE12 cores and for transition GE10/GE12 cores. The probability of occurrence of a malfunction of equipment important to safety as evaluated previously in the SAR is not increased. The GE-12 and GE-10 fuel loaded in this reload have been demonstrated (as documented in GESTAR-II) to meet all acceptance criteria for fuel designs and is manufactured/ constructed under an NRC-approved quality assurance program. The probability of a failure of the fuel cladding when operated in accordance with the fuel thermal limits is not increased from that previously evaluated. The probability of a vessel overpressure and subsequent overstressing of the reactor coolant pressure boundary is not increased from that previously evaluated. This core modification did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The modification of the loading pattern was evaluated to ensure that the fuel performs its intended function during a postulated malfunction of equipment. The results of the transient and accident analysis presented in the SRLR demonstrate that the fuel cladding integrity is maintained when the thermal limits are met. The SRLR provides the operating limits that will be observed to ensure fuel cladding integrity is maintained. There is, therefore, no increase to the consequences of a malfunction of equipment as evaluated in the SAR. This core modification did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The fuel design criteria for the fuel have been shown to be satisfied for GE-12 and GE-10 fuel. Additionally, the only events which may be associated directly with fuel loading are the fuel loading error (mislocated bundle) or misoriented (rotated) bundle error, both of which are analyzed in the SRLR. No additional accident type was introduced with this core modification. This core modification did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The design of the GE-12 fuel is evolutionary with respect to the GE-10 fuel that DAEC has used successfully. The design of GE-12 has been compared to GE-10 and other fuel types to ensure that the form, fit and function are equivalent to GE-10 fuel. Operating experience with GE-12 fuel has been evaluated for any type of defect or degradation that could be different than what was previously evaluated and no such defects or degradations exist. The GE-10 fuel component of Reload 16 is physically the same as that loaded previously at the DAEC. The performance of the fuel during all modes of operation, as described in the NSOA, has been demonstrated in the SRLR. Therefore, the possibility of a malfunction of equipment important to safety of a different type than previously evaluated was not introduced during this reload. This reload did not reduce the margin of safety as defined in the DAEC Technical Specifications as long as the Safety Limit Minimum Critical Power Ratio (MCPR) was changed, prior to operation in Cycle 17. (Which it was). The margin to safety is maintained for all Technical Specification parameters as long as the fuel thermal limits are met. The limits described in 10 CFR 50.46 (maximum cladding temperature, maximum cladding oxidation, maximum hydrogen generation, maintenance of a coolable geometry) are maintained for Cycle 17 operation. Similarly, Technical Specifications, as appropriate, are met for activities associated with reload activities. Therefore, the activities associated with CMP-16 did not reduce the margin to safety as defined in the basis for Technical Specifications.

SE 99-052 Installation Of Test Connections For Turbine Performance Test

Description and Basis of Change

An ECP along with an EMA installed the plant instrument changes as necessary to support the Steam Cycle Performance Test. The reason for performing this test was based on an Electrical Power Research Institute (EPRI) Thermal Performance Peer Assessment conducted at the Duane Arnold Energy Center (DAEC) in June 1999. This assessment recommended that a "limited" performance test be conducted at the Duane Arnold Energy Center. This recommendation was based on a need to improve the plant heat balance model in order to verify the current plant performance level.

Four basket tip probes were added to each Low Pressure (LP) turbine exhaust. The probes are attached to a support that was welded to the exhaust neck approximately six inches above the expansion joint. The piping attached to the probes exits via bulkhead fittings that are welded in the Herzog style exhaust hoods of the LP turbines. The four probes recommended by the testing contractor provide the basis for comparison of the turbines' performance before and after turbine cycle changes.

An orifice flange in the four inch main steam line downstream of the A/B Main Steam Line Supply Valves to Offgas and Steam Jet Air Ejectors was added. This orifice flange was installed before the line splits to go to the Steam Jet Air Ejectors and Offgas System. Each half of the flange has a root isolation valve then tubing running to an existing (abandoned) tubing penetration with an instrument valve and cap.

A tap was added as close as practicable to the High Pressure (HP) Turbine to measure HP turbine fourth stage extraction pressure. Only one root isolation valve was required.

During RFO 16, an attempt was made to determine which LP turbine ports were wanted and where taps should be after the turbine is disassembled. The connections to these ports were added by the appropriate means as determined to be required for accurate test results.

A test connection consisting of tubing tees, valves and caps upstream of the plant equalizing manifold for the Condensate Total Flow Transmitter was added. The addition of these tees and valves did not affect the function of this instrument. The attachment of testing equipment to these connections may affect the flow transmitter, which is addressed by a DAEC Special Test Procedure.

A test connection consisting of tubing tees, valves and caps upstream of the plant equalizing manifold for the pressure differential indicator for the Feedwater Loop A Flow element was added. The addition of these tees and valves did not affect the function of this instrument. The attachment of testing equipment to these connections may affect the pressure differential indicator, and this is addressed by a DAEC Special Test Procedure.

A test connection was added, consisting of tubing tees, valves and caps upstream of the plant equalizing manifold for the pressure differential indicator for the Feedwater Loop B Flow element. The addition of these tees and valves did not affect the function of this instrument. The attachment of testing equipment to this connection may affect the pressure differential pressure indicator, and this is addressed by a DAEC Special Test Procedure.

A test connection consisting tubing tees, valves and caps was installed upstream of the plant equalizing manifold for the pressure differential transmitter for the Moisture Separator Reheater (MSR)-1E-18A 1st stage reheat steam flow. The addition of these tees and valves did not affect the function of this instrument. The attachment of testing equipment to this connection may affect the pressure differential transmitter, and this is addressed by a DAEC Special Test Procedure.

A test connection consisting of tubing tees, valves and caps was installed upstream of the plant equalizing manifold for the pressure differential transmitter for the MSR-1E-18B 1st stage reheat steam flow. The addition of these tees and valves did not affect the function of this instrument. The attachment of testing equipment to this connection may affect the pressure differential transmitter, and this is addressed by a DAEC Special test procedure.

Safety Evaluation Summary

The NSOA review identified no accident, transient, or special event previously evaluated in the SAR that may be affected by this modification. In addition, the probability of occurrence of an accident previously analyzed is not increased by the modifications. The modified design uses applicable piping standards and meets or exceeds acceptable standards. The consequences of an accident are unchanged since this modification maintains the pressure boundary and meets the appropriate seismic requirements. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The consequences of a malfunction of equipment are unchanged as the modified configuration to install the necessary test equipment has only a passive function. The possibility of an accident of a different type and the possibility of a malfunction of equipment of a different type was not created since the modified configuration has only a passive function and is designed to applicable standards. The margin of safety as defined in the basis for any Technical Specification was not reduced. The piping modifications have no impact on safety.

SE 99-054 Temporary Modification - Radwaste Floor Drain Collector Tank Connection For Torus Desludge

Description and Basis of Change

The intent of this temporary modification was to allow performance of torus water cleaning/desludging operation without the use of an underwater filtration process and the need for divers. Torus water was pumped from the torus to the nearby Radwaste Floor Drain Collector Tank. A temporary manway lid which would adopt a hose was installed on the tank. The other side of the hose was attached to a pump used for this cleaning operation. This activity was performed during the refueling outage and the tank was restored back to its original configuration upon completion of the torus water cleaning/desludging operation. This temporary modification met the design, safety, and operability of the liquid radwaste system and the intents of the DBD, NSOA, and UFSAR.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR because the temporary modification met the design, materials, and construction standards as defined in USNRC Reg. Guide 1.143 and the DAEC UFSAR. The equipment introduced by this modification was not an initiator of any accident. This activity did not increase the consequences of an accident evaluated previously in the SAR because the temporary manway lid kept performing its design functions under expected plant and environmental conditions. An inadvertent spill, leak or hose break would have been limited within the confines of the Reactor Building, which would retain and return it to the Radwaste System for additional processing. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR because the designated components were given appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it would operate. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR because the protection against accidental discharge is provided by instrumentation for the detection and alarm of abnormal conditions and procedural controls. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The tank is located in Reactor Building, which is capable of handling a major leak in the largest tank without permitting significant quantities of liquid to escape offsite. The temporary lid, hose, and fittings were suitable for the intended function. The scenario of hose break is no different than the pipe break and this type of accident has been evaluated previously in the SAR. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification because the Technical Specifications do not specify any margin of safety for the Liquid Radwaste System or its components. Also, no surveillance tests are specified for the existing components or the components added by this modification. The temporary manway lid on the tank was installed in accordance with a Radwaste Handling Procedure to maintain secondary containment.

SE 99-057 ECP 1625 - Main Steam Line Drain System Modifications

Description and Basis of Change

ECP 1625 modified the Main Steam Line (MSL) Drain System to ensure that both the inboard and outboard main steam line drain subsystems provide adequate drainage when required. Also, the outboard main steam line drain subsystem was modified to eliminate the flanged joints in the system that were a source of leakage. An additional benefit of the modification is that the flow through the system is reduced during normal plant operation, which provides more steam for power generation.

This ECP modified the piping configuration of the MSL Drain System by relocating and resizing the Low Point Drain Flow Orifice to each outboard drain line. Also a flow orifice was installed as a bypass to the Main Steam

Drain Isolation To Condenser Control Valve and a flow orifice was installed on the inboard drain subsystem to ensure that a sufficient pressure drop occurs prior to the location where the subsystems join. This configuration changed the plant as shown in an UFSAR figure.

The piping as modified meets all design and seismic requirements. The NSOA is unaffected by this modification. This modification did not affect the design capability of the Main Steam Isolation Valve Leakage Treatment System (MSIV-LTS) or power conversion system.

Safety Evaluation Summary

This modification was evaluated and the probability of occurrence or consequences of an accident previously evaluated were not increased. The pressure boundary design requirements and appropriate seismic requirements are met. The radiological calculation used to demonstrate compliance with 10 CFR 100 and 10 CFR 50 Appendix A GDC 19 limits are unchanged with this modification. The equipment important to safety affected by this modification has been evaluated and the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated were not increased. The modified configuration has only a passive safety function and the components in the modified configuration have been designed to applicable piping standards. This modification did not create the possibility of a different type of accident or malfunction of equipment important to safety than previously evaluated. The margin of safety for primary containment was not reduced as the modification did not affect the processing capability of the MSIV-LTS.

SE 99-059 Residual Heat Removal (RHR) System Jumpers

Description and Basis of Change

The purpose of this activity is to install jumpers in the RHR logic to provide adequate isolation capability for the Inboard Low Pressure Coolant Injection (LPCI) Valves during the performance of surveillance testing, relay testing or for any situation where one RHR Loop of Shutdown Cooling is in service while the other RHR Loop's logic is de-energized or inoperable.

This activity installs jumpers for two situations:

(1) Jumpers around the Outboard Shutdown Cooling Suction Valve position indication in the "A" RHR Logic that controls the closure of the "A" RHR Loop Inboard LPCI Injection Valve. This will allow the "A" RHR Loop Inboard LPCI Injection Valve to close as required

without input from the "B" RHR Logic. This jumper would be installed when the "A" loop of RHR is in Shutdown Cooling.

(2) Jumpers around the Inboard Shutdown Cooling Suction Valve position indication in the "B" RHR Logic which controls the closure of the "B" RHR Loop Inboard LPCI Injection Valve. This will allow the "B" RHR Loop Inboard LPCI Injection Valve to close as required without input from the "A" RHR Logic. This jumper would be installed when the "B" loop of RHR is in Shutdown Cooling.

These temporary modifications are installed when one of the RHR loops is in Shutdown Cooling while the other RHR Loop's logic is de-energized or inoperable. These temporary modifications are removed when the other side's logic is re-energized or made operable. Under no circumstances would both jumpers be allowed to be installed simultaneously.

Safety Evaluation Summary

The probability of occurrence of an accident previously evaluated in the SAR is not increased because the isolation capability for the Inboard LPCI Inject Valves are not accident initiators per the SAR. The consequences of an accident previously evaluated in the SAR are not increased because the isolation capability for the inboard LPCI inject valves is being maintained and there is no adverse effect on the other modes of the RHR System. Thus, there is no effect on mitigating the consequences of an accident or on the fission product barriers or dose consequences. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. No other RHR System component could be adversely affected due to maintaining this capability. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased. Any previous SAR evaluation of a malfunction of the Inboard LPCI Inject Valves to close is not adversely impacted. The possibility for an accident of a different type than any evaluated previously in the SAR is not created, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR is not created. The isolation capability of the Inboard LPCI Inject Valves is being maintained and there is no adverse effect on the other modes or components of the RHR System. Based on a review of Technical Specifications, Technical Specification Bases, UFSAR and NSOA, the margin of safety is not reduced since there is no margin of safety defined which could be affected because the isolation capability of the Inboard LPCI Inject Valves is being maintained.

SE 99-062 **ECP 1627 - Installation of Weld Overlays On Recirculation Riser Nozzle To Safe-end Welds**

Description and Basis of Change

While performing ultrasonic examinations of recirculation riser nozzle to safe-end welds, several linear indications were found. Weld overlays were installed as a method of repairing the indications. Qualified personnel from Welding Services Incorporated (WSI) performed the repair activities. Weld repair designs were developed by Structural Integrity Associates (SIA) and IES Engineering personnel. IES Engineering personnel reviewed and approved both the SIA and in-house weld repair designs for use. SIA has performed the design activities of the repairs in accordance with their Quality Assurance (QA) program. IES Utilities has reviewed and approved the QA programs of both WSI and SIA. The weld overlays were installed in accordance with DAEC's design control process via an Engineering Change Package (ECP).

The overlays used Alloy-52, which is resistant to Intergranular Stress Corrosion Cracking and is compatible with existing piping and weld materials. The process of applying the overlay has been shown to not weaken the existing piping or weld and puts the outer diameter of the piping into compression to stop the growth of the indications.

Safety Evaluation Summary

The accident mitigating requirements of the Recirculation System are not adversely affected by this installation. Installation of the weld overlays did not have any affect on the remaining Recirculation System components such as the pumps and flow control circuitry. Hence, installation of the weld overlays did not increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR. Installation of the weld overlays did not increase the probability of malfunction of the Recirculation System pumps or flow control circuitry. The probability of occurrence of a malfunction of equipment important to safety was not increased. The malfunction of equipment important to safety applicable to this installation is a recirculation pipe break, up to and including a circumferential break. This is a bounding event and installation of the overlays cannot increase the consequences of this event previously evaluated in the SAR. Installation of full structural overlays cannot introduce any new failure modes of the Recirculation System piping. Therefore, a different type of accident than any evaluated previously in the SAR was not created. Applying weld overlays externally to the recirculation piping will not have any adverse affect on the Recirculation System function. Therefore, the installation of the weld overlays did not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. The objective of the Reactor Recirculation System is to provide a variable rate of coolant flow to the reactor core so that a proper thermal margin is maintained during normal reactor operation. Installation of the weld overlays did not adversely affect the ability of the system to achieve this function since the installation did not affect the Recirculation System pumps or flow control circuitry. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-064 Main Condenser Mechanical Vacuum Pump EMA

Description and Basis of Change

The enhancement to the Mechanical Vacuum Pump involved replacing two drain plugs on the bottom of the pump with two isolation valves and associated piping and fittings to increase the ease of draining the pump after it has been used. This enhancement did not impact the performance of the Mechanical Vacuum Pump and therefore did not impact any of the systems or components which interface with the Mechanical Vacuum Pump.

Safety Evaluation Summary

The installation of the two isolation valves did not increase the possibility of occurrence of an accident evaluated previously in the SAR. The Main Condenser Air Removal System can not initiate an accident and is not a safety related system. The consequences of an accident evaluated previously in the SAR were not increased. The Mechanical Vacuum Pump does not perform any accident mitigating function. The Mechanical Vacuum Pump does discharge to the Offgas Stack, which is required to mitigate an accident, however the performance of the Mechanical Vacuum Pump can not impact the function or performance of the Stack. The probability of occurrence of a malfunction of equipment important to safety was not increased. Furthermore, the Mechanical Vacuum Pump can not increase the probability of the Offgas failing. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. The Main Condenser Air Removal System is not safety related and can not affect any safety related systems or components. The possibility of an accident of a different type than any evaluated previously in the SAR was not created. The Mechanical Vacuum Pump can not initiate an accident, and the change in the way the pump is drained does not change this fact. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created, and the margin of safety as defined in the basis for any Technical Specification was not reduced. The performance of the Mechanical Vacuum Pump has no impact on systems or components that maintain the margin of safety at DAEC.

SE 99-065 Temporary Modification For "B" Reactor Feedwater Pump (RFP)

Description and Basis of Change

This temporary modification removes the high vibration alarm signal for the "B" RFP. This temporary modification removes the wire connecting the vibration switch/alarm to the control room. The vibration switch was falsely reading high, with the pump operating satisfactorily. This vibration alarm/signal does not trip the pump and is an alarm function only. A review of the NSOA shows the Feedwater Pumps are not used to maintain adequate water level under accident conditions. This alarm is only used to inform the operators that they should investigate the higher-than-normal levels of vibration for the "B" RFP.

Safety Evaluation Summary

Since the pump is not used to mitigate any accident, the removal of this alarm can not increase the consequences of an accident evaluated previously. This temporary modification does not affect any equipment important to safety. The probability of occurrence of an accident evaluated previously in the SAR is not increased. This modification only removes an alarm function and does not affect performance of the "B" RFP. The consequences of a malfunction of equipment important to safety are not increased. The pump is the only component affected by this modification. The pump continues to operate as before and the pump is not used to mitigate any accident. This temporary modification does not create a different type accident or increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously. Complete failure of the vibration alarm or the failure of the alarm to indicate in the control room will only affect the Feedwater System, which is already evaluated. Additionally, this change does not create a malfunction of equipment important to safety of a different type than previously evaluated. This modification does not change the safety margin as defined in the basis for any Technical Specification because this pump and the vibration alarm is not called out in Technical Specifications.

Section B - Procedure/Miscellaneous Changes

This section contains brief descriptions of Procedure/Miscellaneous changes completed during the period October 1, 1998 through February 29, 2000, and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR Section 50.59(b). All changes were reviewed against 10 CFR 50.59 by the Duane Arnold Energy Center (DAEC) Operations Committee. None of the changes involved unreviewed safety questions.

SE 98-002 Auxiliary Heating System Boiler And Main Loop Piping And Instrument Diagram (P&ID) Revision

Description and Basis of Change

Various vent and drain valves on Auxiliary Heating System Boiler and Main Loop P&ID were shown normally open while they are required to be normally closed per the Operating Instruction (OI). While reviewing to resolve this problem, another 33 valves were found to be in positions different from the positions specified on the Valve Lineups included in the OI. The major conflict between the P&ID and OI stemmed from the fact that the P&ID assumes the boiler is in service while the OI assumes the heating loop is shutdown. With the completion of this revision to the P&ID and the revision to the OI, the valve positions shown on the P&ID and OI Valve Lineups agree except as specified by "Note 1" of the P&ID. Note 1 lists those valves whose position change when the boiler is not in service.

An unnecessary personnel hazard was identified associated with the daily water chemistry samples obtained via the boiler surface blowdown line during routine operations of the Auxiliary Boiler. The inboard isolation valve required a ladder for access and the piping is very hot. Leaving this valve in the normally open position would eliminate the hazard, but would also reduce the surface blowdown line to single valve protection. In reviewing "Iowa Statute and Administrative Rules for Boilers and Unfired Pressure Vessels" (Code of Iowa Chapter 89) and ASME, Section VII, "Recommended Guidelines for the Care of Power Boilers", no requirements to maintain double isolation were found. Considering the size of the line (1 inch) and the fact that it discharges directly to the Blowdown Separator, single valve isolation during operation poses no operational or personnel safety concerns. The Auxiliary Boiler is infrequently operated, normally during reactor shutdown in cold weather to provide heating steam and during refuel outages for testing and inerting. Isolation will be maintained through the outboard isolation valve. The P&ID has been revised to show the inboard isolation valve in the normally open position. The OI and a Chemistry Procedure were also revised to incorporate this change.

There also was confusion over the drain valves on the removable spool pieces for the Auxiliary Boiler Steam Supply to the Liquid Nitrogen Vaporizer and to the HPCI Turbine. These are two separate spool pieces, however, the same valve identification number was assigned to the drain valve for both.

Various other errors were identified during research and field walkdowns, and have been corrected. These included: the location of a flow element was shown incorrectly; an autovent symbol at the Chemical Feed line vent valve should have been a screwed cap; the Chemical Feed Tank vent line was shown connected to the drain line for a Y-strainer; and the legend for seal flushing and cooling water connections should have been revised.

Safety Evaluation Summary

This activity does not increase the probability of occurrence of an accident evaluated previously in the SAR, nor are the consequences of an accident increased. The Auxiliary Boiler System is not an accident initiator, and is not required to mitigate the consequences of any accident evaluated in the SAR. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased, nor are the consequences of such a failure. The subject system is not included in the Nuclear Safety Operational Analysis (NSOA) and none of the affected components have any safety significance. This activity does not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. No equipment important to safety was impacted, the system is not an accident initiator nor is it required to mitigate the consequences of any accident evaluated in the SAR, and the function of the Auxiliary Boiler System remains the same as described in the text of the UFSAR. The margin of safety as defined in the basis for any Technical Specification is not reduced since the Auxiliary Heating System Boiler and Main Loop is not included in the DAEC Technical Specifications and no equipment important to safety is impacted.

SE 98-065 Fire Brigade Training And Equipment

Description and Basis of Change

Changes to specific equipment used by the DAEC fire brigade as well as a change to the fire brigade training requirements that are described in NRC fire protection safety evaluation reports and in the UFSAR were made. Changes related to specific fire brigade equipment enable the brigade to operate more efficiently and safely while reducing equipment maintenance costs. A change in the fire brigade training requirements was made to acknowledge the flexibility existing in current training practices. This safety evaluation was prepared to support an UFSAR Change Request.

The following four items represent the scope of Fire Protection Program commitment changes addressed by this safety evaluation:

- Remove hydrant discharge gate valves, where possible, that are installed on one of two 2½ inch outlets on all yard fire hydrants. Two quarter-turn ball valves located in the brigade hose trailer will provide this flow control function to both hydrant outlets, as necessary. For cases in which hydrant gate valves are permanently attached to the hydrant, making removal impractical, the ball valves may be attached to the end of the gate valves at the time of service.

  Gate valves are required on 2½ inch hydrant outlets to comply with the NRC fire protection safety evaluation report (SER) of 6-1-78. The SER states that a 2½ inch gated valve will be installed on one outlet of each yard hydrant. The origin of this requirement comes from a NRC staff position established after an on-site review of the plant's fire protection program occurring in the late 1970's (12-20-77 G. Lear, USNRC to D. Arnold, Iowa Electric Light & Power (IELP)). Staff position PF.21 called for a 2½ inch gate valve on each of the yard hydrant's outlets. The licensee responded (1-12-78 L. Liu, IELP to G. Lear, USNRC) by providing only one gate valve per fire hydrant, which was the configuration subsequently accepted by the regulator in the SER of 6-1-78.

  Ball valves enable rapid flow isolation in the event a fire hose should become uncontrolled, and they allow for flow adjustment with minimal effort. The valve handle clearly indicates the open or closed position of the valve. The use of two ball valves rather than a single gate valve not only enables the use of both hydrant discharges, but can also be used to supply two hose lines feeding a master stream device recently acquired by the plant.

- Abandon the fire hose houses located adjacent to the yard hydrants in favor of the portable hose trailer. The trailer will be fitted with manual fire fighting equipment and a minimum of 300 feet of 1½ inch fire hose and 600 feet of 2½ inch fire hose. This greatly exceeds the plant's original commitment of 200 feet of 1½ inch fire hose and 250 feet of 2½ inch fire hose on the fire brigade hose trailer documented in letters; 8-29-78 L. Liu, IELP to H. Denton, USNRC and in 2-10-81 T. Ippolito, USNRC to D. Arnold, IELP. The amount of hose, originally committed to, augmented the 50 feet of 2½ inch fire hose already existing in each of the hydrant hose houses, as documented in the NRC fire protection SER of 6-1-78.

This change is acceptable since the fire brigade trailer equipment is maintained in the same fashion as the hose house equipment and the portable hose trailer provides the necessary fire hose and auxiliary equipment needed to fight exposure fires in the yard.

- Remove the hose clamp from the hose trailer. The flow isolation function is better served by the hydrant ball valves located in the hose trailer. The basis for inclusion of the hose clamp is a NRC recommendation documented in correspondence to Iowa Electric Light and Power Company of 12-20-77. This correspondence contained an NRC staff position that a hose cart be provided to augment inadequate fire fighting equipment discovered in the hydrant hose houses during an on-site inspection of the DAEC fire protection program earlier that December. A hose clamp was just one of the items listed for inclusion on the hose cart. Given the use of fast-acting ball valves on hydrant outlets, there is no reason to require a hose clamp in the fire brigade trailer. Ball valves provide a greater degree of safety to fire fighting personnel, therefore, removal of the hose clamp from the hose trailer is acceptable from a plant safety standpoint.

- Change the terms used to describe the frequency of fire brigade quarterly training from "every three months" to "four times per year." This more accurately describes the manner in which training is conducted when dealing with refuel outages and other potential scheduling conflicts. This change allows flexibility in scheduling training and fire drills around major plant events such as refuel outages and still meet the Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance (FRACQA) requirements for the conduct of that training. The expected practice will be to meet the quarterly requirement to the extent practical.

Safety Evaluation Summary

Changes in commitments with respect to fire brigade equipment and fire brigade training requirements do not increase the probability or consequences of accidents already analyzed in the SAR, since fire is not an accident described in the SAR. Neither will changes in such fire brigade related commitments increase the probability or consequences of a malfunction to safety related equipment already analyzed in the SAR, since such changes will not adversely affect the ability of safety related equipment to perform its safety function. The proposed changes in commitments with respect to fire brigade equipment and fire brigade training requirements do not create the possibility of an accident or malfunction of safety related equipment of a type not already analyzed in the SAR. The DAEC fire brigade and equipment used by the fire brigade serve to protect against design basis fires, which is an event evaluated in the plant's Fire Hazards Analysis and not a plant accident evaluated in the SAR. Fire protection equipment is not among the safety related equipment and systems discussed in the SAR and the proposed changes can not adversely affect the safety function of such equipment. Changes in commitments concerning fire brigade equipment and fire brigade training do not affect Technical Specifications in any way, since fire protection is not a part of the plant Technical Specifications.

SE 98-075 Fire Protection P&ID Drawing Update

Description and Basis Of Change

The auto/manual stop switch for the electric fire pump was removed by a previous modification upon the recommendations of the Nuclear Energy Liability Property Insurance Association (NELPIA). However, the Fire Protection System P&ID was not updated at that time. Subsequently, the drawing has been updated. The electric fire pump provides the same water pressure to the fire main as it did previously. The automatic start of the electric fire pump was not affected. Shutdown of the pump can still occur locally by the operator only after investigation into the cause of the pump start.

Safety Evaluation Summary

The fire pumps do not provide a safety related function as stated in the Fire Protection Program Design Basis Document (DBD). This change did not impact the plant's ability to safely shutdown in the event of a fire as required by 10 CFR 50 Appendix R, and regulatory commitments are still being met. This activity did not increase the probability of occurrence of an accident previously evaluated in the SAR and the consequences of an accident evaluated previously in the SAR are not increased. Removing the automatic shutdown function decreases the probability of an occurrence of a malfunction of this equipment. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety. The starting logic for the electric fire pump was not changed.

Providing manual stop only does not increase the consequences of a malfunction of equipment important to safety. Safe shutdown capability in the event of a fire was not affected. The chance of fire main over-pressurization or pump damage was not increased. The design of the system adequately provides this protection. This activity did not create the possibility of an accident of a different type than any previously evaluated in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR was not created. Margins of safety, as found in the Technical Specifications, were not reduced. The Technical Specifications do not contain requirements for the Fire Protection System.

SE 98-080 Changes To Well Water System P&IDs

Description and Basis of Change

Changes were necessary to resolve discrepancies in what existed in the field versus what was depicted in the Well Water System P&IDs. Specifically, UFSAR figure 9.2-1, sheet 2 (BECH-M144<2>), was revised to add the heat trace controls for the A, B, and C Well Water heat tracing located at their respective Well Houses, and instrument isolation valve, V44-0397, was deleted since it did not exist in the field. UFSAR figure 9.2-1, sheet 1 (BECH-M144<1>), was revised to change the normal position of V44-0108, Drywell Cooling Water Drain Valve, from open to closed. The Well Water System is utilized to provide cooling water for all the plant ventilation cooling units, supply potable water for plant requirements, and supply required water for the Makeup Demineralizer System. Discharge from the plant ventilation cooling units is reused for cooling water in the offgas recombiner, offgas glycol refrigeration unit, and the containment nitrogen compressor.

Safety Evaluation Summary

These changes did not increase the probability of occurrence of an accident evaluated previously in the SAR. These changes were necessary to resolve discrepancies between what exists in the field and what was depicted in UFSAR figures. The manner in which the Well Water System operates was not revised in any way. The Well Water system does not initiate or cause any of the accidents previously analyzed in the SAR. These changes did not increase the consequences of an accident evaluated previously in the SAR. The probability of a malfunction is not changed, and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. These changes did not create the possibility of an accident of a different type than any evaluate previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The changes had no safety significance. The changes were made to better reflect the field representation of the non-safety related Well Water System. These changes were only enhancements to the existing plant drawings. These changes did not reduce the margin of safety as defined in the bases for any Technical Specification.

SE 98-093 Revision To Instrument And Service Air Compressors' P&ID

Description and Basis of Change

The detail shown on the Instrument and Service Air Compressors' P&ID for the backup instrument air compressor was improved to include individual components, similar to the main instrument air compressors. The schematic for the compressor's trouble annunciator was also detailed to show the exact contacts supplying the alarm. Changes to the vendor flow diagram eliminated inconsistencies with the vendor manual and the actual compressor configuration. All of these changes reflect the required plant configuration.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident previously evaluated. These changes did not increase the consequences of an accident previously evaluated in the SAR. The affected systems are not required for the mitigation of an accident. These changes did not increase the probability of a malfunction of this equipment nor did it increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This equipment still operates in the same manner as it did previously. The consequences of a malfunction of equipment important to safety was not increased, and the possibility of an accident of a different type than any previously evaluated in the SAR was not increased. The changes made were all part of the original designs of the systems. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. There are no Technical Specifications specifically related to the affected systems. The margin of safety as defined in the basis for the Technical Specifications was not affected.

SE 98-102 Standby Gas Treatment (SBGT) System P&ID Revision

Description and Basis of Change

The purpose of this change was to address the removal of an error in the P&IDs for the SBGT Room Sump, which is part of the Reactor Building

Sump System. The SBGT Room Sump has four inlets from the Fire Water Deluge for the carbon bed filters and four more inlets from the Floor Drain System, but no outlet. This was verified by a plant walkdown. The UFSAR Figure showing the P&ID for the SBGT System erroneously depicted an outlet from the sump, and indicated a continuation of the line to another UFSAR Figure, the P&ID for the Radwaste Sump System. The Radwaste Sump System P&ID does not indicate any piping continued from the SBGT System P&ID. For normal operation, there is no need for a sump outlet. When there is a need to dispose of the sump water, it is done through the Reactor Building Sump System using the plant procedures. The outlet for the sump shown on the SBGT P&ID has been deleted.

Safety Evaluation Summary

The Reactor Building Sump System cannot create an accident. Therefore, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The SBGT Room Sump does not normally receive any water from the drains. If it does receive drainage water, the means for its controlled disposal are in place. This activity did not increase the consequences of an accident evaluated previously in the SAR. The SBGT Room Sump has no equipment that has an active safety function. Therefore, this activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. If the SBGT Room Sump should overflow the water will be collected through the Reactor Building Floor Drain System for further treatment and disposal in a safe and controlled manner by the Liquid Radwaste System. Therefore, this activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. Because the Reactor Building Sump System cannot create an accident, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. No new failure modes were created by this activity since no safety function of any equipment was impacted by this change. Therefore, this activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings or safety limits are defined in the Technical Specifications for the Reactor Building Sump System. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-107 Minimum Acceptable Margin For 125VDC and 250VDC Batteries

Description and Basis of Change

This safety evaluation changes the implied minimum margins for the 125VDC and 250VDC station batteries. Nuclear Regulatory Commission Station Blackout (SBO) Rule Conformance Evaluation dated June 15, 1992 (NRC SBO SER) states that 1D2 and 1D4 are acceptable based on a greater than 20% margin for 1D2 and greater than 80% margin for 1D4. Based on this wording, the implied minimum margins are greater than 20% for 1D2 and greater than 80% for 1D4. These values for margin are based on DAEC's response to a previous submittal in which the staff referenced the recommended margins of 10% to 15% given in IEEE 485. The conclusion of this safety evaluation is that the 10% margin referenced in IEEE 485 is the appropriate minimum margin at the end of the SBO coping period which applies to the 125VDC and 250VDC station batteries.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Battery margin has no affect on the probability of an accident, including Station Blackout. The probability of a Station Blackout is dependent on the AC (Alternating Current) System reliability. This activity did not increase the consequences of an accident evaluated previously in the SAR. The revised battery margin provides sufficient capacity and capability to cope with Station Blackout. The probability of a malfunction was not changed, and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. Revising battery margin does not affect testing methodology and will not affect the plant's ability to detect battery degradation nor will the consequences of malfunction be affected. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The revised margin does not affect the batteries' ability to supply station blackout coping equipment or safety related equipment for a design basis event. This change did not reduce the margin of safety as defined in the basis for any Technical Specification. Battery testing described in the Technical Specifications was not affected by this change. This change did reduce the implied margin as stated in the NRC SBO SER. However; the 80% and 20% margin accepted in the SBO SER is excessively conservative as a minimum margin and the NRC staff, during review of the DAEC submittal, endorsed the margin of 10 to 15% stated in IEEE-485.

SE 98-114 UFSAR Change For Reactor Water Cleanup (RWCU) System Supply Isolation Valves' Closure Times

Description and Basis of Change

The UFSAR stroke time limit for the Reactor Water Cleanup System Supply Isolation Valves (MO2700 and MO2701) was based on dividing the nominal line size of 4 inches by the stem speed of 12 inches per minute and multiplying the result by 60 seconds per minute. The actual stem length, or total potential travel for these valves is 4 and 3/8 inches. Using this value instead of the nominal line size and rounding to the next significant digit results in a stroke time limit of 22 seconds. The original vendor quality control documentation for these valves clearly state a 22 second closure stroke time limit acceptance criteria in the valve test reports. Therefore, the Design Closure Times for the Reactor Water Cleanup System Supply Isolation Valves was changed from 20 seconds to 22 seconds in the table contained in the UFSAR.

Safety Evaluation Summary

Changing the limiting closure stroke time for the RWCU inlet isolation valves does not affect the probability of occurrence of a loss of coolant accident (LOCA) or any other accident evaluated previously in the SAR. The consequences of an accident evaluated previously by the SAR are not increased because the valve closing time meets the design basis requirements. Changing the limiting close stroke time for the valves does not increase the probability the valves will not close. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not increased. This change did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The increase in stroke time is within design bases limits for reactor isolation, containment isolation, and high energy line break concerns outside containment and their effect on equipment qualification. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. Since fuel integrity is maintained, margin to radiological dose limits defined in 10 CFR 100 is not reduced. The margin of safety as defined in the basis for any Technical Specification is not reduced.

SE 98-116 Circulating Water System P&ID Revision

Description and Basis of Change

This activity revised the Circulating Water System P&ID to correct the configuration of air lines associated with Cooling Tower 'A' air compressor. It was identified that the line downstream of solenoid valve SV4295 actually vented to the atmosphere and did not tie in to the line containing pressure switch PS4282A, as shown on the Circulating Water System P&ID. The error on the P&ID occurred when the compressor detail was first added to the drawing. The P&ID has been updated to correct this error. While performing a field walkdown, two additional discrepancies between the actual plant installation and the compressor detail shown on the Circulating Water System P&ID were identified. Air receiver 1T435A outlet relief valve PSV4237A still exists in the field. This valve was erroneously deleted from the P&ID by a previous revision. Subsequently, PSV4237A has been added to the P&ID. In addition, SG4299 identified in the equipment database as the "1T435A Outlet Pressure Sight Glass", is actually a cap assembly for the outlet air moisture indicator. This component is clearly identified in the Johnson Controls Vendor Manual. This error has existed since the detail was first added to the P&ID. The equipment identification on the P&ID has been changed from "SG4299" to "MI4299", and the equipment name has been changed to "1T4335A Outlet Air Moisture Indicator".

Safety Evaluation Summary

The Circulating Water System performs no safety related functions. The system is not included in the NSOA, is not an accident initiator, and is not required to mitigate the consequences of any accident analyzed in the UFSAR and NSOA. The affected system components are all non-safety related and are only included in the UFSAR as part of Figures. Therefore, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The Circulating Water System is not required to mitigate the radiological consequences of any accident described in the SAR. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. This activity revised the P&ID to reflect the as-built installation. The ability of the system to perform as designed was not impacted by this activity. This equipment can not adversely affect any safety related SSC. None of the affected components have any safety significance and the system itself is not an accident initiator or mitigator. Therefore, this activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The Circulating Water System is not included in the DAEC Technical Specifications. No equipment important to safety is impacted by this activity. Therefore this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-118 Emergency Operating Procedure (EOP) Changes

Description and Basis of Change

This evaluation included three specific changes made to the DAEC EOPs which were associated with the implementation of the generic guidance contained in the Boiling Water Reactor Owner's Group Emergency Procedure Guidelines/Severe Accident Guidelines (BWROG EPG/SAG) Revision 1. The three changes were:

1. In the Alternate Level Control (ALC) EOP, the allowance for a lower Reactor Pressure Vessel (RPV) water level of 2/3 core coverage (previously was Top of Active Fuel (TAF)) before requiring primary containment flooding. This revised step also defines an EOP to SAG transition step.

2. The revision of EOP Caution 1, RPV water level indication, which allows for evaluation and use of the level instruments during accident induced saturation conditions.

3. In EOP-2, the last step of the drywell temperature leg (DW/T-6) was revised to allow some flexibility at a higher drywell temperature before requiring emergency depressurization (ED) to be performed. This change is closely related to the deletion of a 200 psig hold point in the ED EOP if the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems are the only available source of makeup.

Change #1:
The change in the RPV level in Alternate Level Control (ALC) prior to entering SAG-1 for Primary Containment Flood (PCF) is supported by the original plant design basis for maintaining a floodable volume of 2/3 core coverage post DBA LOCA. Prior NRC approval exists (SER for Core Stabilities and Anticipated Transient Without a Scram (ATWS)) for the reduction in level in this step from the current TAF level to a calculated Minimum Steam Cooling Reactor Water Level (MSCRWL) of -30 inches

at the DAEC. The safety evaluation included both the change from TAF to MSCRWL (+15 inches to -30 inches) and the additional reduction to 2/3 core coverage (-39 inches). Additional related generic guidance, both industry and NRC, acknowledge the significance of avoiding or delaying primary containment venting. Since venting is expected upon entry into Primary Containment Flood (PCF), the delay or avoidance of entry into PCF achieved by this revision is consistent with the strategies provided by that guidance.

Change #2:
EOP Caution #1 identifies conditions under which RPV water level indications may become unreliable or must be considered invalid due to the effects of RPV pressure and drywell temperature. Caution 1 applies to EOP-1, EOP-2, ALC, and the ATWS-RPV Control EOP. Part 1 of Caution #1 identifies conditions beyond which boiling of the water in the instrument legs may occur. This change restored Caution 1, Part 1, to a caution versus a step that disallowed use of water level instruments. The specific change is from:

"RPV water level instruments may not be used when Drywell Air Temperature is above the RPV Saturation Temperature Curve (Graph 1).

to:

"If Drywell Air Temperature is above the RPV Saturation Temperature (Graph 1), RPV water level instruments may be unreliable due to boiling in the instrument runs."

This benefits the DAEC because it may preclude entry into the SAGs for events (such as the DBA LOCA) in which entry into the RPV Sat Curve (while the instruments are still viable for trending) would have previously required RPV flooding. For the DBA LOCA, it would not be possible to establish RPV/F conditions and entry into the SAGs would have been required. This possibility was eliminated by this change. This change is also consistent with the DAEC UFSAR discussions of RPV water level instrumentation and is in accordance with an approved EOP issue package with the BWROG that was completed after the EPG/SAG Revision 1 issuance.

Change #3:
In EOP-2, Primary Containment Control, the last step of the Drywell (DW) temperature leg is to enter Emergency Depressurization (ED). The current decision step prior to entering ED, is step DW/T-6 which stated:

"*WAIT UNTIL...drywell temperature cannot be maintained below 280F*"

The revised step states:

"*WAIT UNTIL...drywell temperature cannot be* ***restored and*** *maintained below 280F*" (bold added to reflect change).

This change allows operator discretion for commencing ED once Drywell temperature reaches 280°F. Prior to this change, the operator was procedurally required to commence ED regardless of the potential restoration of drywell cooling components or consideration of the event in progress that caused the high drywell temperature.

The evaluation under this item also includes the deletion of the 200 psig hold point Continuous Recheck Statement (CRS) in the ED EOP flowchart. The change to step DW/T-6 was driven, in part, by the deletion of the CRS in the ED EOP. That hold point was a deviation from previous generic guidance (EPG Rev. 4). Deletion of the CRS eliminates this deviation and is consistent with the BWROG philosophy that Station Blackout (SBO) actions reside in an event-based Abnormal Operating Procedure (AOP) versus the EOPs. This change to step DW/T-6 is also in accordance with an approved EPG issue resolution package, which was approved (by the BWROG) after the issuance of EPG/SAG Revision 1.

Safety Evaluation Summary

Change #1

The revised EOP steps direct actions to mitigate accidents once an event has already been initiated. The guidance in the revised step does not cause or initiate an event separate from the one being mitigated. The reduced RPV water level associated with this change does not increase the probability of occurrence of the DBA LOCA. The previously evaluated accident associated with this revision is the large break LOCA. For this event, the RPV level can be restored to an elevation slightly higher than the top of the jet pumps (-39 inches). Since the revised transition point can be met within the design basis, the EOPs will ensure that the event proceeds as analyzed and the operators will not unnecessarily enter the EOP path that requires primary containment flooding and RPV or containment venting within design basis space. Since the result of this revision helps avoid this venting, no previously unanalyzed release path is created and radiological dose consequences will not be affected. The post-LOCA containment analysis, shows there is adequate Net Positive Suction Head (NPSH) margin for the Emergency Core Cooling System (ECCS) pumps for the first 10 minutes even if the pump flow is maximized. After the RPV level is restored and maintained above -39 inches, the operators would initiate torus cooling and operate ECCS pumps within design limits. This is also consistent with the post-LOCA containment analysis and ensures that the torus temperature would not exceed calculated values. Therefore the change does not increase the probability of a malfunction of equipment important to safety previously evaluated in the SAR. There are no physical plant modifications or changes associated with this EOP revision. Therefore, this change does not create the possibility of an accident of a different type than any previously evaluated in the SAR. The possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR was not created. An applicable margin of safety associated with this revision is that provided by the plant response to the DBA LOCA as evaluated in the associated analyses. This change does not invalidate any analysis assumptions. The lower RPV water level associated with this change from the current EOPs remains within the NRC acceptance value of that provided within the design and licensing bases of 2/3 core coverage. Margin is actually gained from a radiological release standpoint by the related delay or avoidance of RPV/Containment Venting.

Change # 2

This change allows the evaluation and possible use of RPV water level instrumentation under previously evaluated accident conditions. The associated EOP steps involve mitigation actions for events/accidents if they were to occur. None of the steps associated with the revised caution nor the revised caution itself can actually initiate an event. This change did not increase the probability of occurrence of an accident previously evaluated. This change allows the operator to conclude that exceeding RPV saturation temperature, by itself, does not constitute a requirement for RPV Flooding. Instead, if the saturation curve is entered, the operating crew can look for indications of flashing and evaluate the adequacy of vessel level indication. Since the drywell will be at saturation conditions (not superheated), flashing is not expected to occur. Therefore, transition to RPV flooding should not be made in a design basis LOCA and the operator's actions would be consistent with the assumptions in the design and licensing basis analyses. This activity does not increase the radiological dose consequences for the previously evaluated accidents in the SAR. The potential malfunction of vessel level instrumentation as a result of saturation conditions and plant depressurization is neither increased nor decreased with this change. This allowance may actually decrease the probability of equipment malfunctions by avoiding the transient associated with RPV Flooding. Therefore, this change does not increase probability of occurrence of a malfunction of equipment important to safety which was previously evaluated. This change does not increase the radiological dose consequences of a malfunction of equipment important to safety which was previously evaluated. This change and the use of level indication instrumentation is consistent with UFSAR assumptions of instrumentation operability. No physical plant modifications or changes were made. This change does not create the possibility of an accident of a different type than any previously evaluated in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR was not created. The applicable margin of safety for this change is that provided by avoiding unnecessary RPV or Containment Venting. Since the NRC acceptance value for this parameter is "no venting" (since it was not assumed in the SAR) and since this change helps avoid the need for venting, this change actually adds margin. Therefore, there is no reduction in the margin of safety as defined in the basis for any Technical Specification.

Change # 3

This step involves actions to mitigate the consequences of accidents and/or transients previously evaluated (LOCA and SBO) in the SAR. This change allows operator discretion for commencing ED once Drywell temperature reaches 280°F. The flexibility to exceed a drywell temperature of 280°F does not increase the probability of occurrence of an accident previously evaluated in the SAR. The flexibility to exceed 280°F in the SBO event provides a delay before ED is required which helps preserve the motive force for HPCI and RCIC, which is needed to allow the deletion of the hold point (CRS) in the ED EOP. If ED was performed by the operator (though not expected to be necessary) in the SBO, RCIC availability is assured to provide adequate coolant makeup and core cooling, which is within the bounds of the analysis assumptions and the RCIC System design basis. This change does not impact the peak drywell temperatures for the LOCA analyses previously evaluated. Containment malfunction under accident conditions is based upon peak containment pressures, not temperature. This change has no impact on primary containment pressure during analyzed events. The radiological dose consequences from accidents previously evaluated in the SAR are not increased. This revision does not increase the probability of a malfunction of equipment during previously evaluated LOCAs. The overall strategy of EOP-2 is maintained for managing drywell temperature by using available equipment to maintain drywell temperature below the withstand limit with as much margin as achievable. This strategy is still within the Current Licensing Basis (CLB) that includes allowances for temperature up to 340°F. There is no negative impact or malfunction on the drywell structure as a result of this change. This revision does not increase the probability of a malfunction of equipment previously evaluated during the SBO event. This change helps with event mitigation and does not create the possibility of an accident of a different type. This change does not create the possibility of a malfunction of equipment important to safety of a different type. The margin of safety is not reduced because the NRC acceptance value will not be exceeded in the SBO event as a result of this change. Since 340°F will not be allowed to be exceeded by this change, this margin of safety is preserved. Therefore, these revisions do not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-120 (Revision 1) Change To General Service Water (GSW) and Circulating Water (CW) Chemical Treatment Program

Description and Basis of Change

This activity changed the GSW, CW, and Cooling Tower Systems' chemical treatment program to the new BetzDearborn Dianodic Plus chemical treatment package. This activity changed the treatment chemical added to the GSW and CW Systems, and by virtue of water flow through the cooling towers, the cooling towers also. The chemical addition equipment did not require any physical changes. The new treatment program provides increased cathodic protection, which reduces mild steel pitting without the addition of zinc based corrosion inhibitors. Copper alloy treatment with halogen resistant azoles provide corrosion protection for the copper bearing components in the CW and GSW Systems. This new treatment program is more environmentally friendly since acid feed is reduced by controlling CW at a sightly higher pH. This new program is not susceptible to chemical loss due to chlorine in the make up or cooling tower water (as the previous treatment was) and it allows increased cycles of concentration, lower blow down rates, and better carbon steel and copper alloy corrosion protection.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Corrosion protection and long term performance of the affected systems were improved. This activity meets all design, material and construction standards for the systems in question. System interfaces were not changed by the use of the new chemical treatment program. Since there is no impact on the operation of the affected systems, the consequences of an accident evaluated previously in the SAR were not increased. The probability of a malfunction was not changed, and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased since the affected systems are not relied upon or prevent or mitigate an accident or the radiological consequences of an accident. As no new components or failure modes were added by this activity, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. The GSW, CW, and Cooling Tower System functions were not adversely affected by this activity. There are no Technical

Specification operational criteria associated with the CW, GSW, and Cooling Tower Systems. There are no components that require CW, GSW, or the Cooling Towers to maintain a margin of safety as defined in the basis for any Technical Specification. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-121 Well Water System P&ID Revision

Description and Basis of Change

Two isolation valves that were not shown on the P&ID for the Well Water System were found on pipeline stubs in the plant. This change revised the Well Water System P&ID to add the normally closed instrument valves. The basis for this change was a plant walk down and a previous temporary modification for installing differential pressure instrumentation on this pipeline, which noted the isolation valves already existed. These instrument lines are used to measure the flow rate in the 'A' Drywell Cooling Coil Loop Well Water return line. These valves are not associated with the process or operation of the well water to and from the drywell coolers. They are not safety related, and the subject well water return line has no seismic requirements.

Safety Evaluation Summary

The subject valves are not safety related, and they are on the non-seismic related pipeline of the Well Water System. This system is not an initiator of an accident. Therefore, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The Well Water System, on which the subject valves are installed is not included in the NSOA, and it is not required to mitigate the consequences of an accident. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. The subject valves are normally closed, and are located on instrument tapping lines on the well cooling water return line which has no safety requirements. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The subject valves are associated with small diameter instrument lines, and any postulated malfunction or failure of this equipment will not cause any significant loss of water in the system. The Well Water System is not safety related. Therefore, this activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. No credible accident can be created by this change because, the Well Water System is neither an initiator nor a mitigator of an accident. Therefore, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. No new failure modes can be created by this change because no safety function of any equipment is impacted by this change. Therefore, this activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings or safety limits are defined in the Technical Specifications for the Well Water System. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 98-123 Revision To Abnormal Operating Procedure (AOP) For Station Blackout

Description and Basis of Change

The Station Blackout AOP was revised to instruct control room operators to halt manual depressurization of the RPV between the range of 150 to 200 psig rather than between the range of 200 to 400 psig. This change provides consistency with vessel parameters assumed in a plant specific analysis of containment response to station blackout events. Vessel depressurization is performed during station blackout in order to maintain drywell atmosphere temperature below 300 °F. Halting depressurization between 150 and 200 psig allows for continued RPV injection by the RCIC and HPCI systems and at the same time, ensures no potential impact on operability of components within the drywell required for station blackout coping.

Safety Evaluation Summary

The probability of occurrence of accidents and malfunctions of equipment important to safety is not increased by this activity because this change affects the response to the station blackout event, rather than the initiation frequency of this event. The frequency of other accidents and abnormal operational occurrences described in the SAR are not increased. The consequences of station blackout is not increased because drywell atmosphere temperature remains lower for RPV pressure in the 150 to 200 psig range than for RPV in the 200 to 400 psig range. This enhances operability of components within the drywell. Continued RPV injection is assured because the new RPV pressure is within the range required for operation of RCIC and HPCI. Final torus water temperature is not increased because decay heat energy transferred from the fuel to the torus in the four hour coping period is not substantially changed by this activity. The consequences of other accidents and malfunctions of equipment important to safety is not increased because this change only applies to operational guidance for coping with station blackout. The possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the SAR is not created because this activity only revises procedural guidance for depressurizing the RPV during station blackout. Containment response to station blackout is essentially unchanged and no new event is created. The margin of safety as defined in the basis for any Technical Specification is not reduced because station blackout analysis assumptions are maintained.

SE 98-124 Technical Specification Bases Revision For Source Range Monitoring (SRM) And Intermediate Range Monitoring (IRM)

Description and Basis of Change

The Technical Specification Bases for Surveillance Requirement (SR) 3.3.1.1.6 and SR 3.3.1.1.7 stated, "Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above mid-scale on range 1 before SRMs have reached the upscale rod block (i.e., approximately one-half decade of range)." This statement was contradictory since mid-scale on range 1(0-40) does not correspond to one-half decade of range. Neither method was readily supported by the DAEC Design Bases or operating experience. A Technical Specification Bases change and associated plant procedure changes were made to revise/clarify the SRM/IRM overlap criteria to state, "Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are indicating at least 5/40 on range 1 before SRMs have reached 10^6 counts per second."

Safety Evaluation Summary

The SRM and IRM systems are not an assumed initiator of any analyzed accident previously evaluated in the SAR. This change did not affect the form, fit, or function of any SSC credited in the SAR or the operation of any such SSC. This change did not alter the accident initiators assumed in the SAR. Since this evaluation justifies enhancement of the requirements, which satisfy the design basis, this activity did not increase the probability of accident occurrence as previously evaluated in the SAR. This change did not alter the method by which the plant is operated. The change only altered the method for verifying proper SRM/IRM overlap. The IRM instrumentation is assumed in the safety analyses to mitigate a neutron flux excursion caused by a positive reactivity addition due to a continuous rod withdrawal event. This change did not impact the ability of the IRMs to perform that safety function. The dose consequences of the analyzed accidents in the SAR remain unchanged. The changes continue to ensure proper SRM/IRM overlap is demonstrated while maintaining the current design bases of the SRM and IRM instrumentation. In addition to the SRM/IRM overlap verification, OPERABILITY of the SRM and IRM subsystems are determined by means of channel checks, channel functional tests, and channel calibrations of the associated instruments. As such, this activity did not increase the probability of a malfunction of equipment important to safety. No new failure mode or equipment malfunction was introduced and the potential for any equipment failure is unchanged. This activity did not increase the consequences of a malfunction of equipment important to safety. Therefore, no new failure mode or equipment malfunction was introduced and the dose consequences of the analyzed accidents in the SAR remains unchanged from this evaluation. The accidents evaluated in the SAR remain bounding and no accidents of a new or different type were introduced. This change did not make any changes in plant design or physical operation. No new or different types of malfunctions were introduced. This evaluation did not alter any acceptance limits previously approved by the NRC. The fuel thermal acceptance limits were not impacted by this change. Since there was no change in either the acceptance limit or the failure points of the SSCs, the Margin of Safety was not altered.

SE 98-125 Control Building Air Conditioning System P&ID

Description and Basis of Change

The P&ID for the Control Building (CB) Air Conditioning System was revised as follows:

- The 'B' instrument air supply to the temperature transmitter and controllers for the CB air conditioning unit was erroneously shown from "Common" H&V Instrument Air, while it is actually supplied from the 'B' H&V Instrument Air Compressor System. The "Common" supply isolates to protect the 'A' and 'B' air supplies should a low pressure condition occur. This loss of "Common" air supply equates to a loss of the control air to control components resulting in a closed temperature controller, preventing chilled water flow to the cooling coil of the 'B' side CB A/C Unit. This activity changed the source of air supply to the 'B' H&V Instrument Air Compressor System. This change is consistent with the Johnson Controls panel and provides more reliable air supply to the subject control components. Similarly, the air supply to the control valve for the 'A' side CB air conditioning unit was changed from "Common" to the 'A' H&V Instrument Air Compressor System, which is a more reliable air supply.

- The air supply to the CB H&V Cable Spreading Room Exhaust Temperature Controller air supply was erroneously shown as being supplied by the 'A' H&V Instrument Air Supply System. The air supply to the CB H&V Cable Spreading Room Exhaust Temperature Controller was changed from the 'A' H&V Instrument Air Supply System to the "Common" Instrument Air Supply. This equipment and the chill water supply to the Cable Spreading Room are not safety related. Therefore, this change did not impact the safety function of the CB HVAC System.

Safety Evaluation Summary

The CB HVAC System is not an initiator of any transients, accidents, or special events evaluated in the SAR. Hence, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The Control Building HVAC is a support system that functions to mitigate the consequences of an accident by assuring the habitability of the Control Room. There are no radiological consequences associated with this activity, and this change ensures separate safety related air supplies to the redundant chill water control systems. Hence, this activity did not increase the consequences of an accident evaluated previously in the SAR. The change addressed in this safety evaluation for the equipment associated with the Control Room HVAC only adds additional assurance of a separate, redundant air supply to chill water control components, and does not impact the method of operation of the plant. Therefore, this activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This change improved the reliability of the Control Room HVAC. The change associated with the Cable Spreading Room temperature control is not associated with any safety function. Therefore, this activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. No new failure modes were created by this activity because the air supply to the equipment addressed in this change is either supplied with a more reliable source, or where the supply is changed to less reliable source, the affected equipment is not required to perform any safety function. Therefore, this activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The ability of the Control Building HVAC to perform its safety function was not changed by this activity. The Control Building HVAC System is still operable per DAEC Technical Specifications. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-006 Reactor Coolant Conductivity Monitoring Technical Requirements Manual (TRM) Revision

Description and Basis of Change

The TRM was revised to allow for the use of conservative grab sample conductivity measurements to verify that reactor coolant conductivity requirements are being met when in-line conductivity monitoring is unavailable.

Safety Evaluation Summary

The probability of occurrence of an accident evaluated previously in the SAR was not increased. Reactor Coolant System (RCS) water quality requirements were established to prevent damage to the materials in the primary system. Periodic conductivity measurements are acceptable when continuous conductivity monitoring is unavailable. Conductivity measurements of RCS grab samples will be conservative and the existing limits still apply. Thus, the integrity of the materials in the primary system will not be adversely affected. There are no changes to plant operating practices, system availability/reliability, or accident mitigation strategies. Therefore, the consequences of an accident evaluated previously in the SAR were not increased. The probability of occurrence of a malfunction of equipment important to safety as evaluated previously in the SAR was not increased, and the consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. This change allows for the use of conservative conductivity measurements to verify that RCS water quality requirements for conductivity are being met. There are no changes to plant hardware or operating practices that could be accident initiators. Therefore, the possibility of an accident of a different type than any evaluated previously in the SAR was not created. There are no changes to plant hardware or operating practices that could affect equipment important to safety. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. There are no changes to plant operating practices, system availability/reliability, or accident mitigation strategies. The integrity of the materials in the primary system will not be adversely affected. The margin of safety as defined in the basis for any Technical Specification was not reduced.

SE 99-009 Technical Requirements Manual (TRM) Revision To Extend Snubber Testing Frequency

Description and Basis of Change

DAEC will extend the plant's operating cycle from 18 months to 24 months. A TRM revision to section T 3.7.2, extended the surveillance frequency for both snubber visual examination and functional testing from 18 months to 24 months (with a 25% grace period). The TRM imposes surveillance requirements for visual inspection and functional testing of all safety related snubbers. To verify that a snubber can operate within specific performance limits, functional testing is performed that typically involves removing the snubber and testing it on a specially-designed test stand. Functional testing provides a 95 % confidence level that 90 % to 100 % of the snubbers operate within the specified acceptance limits. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber operability. Visual inspection requirements are based on Generic Letter (GL) 90-09. GL 90-09 already recognizes a 24 month operating cycle and allows extended surveillance frequencies. Functional testing has a statistical bases where the sample plan provides a 95 % confidence level that 90 to 100 percent of the snubber population is operable.

Safety Evaluation Summary

Snubbers are not initiators of accidents previously evaluated in the SAR. This change did not increase the probability of occurrence of an accident evaluated previously in the SAR. A snubber's safety related function is to become rigid during a dynamic event such as an earthquake. A snubber's passive function is to move freely during slow movements such as thermal growth. The SAR does not address accidents that are initiated due to a snubber failure, but assumes snubbers are operable and that their operability will be verified with visual examination and functional testing. Snubber visual examination and functional testing surveillance frequency is not specified in the SAR. Extension of the operating cycle to 24 months did not degrade the snubber population's operability confidence level below 95%, which is the basis for DAEC's Snubber Testing Program. The SAR does not take credit for a snubber's restraint capabilities following an accident. This change did not adversely affect snubber operability pre or post accident, and therefore, it did not increase the consequences of an accident or the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. There was no increase in the consequences of a malfunction of equipment important to safety evaluated previously in the SAR and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. This change did not reduce the margin of safety as defined in the basis for any Technical Specification. This change maintained the 95% confidence level that 90% to 100% of the snubbers will be operable following a 24-month operating cycle.

SE 99-010 Control Building Air Conditioning System P&ID Revision

Description and Basis of Change

The P&ID for the Control Building Air Conditioning System showed the Control Building Humidity Steam Generator Automatic Drain Valve as a solenoid valve when it is actually a motor operated ball valve. The function of the valve is to cycle to drain the condensate periodically. There is no change in the function or logic of the valve. The subject valve has no safety function.

Safety Evaluation Summary

The Control Building HVAC System is not a contributor or initiator of any transients, accidents, or special events evaluated in the SAR. Hence, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The Control Building HVAC is a support system that functions to mitigate the consequences of an accident by assuring the habitability of the Control Room. There were no radiological consequences associated with this activity. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This activity only changed one type of valve actuator to another for an automatic drain for the humidifier. The subject equipment is not associated with any radiation monitoring equipment, and hence there is no impact on control room habitability as a consequence of a postulated malfunction of the equipment. Therefore, this activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. In a worst case scenario, the humidity control for the HVAC building may be affected, but this is not listed as an unacceptable result in the NSOA. Furthermore, the equipment is easily accessible for draining the condensate manually, in case of failure of the subject valve, to preclude the effects of high humidity in the Control Room. Therefore, this activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings or safety limits are defined in the Technical Specifications for the Control Building humidifier. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-012 Emergency Core Cooling System (ECCS) Suction Strainer Design Activities

Description and Basis of Change

The potential for Emergency Core Cooling System (ECCS) pump suction strainer clogging following a Loss of Coolant Accident (LOCA) has been the subject of several NRC bulletins/generic letters. In response to NRC Bulletin 96-03 (NG-97-1909), the DAEC committed to installation of passive strainer devices capable of assuring adequate Net Positive Suction Head (NPSH) margin for the ECCS following a LOCA. This resolution option was identified in NRC Bulletin 96-03 as one method that could be implemented to ensure the capability of the ECCS to perform its safety function following a LOCA. The new passive, large-capacity strainers were installed on the Core Spray (CS) and Residual Heat Removal (RHR) suction lines during Refueling Outage (RFO) 15 in the spring of 1998. The new suction strainers are a stacked disc configuration provided by General Electric (GE) and designed in accordance with the GE Licensing Topical Report (LTR) NEDC-32721P, rev 1, Application Methodology for the General Electric Stacked Disc ECCS Suction Strainer. The structural installation of the new strainers was performed as part of Engineering Change Package (ECP) 1588 and was evaluated as part of Safety Evaluation (SE) 98-29.

At the time of the installation of the larger strainers, the BWROG Utility Resolution Guide for ECCS Suction Strainer Blockage (URG) and GE strainer LTR were awaiting approval by the NRC. Therefore, although the strainers were installed, the work associated with closure of NRC Bulletin 96-03 was delayed pending resolution of issues with the URG and LTR. The purpose of SE 99-012 was to support the remaining design activities including Net Positive Suction Head (NPSH) analysis, containment minimum pressure analysis, debris generation, strainer head loss, UFSAR revisions and other documentation updates.

Evaluation of the need to replace the HPCI suction strainer was performed. Several issues were reviewed, including the debris generation expected for small, medium, and large break Loss of Coolant Accidents (LOCAs) and Probabilistic Risk Assessment (PRA) results for strainer clogging or High Pressure Coolant Injection (HPCI)/Reactor Core Isolation Cooling (RCIC)

System failures. The results of this review showed negligible safety benefit from the installation of replacement strainers for the HPCI.

Safety Evaluation Summary

Neither the probability of occurrence nor the consequences of a previously evaluated accident were increased. This change revised the licensing basis with respect to events that happen after a LOCA. Changes to the containment pressure/temperature analysis, debris generation calculations, debris transport, strainer loading, strainer performance, and NPSH analysis are associated with the response of the containment and ECCS following a LOCA. These changes did not affect or modify any operating conditions or initiators of any of the accidents analyzed previously in the SAR. The changes did not alter, degrade or prevent any actions described or assumed in an accident to mitigate the event.

Neither the probability of occurrence nor the consequences of a malfunction of equipment important to safety previously evaluated in the SAR were increased. The changes to the design methodologies for strainer performance, containment performance, NPSH analysis, and debris analysis are in conjunction with evaluated analytical methods developed by the BWROG and General Electric. The debris generation techniques used for this modification are in accordance with the URG and supported directly by a NRC SER. The work performed by General Electric to determine the strainer head loss has been evaluated and accepted by the NRC via the SER. The head loss determined for the strainer has been input into NPSH calculations prepared by the DAEC, which show the acceptable performance of the ECCS in conjunction with the existing license of the DAEC. The changes to the inputs identified in the UFSAR are in accordance with the current operating parameters of the DAEC, defined in Technical Specifications, and in accordance with other NRC guidance for preparation of ECCS analysis, in both published documents and the original DAEC safety evaluation. Compared to that previously evaluated in the SAR, this change greatly increases the surface area of suction strainers to accommodate the assumptions of debris generation with a large area zone of influence for debris generation in the event of a LOCA. These assumptions are significantly more conservative than that previously evaluated in the SAR.

No new failure modes were introduced since no equipment is being relocated and the systems' functions and operation were not changed. The possibility of an accident of a different type than previously evaluated was not created. The installation of new ECCS suction strainers enhanced the ability of the existing systems to perform their safety functions, with significantly more transient debris on the strainers, following an accident.

Both the new and old strainers were passive devices supporting the performance of the RHR and CS pumps. The possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SAR was not created. The margin of safety as defined in the basis for any Technical Specification was not reduced. Although the ECCS and containment are referenced in the Technical Specifications, there was no change to the required action for the conditions specified and no change in the surveillance requirements. The Low Pressure Coolant Injection (LPCI) and CS pump flow rates and test frequency were not modified. The required minimum water level in the suppression pool and the condensate storage tank and the frequency of their verification remain the same. There was no effect on the average suppression pool temperature.

SE 99-013 Reactor Pressure Narrow-Range Recorder Discrepancy

Description and Basis of Change

UFSAR Section 7.5.1.2 was revised by changing the range of the reactor pressure narrow-range recorder to "800 to 1100 psig" from "850 to 1150 psig". This recorder is a part of the non-nuclear instrumentation system and has no nuclear safety function. The GE Specification for this instrument, which listed the range as "900 to 1100 psig", was also revised to "800 to 1100 psig". The bases for this change are the GE Instrument Data Sheet, the GE Elementary Diagram for the Feedwater Control System and the instrument calibration records.

Safety Evaluation Summary

Because the subject recorder has no safety function or safety significance, and is not an initiator of an accident, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The subject narrow-range pressure recorder receives its signal from a pressure transmitter, for which the only safety-related function is the reactor coolant pressure boundary. This safety function was not affected by this change. The recorder is not required for mitigation of an accident or post-accident monitoring. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. Changing the range of the transmitter and recorder signal did not affect the transmitter pressure boundary function. Therefore, this activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. The overall function of the recorder remains the same. There are other pressure indicators and recorders enveloping the reactor pressure at greater than 1100 psig that the operators can use. Therefore, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings or safety limits are defined in the Technical Specifications for the subject recorder. This change had no affect on the Technical Specification bases for instrumentation. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-014 UFSAR Change – Reactor Water Cleanup Header Isolation Valve Closure Time

Description and Basis of Change

This UFSAR change increased the Reactor Water Cleanup (RWCU) header isolation valve closure stroke time limit from 10 seconds to 20 seconds. The stroke time limit change reflects the original Final Safety Analysis Report (FSAR) table, and matches the discussion found in the paragraph preceding the table on how stroke time maximum limits were calculated. This change still ensures the valve meets its design bases requirements for reactor vessel isolation, and containment isolation, and does not affect the design bases for equipment qualification for line breaks outside containment.

Safety Evaluation Summary

This change did not increase the probability of occurrence of an accident evaluated previously in the SAR. Changing the limiting closure stroke time for the RWCU header isolation valve did not affect the probability of occurrence of a loss of coolant accident or any other accident evaluated previously in the SAR. This activity did not increase the consequences of an accident evaluated previously in the SAR. The proposed limiting closure stroke time meets the design bases requirements for reactor vessel isolation and containment isolation. Since the inboard containment isolation is a check valve, per the design bases the new limiting closure stroke time will not affect the ability to isolate the RWCU header in time to prevent the core from being uncovered, thus avoiding core damage. Also, in accordance with the design bases, the closure stroke time for this valve is not critical due to the relatively long time required for fission products to distribute within the containment. The change in limiting stroke time does not affect the leakage rate after valve closure. This activity did not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The consequences of a valve malfunction have not changed and the valve's containment isolation function was not affected. This activity did not create the possibility of an accident of a different type, and it did not create the possibility of a malfunction of equipment important to safety of a different type, than any evaluated previously in the SAR. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification. The RWCU header isolation valve with the inboard feedwater valve still prevents the core from being uncovered.

SE 99-016 UFSAR Change - Emergency Diesel Generator (EDG) Inspection Requirements

Description and Basis of Change

The purpose of this UFSAR change was to remove the specific time requirements regarding preventive maintenance for the EDG System. The UFSAR wording tied the preventive maintenance requirement to a specific time and stated that the maintenance would be performed in accordance with the manufacturer's recommendations. Recent manufacturer's recommendations allowed for some adjustments to the maintenance intervals, which required an UFSAR change to utilize. The function of the EDG System is to provide a dependable source of emergency AC power to the 4160V essential busses as necessary to safely shutdown the plant and protect against postulated accidents in the event of a Loss of Offsite Power (LOOP). In light of that function, the equipment must be maintained to a high degree of readiness. The preventive maintenance program for the EDG System is based on the original equipment manufacturers (OEM) recommendations, industry operating experience (based on the Fairbanks Morse Owners' group maintenance guidelines) and DAEC operating experience.

Safety Evaluation Summary

The EDG is not part of any initiating event for the accidents described in the SAR. These changes did not result in an increase in the probability of occurrence or an increase in the consequences of an accident evaluated previously in the SAR. Since the preventive maintenance requirements are aimed at determining long term wear, the extension of the inspection cycle and performance on-line did not result in an increase in the probability of occurrence of, or the consequences of, a malfunction of the equipment important to safety evaluated previously in the SAR. There were no changes in the data points for the inspections, therefore system reliability was not decreased. The changes did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and they did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. This activity had no effect on the margin of safety as defined in any Technical Specification.

SE 99-017 UFSAR Change – Essential AC Power Circuit Breakers' Preventive Maintenance

Description and Basis of Change

The purpose of this UFSAR change was to remove the specific time requirements regarding preventive maintenance for the Essential AC Power Circuit Breakers. The UFSAR wording tied the preventive maintenance requirements to a specific time and stated that the maintenance be performed in accordance with the manufacturer's recommendations. Recent manufacturer's recommendations allowed for some adjustments to the maintenance intervals. The functions of the Auxiliary AC Power System are to provide a source of AC power to various plant systems during startup, normal operations and shutdown of the plant and protect against postulated accidents. In light of these functions the equipment must be maintained to a high degree of readiness. The preventive maintenance program for the Auxiliary AC Power System is based on the original equipment manufacturers (OEM) recommendations, industry operating experience and DAEC operating experience.

Safety Evaluation Summary

The Auxiliary AC Power System is not part of any initiating event for the accidents described in the SAR. These changes did not result in an increase in the probability of occurrence or an increase in the consequences of an accident evaluated previously in the SAR. Since the preventive maintenance requirements are aimed at determining long term wear, the extension of the inspection cycle did not result in an increase in the probability of occurrence of, or the consequences of, a malfunction of equipment important to safety evaluated previously in the SAR. There were no changes in the data points for the inspections, therefore system reliability was not decreased. The changes did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and they did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. This change had no effect on the margin of safety as defined in any Technical Specification.

SE 99-019 Offgas Recombiner P&ID Revision

Description and Basis of Change

The P&ID for the Offgas Recombiner was revised to show the manual drains for the shell side and tube side of the Offgas Condenser, as installed in the plant. The P&ID erroneously depicted the tube side drain valve on the shell side, and did not show the shell side manual drain plug.

Safety Evaluation Summary

This change corrected the depiction of the manual drains for the Offgas Condenser on the P&ID. Since the Offgas System is not an initiator of an accident, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. An operational failure or a component failure of the Offgas System will not result in a site boundary dose that is an appreciable fraction of 10 CFR 100 limits. This change was accordance with the intended design and function of the Offgas Condenser. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. Without the change, the plant documentation and labeling could mislead plant personnel, resulting in possible malfunction of the system. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The Offgas System is non-safety related. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The possibility of an accident of a different type than any evaluated previously in the SAR was not created. No failure modes that could impact the performance of the system were identified by this change. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings, or safety limits are defined in the Technical Specifications for the Offgas Condenser. This change had no effect on the Technical Specification bases. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-020 Containment Atmosphere Control System P&ID Revision

Description and Basis of Change

The Drywell and Torus vent line isolation valves, Torus vacuum breaker isolation valves, and the containment purge supply isolation valves of the Containment Atmosphere Control System are operated by pneumatic actuators. The basic design and components in the local air supply piping are identical for all these valves. However, the P&ID depicted the piping components for some of these valves different from the others. The purpose of this activity was to correct this anomaly and show the air supply piping on the P&ID in a consistent manner to correctly represent the installed configuration. No physical change to the plant was involved. The basis for this change was a plant walkdown and vendor drawings.

Safety Evaluation Summary

No physical change to the plant was involved, the fail-safe position of the control valves was not affected, and the system functions as before. Therefore, this activity did not increase the probability of occurrence, or the consequences, of an accident evaluated previously in the SAR. Without this change the plant documentation and labels could mislead plant personnel. This activity did not increase the probability of occurrence, or the consequences, of a malfunction of equipment important to safety evaluated previously in the SAR. Correctly depicting these valves on the P&ID did not introduce any new failure modes. Therefore, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR, and the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not created. No safety margins, safety settings, or safety limits are defined in the Technical Specifications for the subject control valves. This change had no effect on the operability requirements of these valves as given in Technical Specification bases. This activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-033 (Revision 1) **Non-dependency Of Suppression Pool Spray and Drywell Spray Systems On The Residual Heat Removal Service Water (RHRSW) System**

Description and Basis of Change

Suppression Pool Spray and Drywell Spray Systems provide the spray function to the containment atmosphere to condense any existing steam and non-condensable gas to reduce containment pressure if desired during the Design Basis Accident (DBA) Loss of Coolant Accident (LOCA). The Suppression Pool Cooling System in conjunction with the RHRSW System and associated Residual Heat Removal (RHR) heat exchanger, on the other hand, provides the Suppression Pool Spray and Drywell Spray Systems the capability of reducing containment pressure following a DBA-LOCA and is the primary means of containment heat removal. During a LOCA, when the Suppression Pool Spray and Drywell Spray Systems are initiated by operator action, the RHR pump flow is diverted from the reactor vessel to the Containment Spray. The pump flow is routed through a RHR heat exchanger, where it is cooled by Suppression Pool Cooling before being discharged into the Containment Spray headers. The sprayed water collects in the bottom of the Drywell until it rises and drains back down to the Suppression Pool. Hence, both Containment Spray (performed by the spray headers) and Suppression Pool Cooling (performed by the heat exchanger and RHRSW) functions are performed. As analyzed in the GE Topical Report, "DAEC Containment Analysis", GE-NE-T2300752-00-01-R2, dated July/98, the Suppression Pool Spray and Drywell Spray Systems are capable of performing their design functions without the use of the RHR heat exchanger because the suppression pool temperature is maintained below saturated conditions by the Suppression Pool Cooling System during the DBA-LOCA. Suppression Pool Cooling provides Containment Spray the capability of reducing containment pressure following a DBA-LOCA, and is the primary means of containment heat removal. There are Technical Specifications (TS) for assuring the operability of the Suppression Pool Cooling. Therefore, the use of the RHR heat exchanger and RHRSW pumps, while necessary for Suppression Pool Cooling, is not required for the spray function. In addition, per the UFSAR, Containment Spray operation is not required from the standpoint of reactor safety. The purpose of Containment Spray is to provide an alternative method of reducing containment pressure following a DBA-LOCA.

This change clarified the non-dependency of the Suppression Pool Spray and Drywell Spray Systems on the RHRSW System in the TS Bases and Technical Requirements Manual (TRM) Bases.

Safety Evaluation Summary

This activity did not change any plant equipment or the operation of the plant. As long as Suppression Pool Cooling is available, the ability to maintain Net Positive Suction Head to the Emergency Core Cooling System pumps is assured. Sending Suppression Pool water through the heat exchanger of the RHR loop performing Containment Spray, without RHRSW on the other side of the heat exchanger, will not damage the heat exchanger. Therefore, this activity did not increase the probability of occurrence or the consequences of an accident evaluated previously in the SAR. It did not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The possibility of an accident or malfunction of equipment important to safety of a different type than evaluated previously in the SAR was not created. In addition, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-037 Minor Changes To The UFSAR

Description and Basis of Change

This Safety Evaluation evaluates the 10 CFR 50.59 screening of "minor" changes to UFSAR Figures such that a written Safety Evaluation is not required. A large proportion of UFSAR changes and associated Safety Evaluations were performed for UFSAR discrepancies and minor plant changes that caused UFSAR Figures to be revised. These changes rarely altered the design bases as described in the UFSAR text or Tables. Many of these were minor corrections, with no safety significance. The NRC manual, Part 9900 on 10 CFR Guidance, "Changes to Facilities, Procedures, and Tests (Experiments)", Subsection D.7.d states that the intent of 10 CFR 50.59 is to limit the requirement for written safety evaluations to facility changes, tests, and experiments which could impact the safety of operations. By providing a list of criteria for "minor" changes that would not require a written Safety Evaluation as an attachment to the Safety Evaluation Applicability Review (SEAR) procedure, the number of written Safety Evaluations should be reduced. Written Safety Evaluations would not need to be performed for "minor" changes, as the "pre-approved" criteria would have already been evaluated. Full plant P&IDs will remain in the UFSAR since they are useful to the user and easy to update. This activity implemented procedural guidance to the SEAR procedure.

Safety Evaluation Summary

This activity involved allowing "minor" changes to the UFSAR Figures such that all systems, structures, and components (SSCs) will continue to perform their intended functions as described in the SAR. This change did not affect existing design bases, safety analyses, or descriptions of existing structures, systems, components or functions described in the UFSAR. This change did not result in the removal of SCCs from the plant that are described in the UFSAR text or Tables or eliminate functions or procedures described in the UFSAR text or Tables. This change did not result in new design bases or safety analyses, or associated descriptions that must be included in the UFSAR. This activity had no effect on any of the accidents evaluated in the SAR. The equipment and the changes that are listed as "minor" are types that are not initiators of any accidents previously evaluated in the SAR. The equipment and the changes that are listed as "minor" are types that are not used to mitigate the consequences of accidents previously evaluated in the SAR. The "minor" changes are also below that level that can affect the design or result in SSC performance degradation. Therefore, this activity did not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR. The "minor" changes screened are such that they are not used to mitigate the consequences and therefore cannot increase the consequences of a malfunction of equipment important to safety. No new failure modes are possible because of the "minor" nature of the changes. Therefore, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The types of equipment being screened are such that they cannot create the possibility of a malfunction of equipment important to safety. The change did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings or safety limits as defined in the Technical Specifications were affected by this change. Therefore, allowance of screening criteria for certain, specific types of "minor" changes to UFSAR Figures did not reduce a margin of safety associated with the basis for any Technical Specification.

SE 99-038 Reactor Core Isolation Cooling (RCIC) System (Steam Side) P&ID Revision

Description and Basis of Change

An electrical connection was erroneously shown on the P&ID for the RCIC System (Steam Side) between the Barometric Condenser (Vacuum Tank) Low-Level Switch, and the Condensate Pump that drains the tank. No such connection actually existed in the plant. The limit switch had an alarm function only (shown on the P&ID) and it did not provide any input to the pump. The pump receives start and stop signals from the High-Level Switch on the tank. The pump can also be tripped through a hand switch from the Control Room, shown on the P&ID, and this is done when an alarm is received from the Low-Level Switch per an Annunciator Response Procedure (ARP). This change only corrected the P&ID by deleting the electrical connection. No change to any plant equipment was involved.

Safety Evaluation Summary

The subject equipment is not an initiator of any previously evaluated accident. This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Deleting the connection between the subject Low-Level Switch on the Barometric Condenser and the pump does not compromise redundancy and does not prevent the pump from performing its safety action. If the High-Level Switch fails to trip the pump when the level reaches a set level, the subject Low-Level Switch gives an alarm and the pump is tripped through a handswitch. Operation of the handswitch is an operator action per the ARP. Therefore, this change did not increase the consequences of an accident evaluated previously in the SAR. The design function of the equipment was not affected because the stop function of the pump is still achieved by two redundant means. Therefore, this activity did not increase the probability of occurrence, or the consequences, of a malfunction of equipment important to safety evaluated previously in the SAR. Because any postulated failure of the subject equipment cannot create an accident, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. No new failure modes that could create any malfunction of the subject RCIC Condenser Condensate Pump were identified. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings, or safety limits are defined in the Technical Specifications for the subject pump or the level switch. This change had no effect on the function of these components in particular or the operability requirements of the RCIC System in general. Hence, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-039 UFSAR Change For Drywell And Torus Sample Valves

Description and Basis of Change

Drywell and Torus atmosphere analyzer suction and return valves were incorrectly listed in an UFSAR Table with the power to open as AC, whereas DC power is used to open these valves to monitor oxygen and hydrogen concentrations and the radioactivity of the containment atmosphere. A reliable power source is needed to open these valves. DC power is considered a reliable and continuous power source.

Safety Evaluation Summary

This change did not increase the probability of occurrence or consequences of an accident evaluated previously in the SAR. Primary Containment Isolation Valves (PCIVs) are not initiators of an accident. A change to the power supply did not prevent the valves from isolating to accomplish their containment isolation function. This change did not increase the probability of occurrence or consequences of a malfunction of equipment important to safety evaluated previously in the SAR. Once isolation is initiated, the valves will continue to close, even if the condition that caused the isolation is restored to normal. This change did not create the possibility of an accident of a different type than evaluated previously in the SAR. No new system failure modes were introduced by changing the power to open from AC to DC. In accordance with the UFSAR, it is acceptable for these isolation valves to be powered by either standby AC power or plant DC power. DC power is considered a reliable and continuous power source and is used to operate the valves for post-accident monitoring. This change did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The design still meets the requirements of the UFSAR. This change did not reduce the margin of safety as defined in the basis for any Technical Specification. Technical Specifications require reliable operation of PCIVs. The ability of the valves to function was not degraded by this change.

SE 99-041 Revise UFSAR And Technical Specification Bases To Show No Emergency Service Water Flow Requirement For The RHR Pump Seal Coolers

Description and Basis of Change

The UFSAR and Technical Specification Bases were revised to show that there are no Emergency Service Water (ESW) flow requirements for the RHR Pump Seal Coolers (i.e. the RHR Pump Seal water does not require cooling). The basis for this change is that the Borg Warner Type 'U' Mechanical Seals are rated up to 450 °F which is above the maximum fluid temperature for all RHR modes of operation. The ESW cooling function was not disabled (i.e. the ESW cooling water piping remains and provides cooling water to the RHR Pump Seal Coolers). However, ESW cooling water is not relied upon for either safety related purposes or non-safety related equipment performance purposes.

Safety Evaluation Summary

The probability of occurrence of an accident previously evaluated in the SAR was not increased because the operation of, or the integrity of, the RHR pump seals are not accident initiators per the SAR. The consequences of an accident previously evaluated in the SAR were not increased because the RHR pump seals will perform adequately without the requirement of ESW cooling water supplying the RHR Pump Seal Coolers. The RHR pumps will be able to perform all accident-mitigating functions as evaluated previously in the SAR. There was no effect on the fission product barriers or dose consequences. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR was not increased because the RHR pumps and seals will perform reliably without the requirement of ESW cooling water supplying the RHR Pump Seal Coolers. The RHR pumps and seals will not be degraded in any way. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR were not increased for the RHR pumps since the probability of RHR pump or seal damage was not increased. The possibility of an accident of a different type than any evaluated previously in the SAR was not created because the RHR Pump Seal Cooler ESW Cooling Water function could only potentially impact RHR Pump Seals and subsequent RHR Pump operation. Since the maximum potential failure is the failure of RHR Pumps, this failure is within previous SAR evaluations. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. Since the only credible malfunction is a seal failure, the possibility of a different or more severe failure was not increased. The margin of safety was not reduced based on a review of the Technical Specifications, Technical Specification Bases, UFSAR and NSOA.

SE 99-042 Change In Carbon Dioxide Suppression System (CARDOX) Compensatory Measures

Description and Basis of Change

This change modified the UFSAR, and the DAEC Fire Plan to change the fire watch requirement for CARDOX impairments from continuous to hourly and remove the statement regarding fire extinguishing equipment for impairment fire watches. The basis for the change is the DAEC's desire to make the fire watch requirements for the CARDOX consistent with other Fire Plan required suppression systems and remove potential confusion regarding fire extinguishing equipment to be used by impairment fire watches.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Fire is not an entry condition, basis or an assumption for any accident previously evaluated in the UFSAR and the NSOA. The changes did not increase the probability of a fire, inadvertent actuation of a fire suppression system or loss of plant equipment credited in the safe shutdown analysis. This activity did not increase the consequences of an accident evaluated previously in the SAR. The changes did not affect fission product barriers. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not changed by this activity. The equipment involved is not safety related and the procedural changes do not increase the likelihood of equipment malfunction. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR were not increased. Fire watches are not credited with limiting the consequences of equipment malfunction. The ability of safety related equipment to perform its function was not affected. Safe shutdown can be achieved independent of fire watches and fire suppression system actuation. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The equipment involved is provided to protect against design basis fires evaluated in the DAEC Fire Hazards Analysis, not against accidents evaluated in the SAR. The changes did not impact the plant's ability to achieve safe shutdown conditions in the event of a fire. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not increased. This change did not reduce the margin of safety as defined in the basis for any Technical Specification. Fire Protection Systems do not form the basis for any Technical Specification safety margins.

SE 99-046 Noble Metal Chemical Addition

Description and Basis of Change

This was the second NobleChem™ treatment at the DAEC. As with the first treatment, the general process was to add a platinum (Pt) and rhodium (Rh) noble metal compound to the reactor water, and then circulate the water inside the vessel for a period of time at a moderate water temperature. Additional compound was injected to replace that deposited and finally, the water was cleaned if necessary. The only significant change to the process was that the potential temperature range for treatment was expanded based on NobleChem™ treatment at other plants. The available temperature range was determined based on plant operational considerations such as the margin to the shutdown cooling isolation pressure setpoint.

Safety Evaluation Summary

The NobleChem™ application did not increase the probability of occurrence of an accident previously evaluated in the SAR. The primary concern regarding the reactor surfaces is whether Pt and Rh could affect the course of an event by its presence on the surface or in the reactor water. Considering catalytic action, mechanical action, heat transfer, fuel clad and temporary mechanical jumpers, this activity did not increase the consequences of an accident evaluated previously in the SAR. The noble metal layer is passive and did not introduce any new equipment that could fail and cause a different type of anticipated operating transient or accident. The application equipment was connected via mechanical jumpers to existing plant piping. The connection point valve positions were controlled by an approved plant Tagout. Connections were made via ½ inch swageloc fittings, therefore considerations associated with the potential for draining the vessel were not applicable. Since the NobleChem™ application provides Intergranular Stress Corrosion Cracking (IGSCC) protection for certain vessel components and piping, the likelihood of a malfunction due to cracking is reduced. Consequently, the NobleChem™ application did not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR. Deposition of Pt and Rh has been evaluated for normal operations and for the large break LOCA. NobleChem™ treatment did not create the possibility of an accident different than previously evaluated in the SAR. The NobleChem™ application did not create the possibility of a malfunction of equipment important to safety different than any already evaluated in the SAR. An increase in conductivity was expected due to the effect of noble metal chemistry during the application period. During and after the application, the Reactor Water Cleanup System continued to operate to remove excess ions from the reactor water and restore the reactor water conductivity limit to its normal range. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-050 Revision To Reactor Water Cleanup System P&ID

Description and Basis of Change

The purpose of this activity was to revise the Reactor Water Cleanup System P&ID to show additional isolation valves and a level gauge glass on the Demineralized Water Supply Makeup lines to the sample station for the Reactor Building Sample System. This change was required to reflect the current plant configuration.

Safety Evaluation Summary

Because the subject process sampling system is not an initiator of any previously evaluated accident, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The subject sampling station is not required for mitigation of an accident. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. Because the components associated with the change are not required for any safety function, and since the applicable codes and standards were met, this activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The addition of the subject components did not change the intended function of the sampling process in any way because the valves are manually operated and the level gauge glass has an indication function only. The system functions as before. Therefore, this activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. No new failure modes that could create any malfunction of the subject sample station were identified. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings, or safety limits are defined in the Technical Specifications for the subject Reactor Building Sample Station. Therefore, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 99-056 Reactor Building Closed Cooling Water System Operating Instructions Revision

Description and Basis of Change

This change revised the Operating Instructions for the Reactor Building Closed Cooling Water (RBCCW) System to allow the operation of all three RBCCW pumps and heat exchangers to increase the heat removal capability of the RBCCW System. The extra heat removal capability was desired for the time when the Shutdown Cooling Mode of the Residual Heat Removal System was secured during RFO 16. The RBCCW System removed some of the heat from the reactor vessel that would otherwise have been removed by the Shutdown Cooling Mode.

Safety Evaluation Summary

Allowing the operation of all three RBCCW pumps and heat exchangers did not increase the possibility of occurrence of an accident evaluated previously in the SAR. The RBCCW System can not initiate an accident, is not a safety related system, and does not serve safety related loads. Allowing the operation of all three RBCCW pumps and heat exchangers did not increase the consequences of an accident evaluated previously in the SAR. The RBCCW System does not perform any accident mitigating function. Although the RBCCW System supports both Fuel Pool Cooling and Reactor Water Cleanup Systems, which perform accident mitigating functions, operating three RBCCW pumps instead of two does not diminish the RBCCW System's ability to support these systems. The probability of occurrence of a malfunction of equipment important to safety was not increased by this change. The increased RBCCW System flow rate and pressure does not negatively impact the loads it serves. The PCIVs may be impacted, but these valves are not required to be operable when the plant is in Modes 4 or 5 which is when the third RBCCW pump would be operated. Furthermore, the PCIVs will not be damaged if inadvertently stroked during this evolution. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and the possibility of an accident of a different type than any evaluated previously in the SAR was not created. The RBCCW System can not initiate an accident and a small change in system parameters (i.e., flow rate and pressure) will not change this fact. This activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. Allowing the operation of a third RBCCW pump and heat exchanger did not reduce the margin of safety as defined in the basis for any Technical Specification. The RBCCW System helps maintain the margin of safety by maintaining the temperatures of reactor coolant via the Reactor Water Cleanup System and the spent fuel pool via the Fuel Pool Cooling System. Operating a third RBCCW pump and heat exchanger when the plant is in the refuel mode can only serve to increase the margin of safety by providing additional cooling than would otherwise be achieved with two pumps and heat exchangers.

SE 99-058 UFSAR and Fire Plan Changes

Description and Basis of Change

This change modified the UFSAR, and the Fire Plan to extend the LCO from seven days to 14 days for the loss of the Diesel Fire Pump or the Electric Fire Pump. It also changed the reporting requirements for fire detection instrumentation inoperability, deluge and sprinkler system inoperability, Cardox System inoperability, extended fire pump inoperability, and twenty-four hour dual fire pump inoperability from a Special Report to the NRC within thirty days to a description of the event(s) within a NRC Monthly Operating Report issued within forty-five days. Also, the required surveillance interval of the Diesel Fire Pump was changed from weekly to monthly. The basis for the change was the DAEC's desire to make the fire pump LCO more risk-informed, to streamline reporting to the NRC, and to reduce unnecessary testing of the Diesel Fire Pump.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. Fire is not an entry condition, basis or an assumption for any accident previously evaluated in UFSAR and the NSOA. The changes did not increase the probability of any of the following events occurring: a fire, inadvertent actuation of a Fire Suppression System or loss of essential HVAC or other plant equipment credited in the safe shutdown analysis. This activity did not increase the consequences of an accident evaluated previously in the SAR. The changes also did not affect fission product barriers. The probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR was not changed. The equipment involved is not safety related and the procedural changes did not increase the likelihood of equipment malfunction. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR was not increased. This activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The equipment involved is provided to protect against design basis fires evaluated in the DAEC Fire Hazards Analysis, not against accidents evaluated in the SAR. The changes did not impact the plant's ability to achieve safe shutdown conditions in the event of a fire, or create any new or different accidents. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR was not increased. This change did not reduce the margin of safety as defined in the basis for any Technical Specification. Fire Protection Systems do not form the basis for any Technical Specification safety margins.

SE 00-002 Revision To Radwaste Sump System P&ID

Description and Basis of Change

This activity revised the P&ID for the Radwaste Sump System, and the Recombiner Building drainage drawing, to show a manually operated floor drain isolation valve and related piping. The change was required to reflect the current plant configuration.

Safety Evaluation Summary

Because the subject Reactor Building Sump System is not an initiator of any previously evaluated accident, this activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The subject floor drain piping is not required for mitigation of an accident. Therefore, this activity did not increase the consequences of an accident evaluated previously in the SAR. Because the components associated with the change meet the applicable codes, this activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The addition of the subject components did not change the intended function of the Floor Drain System in any way because the valve is normally open, manually closed, and the change conforms to applicable codes. The system functions as before. Hence, this activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. Any postulated failure of the subject equipment cannot create an accident because the fluids are contained within the building. Hence, this activity did not create the possibility of an accident of a different type than any evaluated previously in the SAR. No new failure modes were identified, therefore, this activity did not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No safety margins, safety settings, or safety limits are defined in the Technical Specifications for the subject Reactor Building Sump System. Hence, this activity did not reduce the margin of safety as defined in the basis for any Technical Specification.

Section C – Tests and Experiments

This section contains a brief description of a Test completed during the period beginning October 1, 1998 and ending February 29, 2000. The Test was reviewed against 10 CFR 50.59 by the DAEC Operations Committee. The test did not involve an unreviewed safety question. No experiments were conducted during this time period.

SE 99-55 Special Test Procedure (SpTP)-200 – Turbine Cycle Performance Test

Description and Basis of Change

The reason for performing this Turbine Cycle Performance Test was based on an Electrical Power Research Institute (EPRI) Thermal Performance Peer Assessment conducted at the Duane Arnold Energy Center (DAEC) in June 1999. This assessment recommended that a "limited" performance test be conducted at the Duane Arnold Energy Center. This recommendation was based the need to improve the plant heat balance model in order to verify the current plant performance level. In addition, the results of this test may be used to verify the steam turbine cycle enhancement planned for RFO 17. The SpTP installed and removed test instrumentation necessary to support the Steam Cycle Performance Test. It also specified any plant process system line-up changes and/or non-standard operation requirements.

The instrument tubing and tees allowed the highly accurate test equipment to be installed and removed with no expected impact on other systems. This process was performed using normal instrument calibration practices. The following is a list of the measurements:

- Condensate flow
- Condensate Reject flow
- Feedwater flow
- Moisture Separator Reheater (MSR) (1st Stage Heating Steam) flow
- Steam Jet Air Ejector (SJAE) Steam Flow
- Control Rod Drive (CRD) flow
- Reactor Vessel Pressure
- Main Steam Pressures

- HP Turbine1st Stage Pressure
- MSR Heating Steam Pressure
- HP Turbine 4th stage extraction steam pressure
- MSR Pressure Points
- LP Turbine Pressures
- Feedwater Heater Pressure Points
- LP Turbine Extraction Pressure Points:
- Condenser Shell Pressures
- Turbine Exhaust "Basket Tip" Pressure Line Instruments
- Temperature Indications
- Generator Electrical Metering Test Equipment

There were more than one test run, and more than one plant line-up for the tests. These were all performed in accordance with the DAEC Operating Procedures and the Special Test.

All tests were run at a nominal 100% Reactor Power (1658 MWth). The data collection times were planned for at least one hour. The normal plant operating lineup was utilized for three SpTP conditions and condenser isolation line-ups were used for the official test runs.

The major line-up changes included isolation of Condensate Reject and Make-up flows for the condenser. Cycle monitoring of the CRD System and the Condensate Storage Tank (CST) volume is currently the method used for this portion of the heat balance. Other isolations included valves associated with extra steam/heat loads to the Condenser. For example, the isolation of Main Steam Line Drains was permitted for reasonable amounts of time. To adjust Condenser inlet conditions, the number of operating Cooling Tower fans was also controlled during the test.

Safety Evaluation Summary

This activity did not increase the probability of occurrence of an accident evaluated previously in the SAR. The NSOA review identified areas to evaluate such as Turbine Trip/Load Reject with, and without, Bypass and Reactor Pressure Control. The results showed that the probability of occurrence of an accident, transient, or special event previously evaluated in the SAR would not be increased by this Test. The Test used approved procedures and practices to complete the test steps. This Test did not affect the plant operation, or increase the likelihood of a plant trip. The consequences of an accident are unchanged since the Test connections and valve line-ups maintained the pressure boundary and met the appropriate seismic requirements. This activity did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The Test used applicable piping standards and met or exceeded acceptable standards for reading Reactor Pressure. The valve line-ups, including Condenser cycle isolations and Cooling Tower fan operations, are normal operating practices and were evaluated to have no safety significance. This activity did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The consequences of a malfunction of Quality Level 1 equipment were unchanged since the Test connection configurations had only a passive function, met applicable requirements, and valve line-ups were within the scope of normal operating procedures. The possibility of an accident of a different type was not created and the possibility of a malfunction of equipment of a different type was not created. The Test line-ups had no effect on safety functions. The margin of safety as defined in the basis for any Technical Specification was not reduced. The Test configurations or valve line-ups had no impact on safety or safety margin.

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 October 1, 1998 and ending February 29, 2000 did not alter our commitment to the NRC guidelines contained in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

Volume I - Revision 35

This revision changed the DAEC Fire Plan Volume I by revising equipment testing frequency from cyclic to 18 months. This was needed due to future extension of the cycle from 18 months to 24 months. Various editorial changes were also made.

Volume I - Revision 36

This revision made the following changes to the DAEC Fire Plan Volume I:

Extended the Limiting Condition for Operation (LCO) for loss of the Diesel Fire Pump, or loss of the Electric Fire Pump, from seven days to 14 days.

Changed the reporting requirements for extended fire detection instrumentation inoperability, extended deluge, sprinkler, and CO_2 system inoperability, extended fire pump inoperability, and twenty-four hour dual fire pump inoperability from a Special Report to the NRC within thirty days to a description of the event(s) within a NRC Monthly Operating Report issued within forty-five days.

Changed the required surveillance interval of the Diesel Fire Pump from weekly to monthly.

The following Changes were made in the Cable Spreading Room Carbon Dioxide Suppression System (CARDOX) compensatory measures:

- A continuous fire watch was required when the CARDOX System was not operable. This requirement was changed to match the requirements for other suppression systems included in the Fire Plan and UFSAR.
- Removed the implied statement that the fire watch has portable extinguishing equipment in their possession while performing fire watch duties.
- Removed the allowance for personnel in the Cable Spreading room to act as the continuous fire watch.
- Revised Fire Plan bases to reflect the change in fire watch requirements for CARDOX inoperability.

Section E - Commitment Changes

The information contained in this section identifies and briefly describes a commitment change that was made during the period beginning October 1, 1998 and ending February 29, 2000. The change described was evaluated and is being reported per the Nuclear Energy Institute's "Guideline For Managing NRC Commitments", dated December 19, 1995.

AR 15430

In the "Reply to Notices of Violation Transmitted with Inspection Report 92020", corrective steps were identified to prevent the recurrence of missing increased frequency testing for valve stroke time testing. The date of the last Surveillance Test Procedure (STP) performance was added to the monthly increased surveillance letter to assist in supervisory review and verification of the testing schedule. In addition, a meeting with representatives from appropriate departments would be conducted prior to issuance of the monthly increased surveillance letter to discuss testing and component performance and possible corrective actions. These commitments have been deleted. These commitments were made during our ASME 2nd 10 year interval when the code requirements for test frequency increases applied to valves and pumps. The missed surveillance was caused by applying a 45 day pump test frequency to a valve (requiring a 30 day frequency). We are now in our ASME 3rd 10 year interval and code requirements for increased surveillance apply only to pumps. This has greatly reduced the complexity of managing required surveillance test frequency changes. An increased surveillance letter is no longer issued. New administrative controls to streamline the paperwork and development of computer interfaces for modifying surveillance schedules to meet ASME requirements, along with the lessened impact of the 3rd 10 year interval code requirements, make these corrective steps no longer necessary.