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

10 CFR 50.59

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

Dear Mr. Collins:

In accordance with the requirements of Appendix A to Operating License DPR-49, 10 CFR Section 50.59(b), and NUREG-0737 (Item II.K.3.3), please find enclosed the subject report covering the period from October 1, 1995 through March 1, 1997. 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 various commitment changes.

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

Sincerely,

Gary D. Van Middlesworth for JFF

John F. Franz
Vice President, Nuclear

IE47

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Section A - Plant Design Changes

This section contains brief descriptions of plant design changes completed during the period beginning October 1, 1995 and ending March 1, 1997, and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR Section 50.59(b). All changes were reviewed against 10 CFR 50.59 by the Duane Arnold Energy Center (DAEC) Operations Committee. None of the changes involved unreviewed safety questions.

The basis for inclusion of a modification in this report is operational release of the associated modification at the DAEC during the period beginning October 1, 1995 through March 1, 1997. Portions of some of the Design Change Packages (DCPs) and Engineering Change Packages (ECPs) which are listed were partially closed or partially operationally released in previous years.

DCP 1539 Replacement of Seismic Monitors

Description and Basis for Change

This change replaced the three strong motion triaxial accelerographs (SMAs), located in the basement of the reactor building, on the refueling floor of the reactor building, and in the valve house of the "A" cooling tower, with solid state digital recorders. Another change required was the addition of external preamplifiers with each new monitor to drive the common triggering and event signals between the monitors, as well as replacement of some cabling in the reactor building. The previously installed monitors were soon to be obsolete and replacement parts would have been increasingly difficult to obtain. The replacement monitors are essentially a one-for-one replacement for the previous units except the replacement monitors provide the ability for rapid data retrieval and in-house analysis. The modern technology of the new monitors significantly improves the plant's ability to detect, monitor, and analyze the seismic response of the monitored structures during an earthquake.

Summary of Safety Evaluation

The Seismic Monitoring system does not contribute to the probability of an occurrence of an accident as evaluated in the SAR, therefore the probability of an occurrence of an accident previously evaluated in the SAR is not increased. The consequences of an accident previously evaluated in the SAR are not increased because, per the Nuclear Safety Operational Analysis (NSOA), the Seismic Monitoring System is not considered to mitigate the consequences of any accident. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased because the Seismic

Monitoring system does not contain any equipment important to safety nor does it interface with any system or equipment considered important to safety. The consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased because the change only affects the Seismic Monitoring system which does not contain any safety-related equipment nor does it interface with any safety-related equipment. The possibility for an accident of a different type than any evaluated previously in the SAR is not created because the system which this change affects does not contribute to the probability of an accident, and this change does not affect any other system. This change does 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 of the DAEC Technical Specifications, is not reduced because this change affects only the Seismic Monitoring system which is a non-safety related system which has no impact on the margin of safety for any system specified in the DAEC Technical Specifications.

DCP 1553 Thermo-Lag Electrical Modifications

Description and Basis for Change

Certain cables at the DAEC were protected with fire barrier material to allow them to remain intact and operable following a fire. Thermo-Lag 330-1 material was used for this purpose at the DAEC. As a result of uncertainty regarding the adequacy of Thermo-Lag as installed at the DAEC, modifications to eliminate the requirement for fire barrier material were completed. These modifications included the following:

* Rerouting safe shutdown (SSD) cables, required for shutdown in the event of a fire, above the Control Rod Drive (CRD) repair room to minimize the requirement for fire protection.

* Modifying Residual Heat Removal (RHR) system valves to eliminate the potential for spurious operation. These modifications, along with an analysis of the control circuit for another RHR system valve, performed by this DCP, eliminated the need to protect cables associated with the valves.

* The circuits associated with two cables required for Alternate Shutdown Capability (ASC), were modified to properly isolate these cables following a transfer to the remote shutdown panel.

Summary of Safety Evaluation

The cables and circuits modified by this DCP are not identified as initiators for any of the accidents described in the UFSAR or the NSOA. Failure of any of the modified circuits cannot initiate any of these accidents. None of the initial conditions or assumptions for the accidents described in the UFSAR or NSOA are changed by these modifications.

The cable reroutes and circuit modifications meet the design, material and construction standards applicable to these systems. The rerouted cables maintain required divisional separation and do not operate at a higher voltage or amperage than before. The new disconnect switch is qualified for its design environment and is capable of withstanding the maximum credible fault for its location within the Division I AC Power system.

Interim operation of the ASC system places an additional requirement on operations personnel to operate a breaker in lieu of a molded case switch. The requirement for this operation will not initiate any of the accidents described in either the UFSAR or the NSOA. Because both the interim and final activities of this change did not affect the overall system performance in a manner that could lead to an accident, the probability of an accident previously evaluated in the SAR is not increased.

This modification rerouted power and control cables for systems that are required to mitigate several accidents described in the UFSAR and NSOA. The bases for these systems are not affected by this modification. This modification does not prevent or degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents. Only one channel of one division was out of service at any given time, thus there was no possibility for any system to affect its safe shutdown capability.

The ASC system, AC Power, Control Building HVAC, DC Power, RHR Shutdown Cooling and Low Pressure Coolant Injection (LPCI) operate the same as they did before this modification. No logic changes were performed by the MOV circuit modifications . The MOVs perform exactly as before.

The Fire Hazards Analysis (FHA) confirms that the DAEC can be safely shutdown in the event of a fire in the affected fire zones. Sufficient time exists to perform any additional manual actions as a result of the interim configuration of the AC power system. Therefore, the consequences of a design basis fire are not increased and this modification does not increase the consequences of an accident evaluated previously in the SAR.

Since the interim and final modified circuits met the original system bases, electrical power system requirements, separation requirements and seismic requirements, the probability of equipment malfunction was not increased. This modification does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR.

This modification does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This modification does not prevent or degrade any essential safety function assumed by the NSOA to mitigate the consequences of malfunctions or abnormal operational transients. Functional capability of the circuits involved is not adversely affected. The interim system used all existing components. The new disconnect switch is seismically mounted and is qualified for its installed environment. Prior to declaring the equipment operable, appropriate testing was performed to demonstrate that the systems affected by the rerouted circuits functioned the same way as prior to the modification. Malfunction of the revised circuits in both the interim configuration and following the installation of the new equipment did not challenge a fission product barrier more severely than those currently analyzed in the UFSAR and NSOA. Therefore, this modification did not create the possibility of an accident of a different type than any evaluated previously in the SAR.

The interim and final ASCS modifications enhance the ASCS operation in case of a Main Control Room/Cable Spreading Room fire. No new failure modes are introduced by the cable rerouting activity. The MOV circuit modifications rearrange the existing contacts to provide a normally open isolation contact. No logic changes are performed by this contact rearrangement. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR is not created by this modification and the margin of safety as defined in the basis of any Technical Specification is not reduced.

ECP 1559 <u>10CFR50 Appendix R Modifications</u>

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

The Appendix R rebaseline effort redefined fire areas by consolidating various fire zones. The Fire Protection group identified certain circuits required for safe shutdown (SSD), that could have been adversely affected by a fire in the plant. This modification addressed the concerns identified by the Appendix R rebaseline project safe shutdown evaluations.

To resolve the 125VDC coordination issues, fuse panels were added below 125VDC panels. Installation of fuses to the 125VDC system to achieve selective coordination does not affect safe shutdown capability. The fuses were sized to provide selective coordination under maximum fault current conditions at the panels. The new fuse panels prevent the loss of an entire division of DC power from a single fault due to a fire outside the essential switchgear rooms.

Additional emergency lighting was installed in paths used to perform manual actions. These paths include routes between the control room and various areas throughout the plant and adjacent buildings.

Both torus level instruments were located in the same fire area. One of the transmitters was moved to a different fire zone. This involved tapping into the Torus Drain extension nozzle, installing a new root isolation valve, relocating the level instrument and installing new instrument tubing from the new root valve to the relocated instrument. A pressure transmitter providing drywell pressure indication in the control room as well as input to containment level indication was located in the same fire zone as other torus level instruments but has been relocated to a different fire area. A containment level transmitter was also relocated.

The plant fire zones have been grouped into larger fire areas to reduce costs of maintaining and inspecting the fire barriers. As a result of the extensive walkdowns associated with this rebaseline effort, several fire areas were found which were not adequately separated by fire barriers. This portion of the ECP involved sealing one penetration and installing two UL approved three-hour fire doors.

These changes provide the capability of limiting fire damage so that one train of systems necessary to achieve safe shutdown of the plant is free from fire damage.

Summary of Safety Evaluation

Relocation of a torus level transmitter and pressure transmitter, rerouting of the associated cables, installation of fuse panels consisting of properly sized fuses to achieve selective coordination for 125VDC circuits, addition of emergency lighting to perform manual actions for safe shutdown of DAEC, and the fire barrier upgrades for required separation between the fire areas cannot initiate any of the accidents described in the UFSAR or the NSOA.

The addition of a new pressure tap on the Torus Drain does not introduce any pipe stresses that exceed the existing design envelope for the drain

assembly. The new pressure tap meets the design, material and construction standards applicable to this system. The cable reroute and the new cables required for this modification meet the design, material and construction standards applicable to these systems. The new and rerouted cables maintain required divisional separation and do not operate at a higher voltage or amperage than before. The new fuse panels and fuses are qualified for their design environment and will provide proper coordination with the existing breakers in the event of a fault occurring on one of the feeder circuits. Additional emergency lighting installed in the paths from the control room and various plant areas enhances the lighting levels to perform manual actions more safely and efficiently for the safe shutdown of the plant. The barrier upgrades to achieve required separation between fire areas do not impact any activity associated with the safe operation of the plant. Because these activities do not affect the overall system performance in a manner that could lead to an accident, the probability of an accident previously evaluated in the SAR is not increased.

This modification does not prevent or degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents. By relocating one channel of torus level instruments to a different fire area, the availability of the vital parameters is ensured and the single failure criterion is fulfilled, while safe shutdown capability is not compromised. The systems required for mitigating the NSOA and UFSAR accidents operate the same as before the modification. No logic changes were performed by this design change. All circuits modified perform exactly as before. This ECP enhanced the design of the DAEC to mitigate the effects of a fire in three fire areas. This modification does not increase the consequences of an accident evaluated previously in the SAR. Since the modified equipment and circuits still meet the original system bases, electrical and FHA fire zone separation requirements, seismic and environmental requirements, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased.

The relocated torus level transmitters provide the same information in the control room as before. The selectively coordinated fuse panels in the 125VDC system prevents the loss of an entire division of 125VDC power during a fault and provides more reliable 125VDC power. The addition of emergency lighting in the paths from the control room to the various plant areas enhances operators ability to perform manual actions more safely and efficiently, for safe shutdown of the DAEC. The upgrade of various fire barriers provides required separation between newly created fire areas due to consolidation of existing fire zones. Construction Assurance Testing (CAT) and /or Modification Acceptance Testing (MAT) was

performed to provide the highest level of confidence in the system operation and acceptability. Fuses and cables were introduced in the circuits as new components, but these components would not challenge a fission product barrier more severely than those currently analyzed in the UFSAR and NSOA. Therefore, this ECP does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR.

The relocated instruments provide the same information in the control room via differently routed cables. No failure can be postulated by these changes to create an accident of a different type. The safety related class 1E, properly sized fuses introduced in the 125VDC system eliminate the possibility of losing an entire division of 125VDC in case of a fault on a branch circuit. Addition of emergency lighting and fire barrier upgrades does not adversely affect the capability to safely shutdown. Therefore, this modification does not create the possibility of an accident of a different type than any evaluated previously in the SAR, and it does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

The activities performed by this ECP did not change or affect any setpoints or surveillance requirements as specified in the DAEC Technical Specification. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

ECP 1560 10CFR50 Appendix R, Thermo-lag Removal

Description and Basis for Change

Continued use of Thermo-lag fire barrier material to achieve protection against fire, and compliance with 10CFR50, Appendix R requirements was challenged due to the discovery of testing/qualification problems with the Thermo-lag material. Therefore, some existing Thermo-lag needed to be replaced with acceptable material to protect certain safe shutdown circuits against fire and to be in compliance with 10CFR50, Appendix R requirements. As a result, the following modifications were made:

* Safety Relief Valve (SRV), High Pressure Coolant Injection (HPCI) and Residual Heat Removal (RHR) Shutdown Cooling cables enter the drywell via electrical penetrations. These electrical penetrations have been covered with Darmatt material, a three hour fire barrier material. This material was installed based upon IES Utilities testing as well as additional tests performed to ASTM E-119 (Standard Methods of Fire Tests of Building Construction and Materials) and USNRC GL 86-10 Supplement 1 (Fire Endurance

Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area) requirements.

* SRV cables were rerouted through new raceways, such that the raceways wrapped in Thermolag no longer need to be protected. The raceways associated with the cables were wrapped with Darmatt material. The Darmatt covering provides the required 3-hour fire barrier between the raceways and a fire zone.

* Various cables and their associated conduits are now protected utilizing Darmatt fire barrier material.

* Reactor Water Cleanup (RWCU) Inboard Isolation Valve cables were rerouted. Darmatt fire barrier material is not used since these cables are routed outside of the fire area of concern.

* A Lighting Uninterruptible Power Supply (LUPS) was relocated. Both LUPS units, were located in the same fire area on the turbine operation deck. A fire in this area could have made both LUPS units unavailable. One of the units was moved outside this fire area to the battery corridor area. Cables were rerouted to facilitate the new location. Additional light fixtures were transferred to each LUPS unit so that adequate lighting will be provided in the control room for the operators to perform the necessary manual actions. The final configuration of the LUPS units meet both Appendix R emergency lighting and SBO adequate control room lighting requirements.

* Cabling was rerouted to the High Pressure Coolant Injection (HPCI) System Inboard Isolation Valve. The conduits for the valve were wrapped in Darmatt fire barrier material. One of the conduits had to be rerouted to facilitate the installation of the Darmatt fire barrier material.

Summary of Safety Evaluation

The installation of three hour fire barriers (Darmatt) around electrical penetrations and various raceways could not initiate any of the accidents described in the UFSAR or the NSOA. The three hour fire barrier was installed around various systems which include HPCI, RHR Shutdown Cooling, SRVs and electrical penetrations. This installation did not cause any of the stated systems to operate outside their design basis or limits. Darmatt was installed per tested and qualified configurations that meet the Appendix R three hour fire barrier specifications. The installation of

Darmatt fire barrier material around the electrical penetrations has been analyzed and meets all seismic II/I criteria. The final configurations of the divisional conduits that are protected with Darmatt have been analyzed seismically to preclude structural failure of these raceways and their associated circuits during and after a seismic event. Furthermore, the Darmatt insulation around the raceway was analyzed and also meets all seismic II/I criteria.

The rerouting of cables to the RWCU valve, HPCI and Safety Relief Valve systems met the design, material and construction standards applicable to these systems. The rerouted cables maintain required divisional separation. The addition or subtraction of cable does not adversely affect the components operating voltage and amperage. The RWCU, HPCI or SRV system performance was not be affected.

The modification to the control room lighting uninterruptible power supply could not initiate any of the accidents described in the UFSAR or the NSOA. The modification meets the design and construction standards applicable to the DAEC emergency lighting system.

Because this modification did not affect the overall system performance in a manner that could lead to an accident, the probability of an accident previously evaluated in the SAR is not increased.

The HPCI and RHR Shutdown Cooling systems. HPCI and RHR Shutdown Cooling were not modified in a manner that could prevent the systems from mitigating the NSOA and the UFSAR accidents. The NSOA basis for the HPCI and RHR Shutdown Cooling systems are not affected by this modification. Therefore, this installation does not prevent nor degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents.

The HPCI system is required to provide core and/or primary containment cooling during a pressure regulator failure open event, loss of all feedwater event, loss of all offsite power, feedwater controller failure maximum demand event, LOCA (pipe break) inside primary containment event, LOCA (pipe break) outside primary containment event and station blackout event. The HPCI system was not modified in a manner that would prevent the systems from mitigating the NSOA and the UFSAR accidents. Therefore, the rerouting of cable to the HPCI Inboard Isolation Valve will not prevent nor degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents.
The RWCU system is required to provide core and primary containment cooling during a loss of shutdown cooling. Providing core and primary containment cooling, by utilizing the RWCU system, will avoid a fuel

failure calculated as a direct result of the transient analysis. The RWCU system was not modified in a manner that would prevent the system from mitigating the NSOA and the UFSAR accidents. Therefore, the rerouting of cable to the RWCU Inboard Isolation Valve will not prevent or degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents.

The SRV system is required to provide pressure relief to avoid nuclear system stresses exceeding that allowed for transients by applicable industry codes during a Main Steam Isolation Valve (MSIV) isolation, turbine trip/generator load reject without bypass, loss of all offsite power and feedwater controller failure maximum demand. The SRV system is also required to provide core cooling to avoid fuel cladding temperature in excess of 2200° F or peak fuel enthalpy greater than 280 cal/g during a LOCA (pipe break) inside primary containment and LOCA (pipe break) outside primary containment. In addition, the SRV system is required to provide pressure relief to avoid nuclear system stresses exceeding that allowed for accidents by applicable industry codes during a LOCA inside primary containment and a LOCA outside primary containment. The SRV system will not be modified in a manner that will prevent the systems from mitigating the NSOA and the UFSAR accidents. Therefore, this modification will not prevent nor degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents.

The moving of the LUPS to the Control Building Battery Room Corridor will not degrade or prevent any actions described or assumed in an accident discussed in the UFSAR. Therefore, this modification does not increase the consequences of an accident evaluated in the SAR.

The installation of Darmatt around electrical penetrations and raceways cannot initiate any of the accidents described in the UFSAR or the NSOA. Darmatt installation around various systems which include HPCI, RHR Shutdown Cooling, SRVs and electrical penetrations does not cause any of the stated systems to operate outside their design basis or limits. Darmatt was installed per tested and qualified configurations that meet the Appendix R three hour fire barrier specifications. The final configurations of the divisional conduits that are protected with Darmatt have been analyzed seismically to preclude structural failure of these raceways and their associated circuits during and after a seismic event. Furthermore, the Darmatt insulation around the raceway has been analyzed and also meets all seismic II/I criteria.

The rerouting of cables to the RWCU, HPCI, and Safety Relief Valve systems will meet the original design specifications for design, material and construction standards applicable to these systems. Therefore, the modification to reroute cables for the RWCU, HPCI and SRV systems did not reduce system reliability.

The installation of Darmatt around electrical penetrations and various raceways has been analyzed for conduit and cable tray ampacity derating. Installation of Darmatt around various raceways did not affect the associated cables ability to supply components their rated voltage or amperage.

Moving the LUPS unit to the Battery Room Corridor will meet the original design specifications for design, material and construction standards. This modification does not affect the AC Power system and its ability to provide power.

Since the modifications meet the original design basis and electrical separation requirements, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased.

The installation of three hour fire barrier material around HPCI and RHR Shutdown Cooling raceways, the rerouting of cable to the HPCI Inboard Isolation Valve, the rerouting of SRV cables and the installation of a three hour fire barrier around raceways did not create a challenge to a fission product barrier. The moving of the LUPS to the Control Building Battery Room Corridor did not degrade or prevent any actions described or assumed in an accident discussed in the UFSAR. Therefore, this modification did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and it did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The modifications do not introduce any new type of failure mode. Therefore, 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. The activities performed by this ECP do not change or affect any setpoints or surveillance requirements as specified in the DAEC Technical Specifications. The HPCI and RWCU cable reroute, the three hour fire barrier installation around electrical penetrations and around various raceways, and the LUPS unit relocation did not impact any limiting conditions of operation or surveillance requirement. Therefore, this activity does not reduce the margin of safety as defined in the basis for any Technical Specification.

ECP 1562 Reactor Water Cleanup (RWCU) Reactor Vessel Water Level Isolation Setpoint Change

Description and Basis for Change

General Electric Service Information Letter (GE SIL) 131 was issued March 31, 1975 to inform utilities of the opportunity to improve plant performance by reducing the potential for unnecessary RWCU system isolation and Standby Gas Treatment (SBGT) system initiation through the implementation of isolation circuit logic changes. The improvements in RWCU and SBGT performance would be realized by lowering the isolation setpoint from Level 3 (170″ above Top of Active Fuel (TAF)) to Level 2 (119.5″ above TAF). This was recommended because the void collapse that occurs following a reactor scram from a power level above 50% is sufficient to result in an indicated water level below 170″, causing the isolation and initiation. One of the requirements to implement this change was the installation of a RWCU break detection system, to detect a piping break outside primary containment. Inclusion of high ambient and high differential temperature sensing in the RWCU heat exchanger and pump rooms was part of the initial design. Installation of high ambient temperature sensors to provide protection for a break in the RWCU return line piping area was completed during Refueling Outage (RFO) 11. ECP 1562 lowered the RWCU reactor vessel water level isolation setpoint from 170″ to 119.5″ above top of active fuel.

Summary of Safety Evaluation

The reactor vessel water level setpoint at which the RWCU system isolates does not affect the probability of occurrence of a Loss-of-Coolant-Accident (LOCA) or any other accident evaluated previously in the Safety Analysis Report (SAR). The Safety Analysis assumes that the LOCA happens prior to the isolation, which occurs as a result of the LOCA; therefore, the presence (or absence) of the isolation signal and its setpoint at the time the LOCA occurs does not affect the probability of a LOCA.

The primary isolation signal is high ambient or differential temperature and/or high differential flow. Reactor vessel water level provides a redundant isolation signal. Having the RWCU system isolate on low-low reactor vessel water level maintains the operational philosophy of all isolations (except main steam isolation valves) completing prior to emergency core cooling systems (ECCS) initiation, to prevent interference with identification of the leak location and casualty control. For a break inside the drywell, reactor vessel level provides the only RWCU isolation signal; however, in this case, the termination of the leak is not a primary

containment isolation function. Should the inboard RWCU inlet isolation valve fail to shut on receipt of a primary containment isolation signal, the situation is the same as an unisolable small-break LOCA inside containment, which has been previously analyzed and for which protection is demonstrated. Therefore, changing the setpoint does not increase the consequences of a LOCA or any other accident evaluated previously in the SAR.

Although the instruments supplying the level signals have changed, the failure of any single active component results in, at worst, one (of two) containment isolation valves failing to close. An isolation still occurs on loss of AC power. Because the closure of one isolation valve is sufficient to achieve the containment isolation function, valve position indication is available in the Control Room, and the same conditions existed with the previous setpoint and logic, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased 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.

By lowering the RWCU water level isolation setpoint, the RWCU system is maintained available to perform its secondary function as the alternate boron injection path in the event of an Anticipated Transient Without Scram (ATWS). The potential for flooding with water from a RWCU system leak is not increased. For saturated conditions with vessel pressure greater than 40 psig, the high ambient and/or differential temperature sensors will detect the leak and initiate an isolation prior to level reaching the current isolation setpoint (for a leak outside the drywell), demonstrating the redundancy of the water level isolation signal. For saturated conditions with vessel pressure less than or equal to 40 psig, feedwater makeup will mask the leakage, so the water level isolation setpoint will never be reached, and only temperature sensors are available to detect and respond to the leak. For leaks inside the drywell, the amount of flooding would be no more than that produced by an unisolable small-break LOCA. For breaks anywhere in the system during conditions when the vessel is depressurized, the low leakage rate would allow Operations personnel to identify the source of the leak and take the actions necessary to terminate the event. Manual control of valve position remains available and unaffected by this change, as long as motor operator power is available. Therefore, the possibility of an accident of a different type than any evaluated previously in the SAR is not created.

Technical Specification change, Amendment 217, was approved by the NRC on August 8, 1996, to revise the RWCU reactor vessel water level

isolation setpoint. With this setpoint lowered from 170″ to 119.5″ above TAF, analysis shows that the isolation still occurs at a water level substantially greater than 18.5″ above TAF, which represents the margin to the safety limit of 12″ above TAF. Provided the RWCU isolation is initiated by any of the sensors (ambient temperature, differential temperature, and differential flow), the amount of fluid lost and the increase in reactor building and offsite radiation levels is less than for a main steam line break, which is the limiting accident. Therefore, the margin of safety as defined in the Technical Specifications for water level as well as on-site exposure levels is not reduced.

ECP 1566 Feedwater Control Enhancements

Description and Basis for Change

This modification was intended to eliminate obsolete components, improve system reliability, and add flexibility for future feedwater control enhancements. This project removed steam flow and reactor level density compensation, replaced GEMAC controllers with Moore units, removed unused and obsolete components, and added computer points for trending and monitoring.

Summary of Safety Evaluation

The GEMAC controllers were replaced with digital Moore controllers to avoid obsolete component failures and to remove difficult to calibrate instrumentation. The new controllers are more reliable and are less susceptible to failures such as loss of control signal. The revised power supply to the control components reduces the potential of a controller failure from a common power supply failure mode. The other feedwater control modules in the system were upgraded to more reliable and accurate units. Therefore, the feedwater control enhancement project did not increase the probability of occurrence of an accident evaluated previously in the SAR.

The Feedwater Control system controls the feedwater flow into the vessel via the positioning of the feed regulating valves. This controlling function was not changed by this ECP. The maximum feedwater flow from a controller failure, as defined in the core reload analysis, is determined by the design of the feed regulating valve disk stack and cannot be exceeded by any failure of the Feedwater Control system. The failure of the control signal of either a maximum or minimum has been addressed in the UFSAR. Therefore, this ECP did not increase the consequences of an accident previously evaluated in the SAR.

The Feedwater Control system is a non-safety system and does not provide any input to any safety related equipment or system. The ECP work scope did not change the failure modes identified in the UFSAR. This ECP reduced the probability of a system transient from a component failure. Component number reduction in the control system, and the change in power supply, reduces the potential of a component failure. Therefore, this ECP did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR, and it did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR.

The loss of a feed, steam, or level signal is no different in transient response now as compared to a loss of any of these signals in the previous configuration. This modification replaced obsolete and difficult to calibrate components with new components some of which are now digital instead of analog. The overall function of the Feedwater Control system is not changed and therefore does not create the possibility of an accident of a different type than any evaluated previously in the SAR. This ECP 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 Feedwater Control system is not identified in the Technical Specifications, however some of the instrumentation is identified in a Technical Specification table. The basis for this section only identifies the instrumentation calibration frequency. There is no Technical Specification safety basis identified for the Feedwater Control system. Therefore, this ECP does not reduce the margin of safety as defined in the basis for any Technical Specification.

ECP 1567 Offgas Scram Frequency Reduction - Electric Heater Removal

Description and Basis for Change

As part of the ongoing effort to increase plant availability, the Offgas and Recombiner system was evaluated to determine what modifications would reduce the number of possible Offgas related scrams or forced shutdowns. As a result the following modifications have been made:

* Bypass lines were installed around the Preheater Steam Supply Pressure Control Valves (PCV)s. This modification provides isolation capabilities for these valves, allowing the system to remain operational while maintenance is performed on-line, if the need were to occur.

* The dew point temperature function was removed from the electric heater outlet in the Offgas system. This was accomplished by decommissioning a moisture element and moisture transmitter. The black pen of the moisture recorder had its inputs removed and was de-energized. The associated annunciator was also disconnected. These instruments had many maintenance related problems and were not accessible for maintenance while the plant was in operation, resulting in inaccurate and unreliable information being supplied to the moisture recorder.

* The two pressure switches which provided trip signals to the Steam Jet Air Ejector (SJAE) Offgas Discharge Header Isolation Control Valve were determined not to be needed and were removed from service. They were also the initiating devices for an alarm on a control room panel, which was also eliminated. The system original design was to protect an Offgas system without recombiners

* A pressure switch which provided a redundant isolation signal to the Offgas system on low steam supply pressure, to prevent explosions in the Offgas system by ensuring proper dilution flow to the Offgas stream, was determined to be unnecessary and was removed. Offgas dilution flow is provided by the Offgas jet compressor, and the low steam flow isolation is provided by the Offgas Jet Compressor Low Steam Flow Switch. The logic supplied by this switch was modified to a two-out-of-two energize to actuate configuration using a low pressure signal from the Offgas Jet Compressor Steam Pressure Switch. The new logic helps prevent spurious isolations of the Offgas system. The Offgas Jet Compressor Steam Pressure Switch was replaced with a dual acting switch to accommodate the logic modification.

* The trip function of the Closed Cooling Water Supply/Return Header Low Pressure Differential Indicating Switch was removed. This reduces the chance of a spurious Offgas isolation and resultant Main Turbine trip, while not significantly jeopardizing operation of the Offgas system, since the Offgas Recombiner Closed Cooling Water Low Flow Switch will act to start the standby closed cooling loop pump on a low flow signal, and in turn annunciates an alarm in the control room.

* Chain link operators and reach rods for the SJAE first stage suction valves were replaced with manual handwheels. The operators had a history of binding and were difficult to operate. To accommodate easy access to the modified valves, rolling platforms were also located in the area.

Summary of Safety Evaluation

The bypass line installation, installation of valve handwheels and system circuitry modifications will not prevent the Offgas system from performing its design function. The Offgas system modifications cannot initiate any of the accidents described in the UFSAR or the NSOA. The Offgas system is not safety related. Because the modification does not affect any system performance in a manner that could lead to an accident, the probability of an accident previously evaluated in the SAR is not increased. This change will not prevent nor degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents. Therefore, this modification does not increase the consequences of an accident evaluated in the SAR.

Single failure proof criteria do not apply to the Offgas system. The PCV bypass lines were designed to the same standards as the original system and will withstand the same transients as previously analyzed. The remaining modifications either removed unnecessary trips or improved trip logic to prevent spurious trips. System reliability is enhanced as a result of these modifications. Since the modified process piping and circuitry still meet the original design basis and specification requirements, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased.

The piping and instrumentation modifications will not affect the offsite doses as a result of a malfunction. The bypass line around the preheater PCV will only be used to provide manual steam pressure control to the preheater in the event of the PCV failing. The bypass line will normally be isolated. The manual valve is less likely to fail than the PCV and would have the same results as the failure of the PCV. Because of this and the bypass line being designed to the same standards as the original system, the consequence of a malfunction is not increased. The possibility of an accident of a different type than any evaluated previously in the SAR is not created by this modification.

System logics which were removed have redundant trips in place or were removed because an actual transient would not have caused them to actuate. The Offgas isolation based on jet compressor steam flow was made a two-out-of-two logic to prevent spurious isolations. This logic modification will still perform its intended function on low steam flow. Therefore, this modification does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR.

The failure modes of the PCV bypass lines have been evaluated for impact on the Offgas system. No new mechanical failure modes have been introduced that will affect equipment important to safety. The logic changes performed did not impact equipment important to safety. The jet compressor low steam flow trip is the only logic modification that could possibly malfunction. The results of a malfunction in this logic is the same as prior to the modification. Failure of the Offgas Jet Compressor Steam Pressure Switch or the Offgas Jet Compressor Low Steam Flow Switch would prevent an Offgas isolation on low jet compressor steam flow (which is the same as the prior failure of the Offgas Jet Compressor Low Steam Flow Switch). Therefore, the modification does not introduce any new failure modes into the Offgas circuitry. Therefore, this modification did not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

This modification did not change or affect any setpoints or surveillance requirements as defined in the Technical Specifications. Therefore, this modification does not reduce the margin of safety as defined in the basis for any Technical Specification.

ECP 1569 SBGT Heater Temperature Element and Switch Removal

Description and Basis for Change

This modification removed Heater Sheath Temperature Elements and Switches from service in the Standby Gas Treatment (SBGT) system. These elements and switches were obsolete and no longer manufactured. Design basis documents, system operation and design documents were reviewed for design requirements and operational functions applicable to the SBGT electric heaters. The temperature elements and switches were removed based on the following:

* Redundancy of overtemperature protection.

* Alarm annunciation for high outlet temperature conditions in the control room.

* Overtemperature protection meets ASME N509-1989 without sheath temperature element/switches.

* Manual reset switches are not recommended, per ASME N509-1989.

* SBGT fan must be energized to energize electric heater.

* SBGT air flow greater than 900 CFM to energize electric heater.

Summary of Safety Evaluation

The electric heaters are provided in the SBGT system to reduce the relative humidity of the incoming air to less than 70 %. Removal of the Heater Sheath Temperature Elements and Switches, does not prevent the SBGT electric heater from performing its design function. The removal of the heater temperature elements and switches from the SBGT system cannot initiate any of the accidents described in the UFSAR or the NSOA. Because these activities did not affect the overall system performance in a manner that could lead to an accident, the probability of an accident previously evaluated in the SAR is not increased.

The SBGT system is required to establish secondary containment integrity during a fuel handling accident, Loss-of-Coolant-Accident (LOCA) inside primary containment and LOCA outside primary containment. Establishing secondary containment integrity, by utilizing the SBGT system, will avoid a radioactive material release that exceeds 10 CFR 100 limits. The SBGT system was not modified in a manner that would prevent the system from mitigating the NSOA and the UFSAR accidents. The NSOA basis for the SBGT system is not affected by this modification. Therefore, this ECP does not prevent nor degrade any essential safety function assumed by the NSOA to mitigate the consequences of design basis accidents. This modification did not increase the consequences of an accident evaluated previously in the SAR.

Redundancy exists for overtemperature protection. The modification to the heater circuitry still meets the original design specification installation and separation criteria. The SBGT system is a single failure proof system. The modified circuitry does not affect the SBGT system and its ability to meet single failure criteria. Since the modified circuitry still meets the original design basis and electrical separation requirements, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased. The removal of the temperature elements and switches will not challenge a fission product barrier more severely than those currently analyzed in the UFSAR and the NSOA. Therefore, this modification did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. No failure can be postulated by this modification that will create an accident of a different type. This modification did not create the possibility of an accident of a different type than any evaluated previously in the SAR.

This modification does not introduce any new failure modes into the SBGT electric heater circuitry. The SBGT system is divisionalized such that a single failure will not result in the loss of both divisions. This modification did not introduce any new failure modes that have not been previously identified and evaluated in the NSOA or UFSAR. This modification 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 activities performed by this ECP did not change or affect any setpoints or surveillance requirements as specified in the DAEC Technical Specification. The SBGT heater control logic modification did not affect the heater output capability or impact the limiting conditions of operation duration. Therefore, this modification does not reduce the margin of safety as defined in the basis for any Technical Specification.

ECP 1573 Noble Metal Chemical Addition Monitoring Equipment

Description and Basis for Change

The Noble Metal Chemical Addition (NMCA) process has the advantage of providing Intergranular Stress Corrosion Cracking (IGSCC) protection to all noble metals treated surfaces at low hydrogen injection rates with minimal impact in plant operating dose rates. The process also provides IGSCC protection for a greater number of in-vessel components than with increased Hydrogen Water Chemistry (HWC) alone. Due to this effect and the added benefits of NMCA, DAEC elected to implement the NMCA process. An adverse effect of increased HWC is an increase in on-line dose rates due to increased N_{16} in the steam phase. A modification to install monitoring equipment was not required in order to perform the NMCA process, but was required to monitor the effects of the process. Following Refueling Outage 14, benchmark testing of hydrogen injection levels was completed to determine the optimum injection rates corresponding to NMCA.

This project installed equipment required to monitor the effectiveness and efficiency of the NMCA treatment. The following equipment was installed:

* An in-pipe electrochemical potential (ECP) monitor on the 'A' recirculation line suction decontamination flange replaced a blind flange at the connection. Sensors in the monitor include four working electrodes and four reference electrodes.

* In-core monitoring was provided by two modified Local Power Range Monitoring (LPRM) assemblies. An ECP-LPRM was installed in core location 32-09. This ECP-LPRM is a fully

functional LPRM string with four LPRM detectors and a full length traversing in-core probe (TIP) calibration tube, and six ECP probes. Electrical connections to the ECP sensors were made with existing plant wiring installed under-vessel.

* A double cantilever beam (DCB)-LPRM was installed in core location 16-41. This assembly contains four LPRM detectors, and one bellows loaded DCB crack growth sensor at the core support plate elevation. This LPRM string does not contain a TIP calibration tube. The bellows is hydraulically expanded to maintain a constant load on the DCB. A 1/8″ capillary tube is connected to the LPRM string, and is routed outside of primary containment to a load frame. The load frame provides the hydraulic force necessary to load the DCB sensor. This configuration requires a primary containment penetration designed for a 1/8″ capillary tube. The capillary tubing is seismically supported and field routed from underneath the vessel, through the drywell to a spare 1″ pipe penetration. One manually operated isolation valve was installed at the outboard connection to the drywell penetration. The capillary tube outboard of the isolation valve, as well as the load frame are non-safety related. Electrical connections to the DCB sensor was made with existing plant wiring installed under-vessel.

* The missing TIP at core location 16-41 required that the TIP guide tube be capped under vessel. To prevent inadvertent use of this TIP channel the shorting screws were set to establish this channel as a "spare". An Operating Instruction procedure (OI) was revised to instruct operators not to scan this channel. LPRM detector calibration is accomplished within 3D Monicore using symmetrical core TIP data, or core modeling data.

Summary of Safety Evaluation

The LPRM assemblies and the recirculation in-pipe monitor provide a safety function for maintaining reactor coolant pressure boundary. The analyzed event evaluated is a loss of coolant accident resulting from a failure of these components. The design, fabrication, installation, and testing of these components was performed in accordance with ASME Boiler and Pressure Vessel Code Section III, Class 1. Material requirements meet the same code. Seismic Category 1 requirements are met. These requirements meet or exceed the original codes and standards for plant design. Based on this, the change does not increase the likelihood of a loss of coolant accident.

The 1/8″ capillary tube, primary containment penetration, and isolation valve provide a safety function for maintaining primary containment integrity. The bounding event for instrument lines is an unisolated break of a single instrument line into the secondary containment. The tubing, penetration, and isolation valve were designed, installed, and tested in accordance with ASME Section III, Class 2. This ensures quality and safety commensurate with other instrument lines that penetrate primary containment. A failure of the new instrument line is no more likely than a failure of existing instrument lines. Therefore, the modification does not increase the probability of occurrence of an accident previously evaluated in the SAR.

The reactor coolant pressure boundary components of the LPRM assemblies and the recirculation flange are essentially identical to existing plant components. The highest level codes and quality requirements ensure that the integrity of the reactor coolant pressure boundary is maintained. The plant safety analysis evaluates a full spectrum of primary system leaks, from small leaks to the Design Basis Accident - Loss of Coolant Accident (DBA-LOCA). The consequences of a leak from the new components and the resulting loss of reactor coolant is no more severe than previously analyzed. The consequences of an unisolated instrument line break are analyzed in the UFSAR. This analysis assumes that a 1″ instrument line with a 1/4″ orifice breaks outside primary containment and discharges into the secondary containment. This analysis concludes that 10CFR100 release limits are not exceeded. The magnitude of this event is significantly greater than a postulated break of the 1/8″ capillary tube. Therefore, the existing accident analysis is the bounding event for an instrument line break. Nuclear safety requirements for primary containment integrity are maintained, therefore, the ability of the primary containment to mitigate the consequences of an accident is not affected. This modification does not increase the consequences of an accident previously evaluated in the SAR.

The recirculation flange assembly installed is essentially identical to the blind flange previously installed. The possibility of a malfunction of the flange assembly is no more likely than that of the previously installed blind flange. All pressure boundary components meet or exceed the design and quality requirements of the Reactor Recirculation system. Seismic qualification, loose parts, and flow induced vibration have been addressed and documented in the design specification.

The design of the modified LPRM assemblies is essentially identical to that of standard LPRMs. The pressure retaining portions of the LPRM strings meet or exceed the design and quality requirements of the reactor

coolant pressure boundary. The possibility of a malfunction of the pressure boundary portion of the LPRM is no more likely than that of a standard assembly. Individual LPRM detectors within the assembly are standard in design.

The capillary tube installation results in a new penetration through primary containment. This results in new equipment that is relied upon to maintain the primary containment boundary. The design of the tubing, penetration and isolation valve is in accordance with the codes, standards, and quality requirements for primary containment. The probability of the new instrument line failing is not greater than other similar lines. Therefore, the modification does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

A failure of the pressure boundary components of the LPRM assemblies or the recirculation in-pipe monitor will not result in more severe consequences. To maintain the required safety related neutron monitoring, the minimum number of LPRM inputs must be maintained. Failure of any LPRM detector requires plant operators to take necessary actions specified in Technical Specification Limiting Conditions for Operation. Therefore, the consequences of an LPRM failure are not increased. The primary containment penetration utilizes a design detail that is different from previous plant design. The design requirements are less stringent, however, compliance with design requirements specified in the Design Safety Standards ensures that the highest level nuclear safety requirements are still met. The existing safety analysis evaluates the consequences of an unisolated instrument line break into the secondary containment. The consequences of a worst case failure of the new capillary tube are significantly less than this bounding analysis. Therefore, this modification does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

Based on the modifications, the types of credible events that could result are a loss of reactor coolant inside primary containment, primary containment and/or reactor coolant leak into secondary containment, and loss of inputs to the power range neutron monitoring system. These events are adequately addressed in the plant safety analysis. Therefore, this ECP does not create the possibility of an accident of a different type than any evaluated previously in the SAR.

The design of the equipment is consistent with existing plant design bases. The capillary tube is an added feature that is somewhat unique in its application at DAEC. Multiple failures must occur in order to cause a reactor coolant leak into the primary or secondary containment. Although

the capillary tube is a new design, it is comparable in function and design to other instrumentation lines and their associated accident analysis. The ability to manually isolate the capillary tube meets established nuclear safety criteria, and automatic isolation capability is not required. No new equipment failure modes are created by this modification. This ECP does not create a possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR. This ECP does not reduce the margin of safety as defined in the basis for any Technical Specification. The design of the recirculation in-pipe monitor meets the pressure requirements for reactor vessel design pressure. The installation of the modified LPRM assemblies had no impact on the Average Power Range Monitoring (APRM) system or Rod Block Monitoring (RBM) system operability. Neutron monitoring bases in the Technical Specification states that the LPRM detectors will be calibrated every 1,000 effective full power hours using TIP traverse data. For LPRMs at core location 16-41, calibration will not directly use TIP traverse data. The calibration method uses either core symmetrical TIP data or values derived from full core modeling. Both of these alternate calibration methods are acceptable, and have previously been utilized at the DAEC. This has no impact on the neutron monitoring margin of safety as defined in Technical Specifications. The design of the capillary tube, primary containment penetration, and isolation valve comply with the requirements for primary containment integrity specified in the Technical Specifications.

ECP 1575 High Pressure Coolant Injection (HPCI) Exhaust Steamline Modification

Description and Basis for Change

This ECP removed the HPCI system exhaust steamline swing check valve from its location on the HPCI exhaust line and re-installed it closer to a plug lift check valve on the HPCI exhaust line, so that the length of pipe between the two check valves (and therefore, the volume available for vacuum conditions to exist and the volume available for water to become trapped in) was reduced from 58 feet to 6 feet. A new pipe piece was installed at the void created by the removal of the check valve. All the supports (hangers/restraints/snubbers) on the line were reevaluated for their structural integrity and modified as necessary.

The intent of this modification was to reduce the volume available for water to accumulate in the exhaust line. Reducing the amount of water that can accumulate in the volume between the two check valves creates a situation where the turbine exhaust steam that enters this volume upon subsequent turbine operation will be able to expel the accumulated water in a significantly reduced amount of time. The water is expelled before

the resulting exhaust line pressure can increase to a level that would otherwise adversely impact system operation. The pipe loads that can be caused by the expulsion of water trapped between the two valves is significantly reduced. Less water plug formation reduces reaction forces on the pipe bends and turns.

Summary of Safety Evaluation

The HPCI system provides safety functions of core and primary containment cooling under abnormal operational transients of loss of feedwater and offsite power. HPCI also provides this safety function under accident conditions of loss of coolant accident (pipe break) inside and outside primary containment. The design, fabrication, and testing of this modification were performed in accordance with applicable codes and standards. Material requirements meet the same code as originally constructed. Seismic Category 1 requirements continue to be met. These requirements meet or exceed the original codes and standards for the plant design. Based on this, the change does not increase the probability of occurrence of an accident evaluated previously in the SAR.

The same check valve was used which continues to provide positive actuation for immediate isolation in the event of a break upstream. This check valve has been maintained in the condition to provide primary containment boundary. Its relocation had no impact on the existing pipe support on the line. All pipe supports are qualified as Seismic Category 1 structures in accordance with ASME Section III. The nuclear safety requirement for primary containment integrity is maintained, therefore, the ability of the primary containment to mitigate the consequences of an accident is not affected. Therefore, this modification does not increase the consequences of an accident evaluated previously in the SAR.

The new piece of pipe installed at the old location of the check valve was procured to the original specifications. Installation was to the original construction code. The pipe and the pipe supports were analyzed as Seismic Category 1 Structures in accordance with ASME Section III. This modification decreases the probability of a malfunction of equipment important to safety, due to the reduction in volume between the two check valves, and the resulting reduced pressure transient and reduced reaction forces. This modification reduces the consequences of water plug formation from water trapped between the check valves. Therefore, this modification does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR.

There is no new malfunction associated with the check valve since the same valve is being utilized. The consequences of a malfunction of equipment impacted by this modification are not increased because HPCI is backed up by the Automatic Depressurization system and the low pressure Emergency Core Cooling systems, which are not affected by this modification, and leakage past primary containment is contained by the secondary containment, which is unaffected by this modification. Therefore, this modification does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR.

Relocation of the check valve has no adverse impact on the structural integrity of piping, pipe supports, turbine and torus nozzles due to loads induced by transients, deadload, earthquake, and thermal effects. This modification will not create the possibility of an accident of a different type than any evaluated previously in the SAR.

The modified configuration of the HPCI steam exhaust line is consistent with the existing plant design basis. Its intended function of discharging steam exhaust from the turbine to the suppression pool is not impacted. The new configuration is no more susceptible than the previous configuration to drawing water from the torus following turbine operation. The amount of water that can be trapped between the check valves in the new configuration is greatly reduced, thus resulting in less high drain pot alarms and reduced water hammer loads on the exhaust line piping and check valve. The check valve continues to perform its intended function. This modification will not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR.

This modification has no impact on the HPCI system's safety function of maintaining adequate coolant inventory for core cooling and the system will continue to perform its intended function based on the specified low level scram and initiation setpoints. This modification also has no impact on the trip level setting of the HPCI turbine exhaust diaphragm high pressure. The relocation of the check valve does not have any impact on its function of providing primary containment isolation since the same check valve is being used at its designed location. This modification will not reduce the margin of safety as defined in the basis for any technical specification.

ECP 1577 Rod Block Monitoring Modification

Description and Basis for Change

GE Services Information Letter (SIL) 599 describes an undocumented failure mode with the Rod Block Monitor (RBM) subsystem of the Neutron Monitoring System. SIL 599 states that either a blown fuse (F29) in one of the 5-volt power supplies (PS19) or a failure of the power supply itself could potentially allow the reactor operator to withdraw a control rod further than permitted by the design analysis. Testing indicated that the failure mode described in SIL 599 is applicable to the DAEC. To restore the system to meet its original design intent, the Rod Block Monitor (RBM) subsystem was modified to protect against failure of a 5-volt power supply or a blown fuse. Both channels of the RBM subsystem were modified. This consisted of wiring an existing spare trip unit to monitor the output of PS19.

Summary of Safety Evaluation

The RBM is not adversely affected by this modification. The ability of the RBM to automatically block the rod withdrawal as described in the SAR is not compromised by this modification. The only accidents initiated by withdrawal of control rods are the Rod Withdrawal Error (RWE) and Control Rod Drop Accidents. These accidents are analyzed in the SAR. The RWE requires withdrawal of a rod out-of-sequence while Control Rod Drop Accidents require decoupling of the control blade. Modifications made by this change package do not affect these analyzed accidents in any way. The new relays introduced by this activity eliminate the potential for inadvertent rod movement by the operator in the scenario postulated by the SIL and do not have any adverse effect on the overall operation of the RBM system. The components added by this ECP maintain the original requirements for electrical separation between RBM channels and meet all design, material, environmental suitability, and construction acceptance criteria. Because this activity did not affect the overall system performance in a manner that could lead to an accident, the probability of occurrence of an accident previously evaluated in the SAR is not increased.

This modification installed two relays, bias resistors, and internal wiring changes within a control room panel for the RBM subsystem of the Neutron Monitoring System. Contacts from the power supply relays were wired into each channel to monitor the power supply failure in either of the channels. This modification has no adverse effect on the requirements of the RBM system to function to mitigate the accident described in the UFSAR. None of the bases is adversely affected by the activity of this

modification. The RBM system still meets the requirements of the bases defined in the Design Basis Documents (DBD)s. This modification does not increase the consequences of an accident evaluated previously in the SAR.

The RBM subsystem is non-safety related. The capability of the RBM subsystem to supply a trip signal to the reactor manual control system (to inhibit control rod movement) is not compromised by this modification. The modified circuits utilize relays which are hermetically-sealed, suitable for mild environment, and provide coil-to-contact isolation. The wiring method used within the cabinet is consistent with the existing RBM design. The installation of the relays and the associated wiring meets the required separation criteria. Because the modified circuits meet the original system design bases, electrical separation, and environmental requirements, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased.

The consequences of a malfunction of equipment important to safety previously evaluated in the SAR are not increased because the ability of the RBM subsystem to perform its function is not adversely affected by this modification. Testing following the implementation of this modification assured that the RBM subsystem will perform its function. If the coils of the relays used in this application were to open, the INOP signal would create a rod block signal in both the channels in the fail-safe condition. Thus, the RBM is designed to be "fail-safe". Because both channels, instead of one, respond to a single power supply failure, the RBM system is made more conservative and hence does not affect any design bases. The relays used are suitable for the application in the mild environment of the control room and will operate in all design-basis events. Therefore, this modification does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR.

This modification adds a power failure monitoring relay in each channel of the RBM subsystem to provide an INOP signal in case of a power supply failure or a blown fuse such that rod withdrawal can be blocked. The RBM subsystem's design function is to prevent localized fuel failures due to a Rod Withdrawal Error (RWE) accident, which have been previously analyzed. This design function is not changed in any way by this modification. The RBM logic changes performed by this activity make the existing system meet all functional and separation requirements. The components added by this modification by themselves cannot cause any accident. Thus, no failure can be postulated by these changes to create an accident of a different type. This modification does not create the

possibility of an accident of a different type than any evaluated previously in the SAR.

The modified circuit provides an INOP signal to block rod movement in case of a power failure or a blown fuse. This activity eliminates the potential for an unmonitored inadvertent rod withdrawal upon loss of the power supply or a blown fuse in any channel of the RBM. The relays used are suitable for the application in the mild environment of the control room and will operate in all design-basis events. This installation does not have an adverse effect on any other system or component. Therefore, this modification does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

None of the setpoints or surveillance requirements set forth in the DAEC Technical Specifications are affected by the activities of this modification. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

ECP 1578 Pressure Relief Between MO1908 and MO1909

Description and Basis for Change

The purpose of this ECP is to prevent overpressurization of the piping between MO1908 and MO1909. These are the containment isolation valves for the piping which connects the "B" Recirculation Loop pump suction piping to the Residual Heat Removal (RHR) system pump suction piping. This piping is used for the Shutdown Cooling Mode of operation. MO1908 is the inboard isolation valve and MO1909 is the outboard isolation valve. These valves are normally closed during normal plant operation.

This ECP installed 1/2" piping and valves (with appropriate pipe supports) which provide a flowpath around MO1908. This flowpath contains a check valve which allows flow from the piping between MO1908 and MO1909 to upstream of MO1908 and prevents flow in the opposite direction. This check valve is considered an inboard containment isolation valve. This new arrangement does not allow the piping between MO1908 and MO1909 to be at a higher pressure than the recirculation piping to which MO1908 is attached.

This configuration eliminates the potential for overpressurization of the subject piping. This installation was designed in accordance with all of the required mechanical and seismic requirements necessary for safety

related equipment. With respect to single failure criterion, only one check valve was installed because the failure to open is considered a non-credible failure. The non-credible description is predicated upon adequate testing of the check valve per the Inservice Testing (IST) Program.

Summary of Safety Evaluation

The probability of occurrence of an accident previously evaluated in the SAR is not increased. The loss of RHR Shutdown Cooling Mode is not an accident previously analyzed in the SAR, because it is not considered to be an accident. Reactor coolant pressure boundary items are postulated as rupturing and are evaluated in the SAR. Since the piping and valves installed per this ECP, which are considered part of the reactor coolant pressure boundary and subject to the possibility of rupturing, meet all required design acceptance criteria, the probability of occurrence of a pipe rupture is not increased.

The consequences of an accident previously evaluated in the SAR are not increased because the installed piping and valves meet all design, pressure and temperature requirements and the check valve arrangement meets primary containment isolation integrity design and licensing requirements. The piping and valves installed for this ECP were tested to verify they could adequately perform their intended function. There is no adverse effect on shutdown cooling operation. Therefore, this modification does 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 previously evaluated in the SAR are not increased because the piping between MO1908 and MO1909 is less susceptible to overpressurization, the installed piping and valves do not adversely affect the capability of MO1908 or MO1909 to perform their isolation functions, and the overall isolation capability for the containment penetration is not adversely affected. The check valve is considered to be an inboard containment isolation valve with MO1909 being the outboard containment isolation valve; thus, containment integrity is adequately maintained even with a single failure of MO1908 or MO1909. With respect to single failure criterion (for the open direction only, since the single failure for the closed direction utilizes MO1909 as the outboard containment isolation valve), only one check valve was installed because the failure to open is considered a non-credible failure. The determination that failure to open is non-credible is predicated upon adequate testing of the check valve per the IST Program. This modification does not create the possibility for an accident of a different type than any evaluated previously in the SAR, and

it does 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 not affected because the installed piping and valves meet all design, pressure and temperature requirements, the check valve arrangement meets primary containment isolation integrity requirements and the installed piping and valves do not adversely affect the isolation capability of MO1908 or MO1909. The margin of safety is actually increased because the piping between MO1908 and MO1909 is less susceptible to overpressurization and subsequent failure. Therefore, this modification does not reduce the margin of safety as defined in the basis for any technical specification.

Section B - Procedure/Miscellaneous Changes

SpTP 197 Noble Metal Chemical Addition

Description and Basis for Change

Special Test Procedure (SpTP) 197 was written to provide the plant conditions needed to perform Noble Metal Chemical Addition (NMCA) application and sampling of reactor water and sampling coupons during the application. NMCA employs the reactor coolant as the transport medium to deposit minute amounts of noble metal material on all wetted reactor components. With stoichiometric excess hydrogen, the corrosion potential decreases dramatically and crack initiation and growth are greatly reduced, even at high oxygen and hydrogen peroxide levels. The NMCA process was applied during the normal cooldown sequence prior to Refueling Outage 14.

Summary of Safety Evaluation

The vessel, Recirculation system, Reactor Water Cleanup system and the Residual Heat Removal system components were treated. The treatment did not increase the probability of a component to fail. The control rod drop accident was not affected since control rod drive mechanisms other than the control blades and guide tubes were not treated. Since the thickness of the layer is far less than the manufacturing tolerance of the blades and guide tube, the layer does not interfere with CRD operation. Since the NMCA treatment does not affect electrical or electronic components and does not degrade surface condition, the propensity of a component to fail and cause an Anticipated Operating Transient (AOT) does not change. Consequently, the NMCA treatment does not increase the probability of an accident or anticipated operating transient previously evaluated in the SAR.

The primary concern regarding treatment of reactor surfaces is whether platinum (Pt) and rhodium (Rh) could affect the course of an event by their presence on the surface or in the reactor water. After treatment, Pt and Rh are bound to the surface and will be difficult to remove without also removing the underlying material. However, reactor crud that is resident on components such as the fuel will be treated and may loosen during a transient or accident. Consequently, there is a potential for noble metal treated particles to be released into the reactor water during a transient or accident. If the accident were to occur during the NMCA process, noble metal compounds or particles would already be present in the reactor water. The catalytic action during accidents and (AOTs) is negligible. The presence of noble metal compounds will not adversely

affect the accident consequences due to catalytic action. NMCA treatment will not interfere with safety actions of mechanical components. The impact of treating fuel cladding is negligible. Consequently, NMCA treatment does not increase the consequences of an accident previously evaluated in the SAR, and it does not increase the probability of a malfunction of a safety related structure, system or component previously evaluated in the SAR. It does not create the possibility of an accident which is different than any already evaluated in the SAR.

The NMCA treatment is a passive catalytic material embedded into the base materials, to improve water chemistry and reduce the potential of stress corrosion cracking. NMCA treatment does not create the possibility of an accident, different than previously evaluated in the SAR, and it does not create the possibility of a malfunction of a safety related structure, system or component different than any already evaluated in the SAR.

The Technical Specification basis does not describe any margin of safety for the vessel water chemistry. A change of the reactor water conductivity limit to a higher value was required to support the implementation of the NMCA process during shutdown. Technical Specification change, Amendment 218, was approved by the NRC on October 3, 1996, to raise the reactor water conductivity limit. The expected increase in conductivity was due to the effect of noble metal chemistry during the application period. During and after 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.

SpTP 198 Emergency Service Water (ESW) Make-up To Spent Fuel Pool (SFP)

Description and Basis for Change

The purpose of this Special Test was to demonstrate that the ESW to SFP make-up line flow rate would exceed the minimum make-up flow rate specified in the UFSAR. Generic Letter 89-13 initiated a review of the safety performance of the plant's safety related service water systems. As a result, ESW to the SFP was tested with unacceptable results. Hardware and procedure changes were made. This test demonstrated that the UFSAR requirement of 43.11 gpm for a full core off-load can be met.

Summary of Safety Evaluation

Testing the ESW make-up to the fuel pool resulted in changes to the ESW flow distribution, but did not affect anything that is a potential accident

initiator. ESW is a support cooling water system. Therefore, the probability of an accident was not increased. The minimum ESW flow required was supplied to the major loads required to be operable during the limiting case scenario. Flow to the High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system room coolers was interrupted as allowed by Technical Specifications. In the event that the room coolers were needed to support HPCI or RCIC, the test would have been aborted and the coolers placed in service. Therefore, the consequences of an accident were not increased, and the consequences of a malfunction were not increased. The margin of safety as defined in the basis for any Technical Specification was not reduced by this testing. The effect of this event on the Control Building chiller was expected to be less severe than a normal start of the ESW system. Throttling ESW loads to idle equipment or room coolers did not affect their reliability since total ESW flow remained above 300 gpm, which was procedurally controlled, and minimum component flowrates were maintained. Therefore, the probability of occurrence of a malfunction of equipment was not increased, and the possibility of a malfunction that was not previously considered was not created. Since only one loop of ESW was tested at any one time, the possibility of an accident of a different type was not created.

SE 95-03 Revision of FHA-800 To Supersede Appendix 'A' Requirements With Appendix R Requirements

Description and Basis for Change

Design Document Change (DDC) 3151 revised FHA-800, Branch Technical Position APCSB 9.5-1 Appendix A Commitment Cross-Reference Index (also known as FHA Appendix C), to reflect the fact that some portions of Appendix A have been superseded by DAEC commitments to 10CFR50 Appendix R.

The following Appendix A commitments have been superseded:

* 3-hour rated fire barriers surrounding the Control Room

* 3-hour rated fire barriers surrounding the Cable Spreading Room

* 3-hour rated fire barriers surrounding the Switchgear Rooms

* 3-hour rated fire barriers surrounding the Station Battery Rooms

* 3-hour rated fire barriers surrounding the Main Turbine Oil Storage Area

* 3-hour rated fire barriers surrounding the Emergency Diesel Generator Areas

* 3-hour rated fire barriers surrounding the Diesel Fuel Day Tank Rooms

* 3-hour rated fire barriers surrounding Safety Related Pump Areas

* 3-hour rated fire barriers surrounding the Heater Boiler Room

* 3-hour rated fire barriers surrounding Stairwells

* 3-hour rated fire barriers surrounding the Radwaste Building

* 3-hour rated fire barriers for building exterior walls

* Emergency lighting for safety-related areas including those areas where fires could result in the release of radioactive materials

Instead of these Appendix A commitments, the determination of areas requiring three hour rated fire barriers and emergency lighting is based on the safe shutdown analysis performed to meet Appendix R. Three hour rated fire barriers are maintained, where required, to meet the separation requirements of 10CFR50 Appendix R, Section III.G. Emergency lighting is maintained in areas required for operation of, or access to safe shutdown equipment as required by 10CFR50 Appendix R Section III.J. The Appendix R barrier and lighting requirements may include some of the same areas that were maintained for Appendix A.

Fire barrier requirements for a barrier also pertain to any penetration seals, fire dampers, firedoors, protective cable coatings and structural steel fire proofing that form part of the fire barrier. If a fire barrier, installed in response to our previous commitment to Appendix A, is no longer required for DAEC compliance to Appendix R then the doors, cable coatings and steel fire proofing for that barrier are also considered to be superseded.

Summary of Safety Evaluation

The accidents previously evaluated for the DAEC are described in the UFSAR and the NSOA. Neither fire barriers nor emergency lighting are initiators for any of these accidents. Relaxation of BTP 9.5-1 Appendix A commitments that are superseded by 10CFR50 Appendix R commitments does not alter any of the initial inputs, conditions or assumptions for the

probabilities of these accidents. Therefore, this DDC does not increase the probability of occurrence of an accident evaluated previously in the SAR.

Fire barriers and emergency lighting are relied upon to meet the safe shutdown requirements of Appendix R to prevent fuel failure and offsite releases in the event of a design basis fire. Because the Appendix R requirements are still met, the consequences of a design basis fire do not prevent the plant from achieving safe shutdown conditions in the event of a fire. As both Appendix A and Appendix R do not postulate a fire concurrent with any other accident, the changing of fire protection requirements does not prevent plant systems or components from performing safety related functions required to mitigate the offsite dose consequences of non-fire accidents. Therefore, this change does not increase the consequences of an accident previously evaluated in the SAR, and it does not increase the probability of occurrence of malfunction of equipment important to safety evaluated previously in the SAR.

Neither fire barriers nor emergency lighting are required to mitigate any of the malfunctions evaluated in the DAEC UFSAR or NSOA. This change does not prevent other plant systems or components from performing safety functions required to mitigate the offsite dose consequences of malfunctions. Therefore, this change does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR.

Fire barriers and emergency lighting are provided to protect against design basis fires which are accidents currently evaluated in the UFSAR. Fire barriers and emergency lighting are still maintained where required by Appendix R to protect the ability of the DAEC to achieve and maintain safe shutdown in the event of a fire. This change does not affect the maintenance of barriers for structural, seismic, missile, shielding, flooding, tornado or Environmental Quality (EQ) concerns. Therefore, this change does not create the possibility of an accident of a different type than any evaluated previously in the SAR, and it does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

Fire protection has been removed from the scope of the DAEC Technical Specifications, and this change does not affect the Technical Specifications for any other systems. Therefore, this DDC does not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 95-06 Emergency Service Water (ESW) Flow to Spent Fuel Pool (SFP)

Description and Basis for Change

The ESW to Spent Fuel Pool (SFP) make-up line was repaired such that the available make-up flow rate would exceed the minimum make-up flow rate assumed in the UFSAR. As a result of the installation of high density fuel racks into the SFP, the amount of flow required to make-up for evaporative losses under assumed worst case SFP heat load conditions increased from 38.8 gpm to 43.11 gpm. This issue was resolved when the makeup flow from ESW available to the SFP exceeded 43.11 gpm, however, it was proposed that the ESW make-up flow rates available prior to final resolution may have been less than 43.11 gpm. An evaluation was performed to determine if flow rates less than 43.11 gpm from ESW to the SFP constituted an unreviewed safety question. This evaluation only evaluated the acceptability of continuing to operate the DAEC without restriction up to the time when the core was again off-loaded to the SFP. Consequently, the safety evaluation only evaluated the worst case heat load that would have existed throughout Operating Cycle 14. No unreviewed safety question was identified. This safety evaluation was reported in NG-95-2971, "1995 Cyclic Report of Facility Changes, Tests and Experiments and Fire Plan Changes", from John F. Franz to Mr. William T. Russell, dated October 19, 1995. This evaluation has since been revised to determine the acceptability of the worst case heat load that existed in the SFP during Refuel Outage 13 (RFO-13), when the ESW system was not able to provide 43.11 gpm of makeup water. In addition, revised heat generation curves were used to estimate the minimum time to boil that would have actually been present during RFO-13.

The ESW system is only required to provide makeup water to the SFP when the SFP gates are installed and the SFP is isolated from the Reactor Cavity. The basis for the 43.11 gpm ESW makeup requirement is a postulated complete loss of Spent Fuel Pool Cooling (SFPC) approximately 7.6 days after a complete core off-load following the last plant operating cycle, which results in a completely filled SFP. The assumed heat load under these conditions is 20.1 MBtu/hr. This assumes the loss of SFPC occurs due to a failure of one of the fuel pool gates. The resulting heat load and, consequently, the ESW makeup requirement is the same regardless of what caused the loss of SFPC. The 43.11 gpm makeup requirement for ESW is derived from this heat load using the latent heat of evaporation and specific volume of water at 212° F. For this analysis, the actual heat load present in the SFP during RFO-13 was used to determine the ESW makeup flow requirement. For the actual heat loading present during RFO-13, 17.0 MBtu/hr after 7.6 days and 15.0 MBtu/hr after 12

days, the ESW make-up requirements were determined to be 36.6 gpm and 32.2 gpm, respectively.

During RFO-13, the RHR system was available to be aligned to provide SFP cooling during the time when the core was off-loaded to the SFP and the SFP gates were installed. After the SFP gates were removed, several systems were available to provide makeup to the combined Reactor Cavity/SFP, to makeup for any evaporative losses that could have been present had the SFPC system been lost.

The minimum time to boil that occurred during RFO-13 was more than an adequate amount of time to allow the operating crew to align the RHR system to the SFP, had a loss of the SFPC system occurred during RFO-13. Also, the RHR system (one loop) possessed adequate capacity to remove the heat load that was present. Thus the need for the ESW system to provide flow to the SFP to makeup for evaporative losses due to boiling was effectively precluded. Therefore, an adequate level of safety existed during RFO-13.

Summary of Safety Evaluation

The ESW to SFP piping and hose connection do not contribute to the probability of occurrence for any accidents or transients previously evaluated. The loss of SFP cooling event that the ESW makeup flow would be required to mitigate is not considered an accident or transient, and the ability of ESW to provide the UFSAR assumed makeup flow does not influence the relative probability of occurrence of the event in the first place. The function of providing SFP cooling is a safety action for planned operations and loss of SFP cooling is not considered an accident or transient that is evaluated in the SAR. The condition of the ESW makeup to SFP piping will not affect the probability of occurrence of a malfunction of the SFPC system. Since other means are available for mitigating a loss of SFPC such that makeup from ESW would not be reasonably expected to be required, the consequences of such an event are not increased. Since the ESW to SFP makeup line does not interface with any systems, structures or components that can act as initiators of any accidents or transients, the relative condition of the makeup line cannot create the possibility of an accident or transient of a different type than previously evaluated. The activity of either restoring the ESW system's ability to provide makeup, or allowing the ESW system's makeup ability to remain degraded until repaired cannot cause a malfunction of a different type than any previously evaluated. The margin of safety as defined in the bases of any Technical Specification is not reduced.

SE 95-09 Core Operating Limits Report (COLR) For Cycle 14

Description and Basis for Change

The original safety evaluation was reported in NG-95-2971, "1995 Cyclic Report of Facility Changes, Tests and Experiments and Fire Plan Changes", from John F. Franz to Mr. William T. Russell, dated October 19, 1995. This safety evaluation has since been revised to add additional information and justification. The COLR is prepared to support the addition of new fuel to the core and the relocation of the existing fuel remaining in the core for the next operating cycle. The COLR contains the thermal limits for the fuel, which are derived from the results of the analysis of the limiting operating transients and accident analyses in the UFSAR. The cycle-specific analysis of these limiting transients and accidents is performed using NRC-approved methods, as described in GE's Standard Application for Reactor Fuel (GESTAR: NEDE-24011-P-A), and the results are presented in the Cycle 14 Supplemental Reload Licensing Submittal (SRLS) for the DAEC. The fuel design used for Cycle 14 is the same fuel design (GE10) as that loaded in previous reloads at the DAEC (i.e., Cycles 11, 12 and 13) This fuel design is licensed by the NRC via GE's topical report, NEDE-31152P, "GE Fuel Bundle Designs".

Summary of Safety Evaluation

The changes in the Cycle 14 thermal limits and addition of fresh fuel did not increase the probability of a previously evaluated accident. The GE10 fuel design met all requirements for fuel and is a like-for-like replacement of the fuel previously loaded in the DAEC core. Compliance with the thermal limits for this core design ensures that the fuel design requirements are satisfied during reactor operation in all applicable Operating States in the NSOA. The fuel, proper, is an event initiator only for the Fuel Loading Error (either mislocated or rotated bundle). Because no changes to the fuel loading and verification procedures were made as part of this change, and the GE10 fuel type has the same verification attributes as the previous fuel designs, the probability of a Fuel Loading Error is not increased. Thermal limits were derived using NRC-accepted methods, which demonstrate, in the SRLS, that the consequences of the limiting events in UFSAR Chapter 15 are within the acceptance criteria for such events. Consequently the consequences of any previously-analyzed event are not increased, and the margin of safety as defined in the basis for any Technical Specification is not reduced. The probability of a failure of the fuel cladding, the equipment important to safety, when operated in accordance with the thermal limits is not increased from that previously evaluated. The ASME Vessel Overpressure analysis in the SRLS

demonstrates that the peak RPV pressure is well within the design allowable. Thus, the probability of a vessel overpressure and subsequent over stressing of the Reactor Coolant Pressure Boundary are not increased from that previously evaluated. Consequently, the probability of occurrence of a malfunction of equipment important to safety previously evaluated is not increased, the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR is not increased, and the possibility of a new or different type of malfunction is not created. The fuel design criteria for the new fuel have been shown to be satisfied. There is not the possibility of the fuel, by virtue of its design, creating an event that is different from any previously evaluated. Since the only events which are associated with loading of the fuel have been previously analyzed, there is no possibility of creating a new or different type of accident than any previously evaluated in the UFSAR.

SE 96-05 Fuel Shuffle After 120 Hours of In-Vessel Decay

Description and Basis for Change

License Amendment 195 increased the spent fuel storage capacity of the Spent Fuel Pool (SFP). Thermal hydraulic analyses for the SFP and its cooling system were performed by Holtec International. The scenarios shown in the Holtec Analysis are discussed in DAEC's UFSAR. The Holtec analysis for a fuel shuffle assumes that DAEC moves fuel from the Reactor Pressure Vessel (RPV) to the SFP at 150 hours after reactor shutdown at a rate of 144 assemblies per 24 hours. Only one loop of Fuel Pool Cooling and Cleanup (FPCCU) is assumed to be in service. This analysis was performed to provide a comparison of the FPCCU to the provisions in NUREG-0800, SRP 9.1.3.

Typical practice at DAEC has been to perform a full core offload each refuel outage. The results of the Holtec Analysis documenting a full core offload shows that DAEC can move fuel from the RPV to the SFP at 120 hours after reactor shutdown at a rate of 144 assemblies per 24 hours. Both loops of FPCCU are assumed to be in service. This analysis also bounds the conditions for a partial offload or shuffle. This safety evaluation was performed because the Holtec Analysis did not provide a specific calculation of the exact scenario proposed for the Refuel Outage 14 (RFO-14) fuel shuffle. The only portion of the Licensing Basis needing clarification was the amount of time required for in-vessel decay of the nuclear fuel when DAEC shuffled fuel. The time changed from 150 hours to 120 hours. All other bounding conditions as specified in the Holtec Analysis for a full core offload were applied to the scenario for the RFO-14 fuel shuffle. This safety evaluation shows that it is acceptable to

perform a fuel shuffle at 120 hours after reactor shutdown at a rate no greater than 144 assemblies in any 24 hour period.

Summary of Safety Evaluation

This change is within the bounds of the Licensing Basis as identified in the Holtec Analysis. This change does not increase the probability of an accident. The reduction in the time to offload for a fuel shuffle does not increase the likelihood of equipment failure, dropped fuel assembly or any of the other credible modes of failure. Therefore the amount of radioactive material that could be released has not increased. This change does not increase the consequences of an accident. All of the applicable equipment used was rated for continuous duty and any preventive maintenance was completed prior to the commencement of fuel movement. This change does not increase the probability of equipment failure. The amount of decay heat is removed from the SFP at a greater rate. Removal of the decay heat at a more rapid rate would actually increase the time to boil. The same equipment was used in the fuel shuffle regardless of when the fuel movement was started, and the same amount of nuclear materials were present. Therefore, this activity did not increase the consequences of an equipment malfunction, and it did not create the possibility of a different type of accident. This activity did not create the possibility of a different type of malfunction than previously evaluated. This change did not reduce the margin of safety as defined for any Technical Specification. The fuel pool level was maintained at 36 feet (or greater) as required by Technical Specifications. None of the operating specifications were decreased due to this activity.

SE 96-06 Use of the Term "Safe Shutdown" Instead of "Safety Related" For DAEC Fire Protection Issues

Description and Basis for Change

The purpose of this evaluation was to revise the scope of the DAEC Fire Protection Program Administrative Control Program procedures and checklists so that these documents review the impact of the scope of the procedure or checklist on "Safe Shutdown" equipment and systems rather than "Safety Related" equipment and systems. In addition, various documents pertaining to the DAEC Fire Protection Program use the term "Safety Related" or some variation such as "Structures, Systems, and Components Important to Safety". These documents include 10CFR50 Appendix R, BTP 9.5.1 Appendix A, FRAC&QA, UFSAR, SER, ACPS, DAEC Fire Plan, DAEC FHA, and DAEC Fire Barrier Evaluations. This evaluation also clarifies the usage of these various and diverse terms as they pertain to the DAEC Fire Protection Program and evaluates the

interpretation of these various and diverse terms as "Safe Shutdown" instead of the terms mentioned above. This generic change reflects changes made in the DAEC fire protection philosophy as a result of the comprehensive review of the DAEC Fire Protection Program performed as part of the DAEC Appendix R re-baseline effort.

Summary of Safety Evaluation

The change in the DAEC Fire Protection Program Administrative Control Program procedures and checklists and the proposed change in interpretation of other documents did not increase the probability of occurrence of an accident evaluated previously in the SAR. This change required that the DAEC Fire Protection Program review the impact of the scope of the procedure or checklist on "Safe Shutdown" equipment and systems rather than "Safety Related" equipment and systems. The appendix R Re-baseline Program has analyzed which equipment and systems at the DAEC are required to safely shut down the reactor in the event of a fire in any single plant fire area. This listing is controlled by DAEC Fire Protection Engineering, and includes both safety related and non-safety related equipment and systems and fulfills the intent of the wording such as, "Safety Related, Safe Plant Shutdown and Structures, Systems and Components Important to Safety", used in the various documents. A review of only safety related equipment and systems may fail to evaluate the impact of proposed work on non-safety related safe shutdown equipment which is required and analyzed to safely shutdown the DAEC in the event of a fire. As the intent of the DAEC Fire Protection Program is to reduce the likelihood of a fire, promptly detect and extinguish fires if they occur, maintain the capability to safely shutdown the plant if fires occur, and prevent the release of a significant amount of radioactive material if fires occur, all reviews of potential changes to the DAEC, including control of ignition sources, should be reviewed against the listing of equipment and systems which have been analyzed to safely shutdown the DAEC in the event of a fire.

The change did not impact any accident analysis, equipment or system installed at the DAEC including their design bases and assumptions. Therefore, the change did not affect the consequences of any accident, and it did not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. It did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The interpretation of the various and diverse terms such as, "Safety Related, Safe Plant Shutdown and Structures, Systems and Components Important to Safety", as "Safe Shutdown" is conservative and is consistent with the methodology and analyses developed during the Appendix R Re-baseline Program. It does

not result in an accident of a different type than evaluated previously, and it does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. Fire Protection is not discussed in the DAEC Technical Specifications. Therefore, this change in the DAEC Fire Protection Program Administrative Control Procedures and checklists to review the impact of the scope of the procedure or checklist on "Safe Shutdown" equipment and systems rather than "Safety Related" equipment and systems does not reduce the margin of safety as defined in the basis of any Technical Specification.

SE 96-08 Revision to Operating Instruction (OI) 149, Residual Heat Removal (RHR) System

Description and Basis for Change

GE SIL 69, "Operation of RHR Shutdown Cooling & Recirculation In Parallel", discusses the potential for reverse flow to occur in some piping systems of the plant. Operating two pumps in parallel could cause a hydraulic "Beat", flow stagnation or flow reversal in the pump having the lower discharge pressure. The SIL gives guidance on recommended and discouraged pump operation configurations.

Upon re-review of the SIL and upon further evaluation it was determined that certain pump operation configurations are acceptable for DAEC. An Operating Instruction (OI) was revised to address the operation of the Recirculation system (RECIRC) and the Residual Heat Removal system in the Shutdown Cooling mode (RHRSDC) in parallel for the following situations:

* With RECIRC already in operation, placing RHRSDC in service.

* With RHRSDC already in operation, placing RECIRC in service.

* Securing from the above-mentioned evolutions.

Operation with RECIRC and RHRSDC in parallel enhances shutdown and startup activities by eliminating periods of "no forced circulation" in the vessel in shutdown or startup modes. The following issues were considered for this change:

* Core Flow - ensuring adequate flow through the core to ensure proper mixing of reactor coolant to avoid thermal stratification and unanticipated core boiling.

* RHR and Recirculation Pump Protection - preventing reverse flow through a recirculation loop which would have the potential to deadhead the RHR and/or RECIRC pumps, and ensuring Net Pump Suction Head (NPSH) requirements are met in order to prevent pump cavitation.

* Thermal Transients - prevent exceeding heatup and/or cool down limits.

Summary of Safety Evaluation

The probability of occurrence of an accident previously evaluated in the SAR is not increased because the operation of RHRSDC and RECIRC in parallel is not an initiator of an accident which is evaluated previously in the SAR and the operation of RHRSDC and RECIRC in parallel (both during and after parallel operation) will not damage the RHR and/or Recirculation pumps. Nor will operation of RHRSDC and RECIRC in parallel cause any adverse effects on the RECIRC piping, whose failure is an accident evaluated in the SAR. The consequences of an accident previously evaluated in the SAR is not increased because the plant condition during which RHRSDC and RECIRC are operated in parallel is not the most limiting condition which is evaluated in the SAR. The ability of RHRSDC and/or RECIRC to perform any of their required functions is not adversely affected and the ability to maintain piping stresses within allowable limits is not affected. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased. The RHR system contains equipment important to safety and RECIRC system piping can be considered as equipment important to safety. RHRSDC and RECIRC will be fully capable of mitigating the consequences of an accident by being able to perform any of their required functions. The consequences of any malfunctions of RHRSDC and RECIRC when operated in parallel, are within the bounds of any evaluations of malfunctions of RHRSDC and RECIRC currently contained in the SAR. Therefore, this change does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The possibility for an accident of a different type than any evaluated previously in the SAR is not created because the maximum potential adverse effect is increased stresses on the RECIRC piping which could potentially cause a failure of the piping and this accident is previously evaluated in the SAR. This change does 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 operation of RHRSDC and RECIRC in parallel will not damage the RHR and /or RECIRC pumps and the ability to maintain piping stresses within

allowable specifications is not adversely affected, the margin of safety as defined in the basis for any Technical Specification is not reduced.

SE 96-09 Noble Metal Chemical Addition (NMCA) Pretreated Rods

Description and Basis for Change

This evaluation was performed to assure the safety of using a fuel bundle with six fuel rods treated with noble metals, platinum and rhodium, in the reactor. NMCA was performed during cool down at the end of Cycle 14. During this process, all of the fuel present in the reactor at that time was treated by the NMCA process. The pretreatment of six fuel rods in the test GE10 bundle was needed to allow monitoring the performance of fuel rods which have not been exposed to the reactor environment prior to NMCA treatment. This was necessary to demonstrate that NMCA treatment has no adverse effect on fresh Zircaloy cladding. This demonstration will serve as a basis for qualifying possible future GE options, to applying the NMCA process during a startup after refueling with fresh fuel surfaces. To qualify the NMCA process, tests were performed, both in the GENE laboratory and in the Halden reactor in Norway. These tests have shown that no detrimental effects to the fuel cladding from NMCA treatment are expected. The effect of the presence of noble metal on the surface of the fuel clad with respect to heat transfer, metal-water reaction, transients and accidents have been evaluated and have been found to be negligible.

Summary of Safety Evaluation

The NMCA pretreatment of fuel rods is passive and does not affect the functional performance of the fuel or cladding. Therefore, there is no increase in the probability of occurrence of an accident previously evaluated in the SAR. If the six pretreated fuel pins were to fail, the radiological consequences would be bounded by the existing Regulatory Guide 1.3 analysis which requires an assumption of extensive fuel failures. Therefore, this change has no impact on the consequences of an accident previously evaluated in the SAR and the consequences of a malfunction of equipment important to safety are not increased. The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased because the noble metal is passive and does not introduce any new equipment that could fail or experience a malfunction. The possibility of an accident of a different type than previously evaluated in the SAR was not created as a result of the installation of the bundle containing the NMCA pretreated fuel rods (PFR) because this change does not affect the functional operation of the bundle. The possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the SAR is not created as a

result of the installation of the bundle. The installation and use of the NMCA PFR bundle does not reduce the margin of safety as defined in the basis for any Technical Specification because the change does not result in violating any of the specified acceptable fuel design limits contained in GESTAR II.

SE 96-11 UFSAR Change: Turbine Building Flooding

Description and Basis for Change

DAEC's UFSAR discusses flooding of the Turbine Building (TB) basement that could be caused by a ruptured Circulating Water system expansion joint. The UFSAR was reworded to acknowledge the existence of safety-related equipment in the floodable space, while also stating that safe plant shutdown is not jeopardized in the event of TB flooding. The following Quality Level 1 components located within the floodable space of the TB basement were identified:

* Main Steam Line (MSL) High Temperature Switches
* MSL Low Pressure Switches
* High Pressure (HP) Turbine First Stage High Pressure Switches
* Turbine Building (TB) 480VAC Motor Control Center (MCC)
* Buried Cabling Supplying Division II Safety-Related Equipment in the Pumphouse and Intake Structure

Summary of Safety Evaluation

The failure of MSL High Temperature Switches and MSL Low Pressure Switches when flooded would cause an MSIV isolation, which is an accident evaluated in the SAR. The existence of this instrumentation in the floodable depth of the TB basement does not shift the MSIV Isolation event to a higher frequency category, since the flooding event would be extremely infrequent.

The failure of HP Turbine First Stage High Pressure Switches when flooded would cause a turbine trip to occur causing a reactor scram, regardless of reactor power. This scenario is bounded by the Turbine Trip events evaluated in the SAR. These instruments fail in a conservative direction in the event of TB basement flooding and would play no part in the occurrence of any of the accidents evaluated in the SAR.

Failure of the TB 480VAC MCC and loss of its loads would not contribute to the increased occurrence of any accident evaluated in the SAR, and would have no detrimental impact on other safety related equipment.

The cabling running below the TB basement floor and through watertight conduits would be unaffected by the TB basement flooding event.

Therefore, this change does not increase the probability of occurrence of an accident evaluated previously in the SAR, and it does not create the possibility of an accident of a different type than any evaluated in the SAR. It does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR, and it does not increase the consequences of an accident evaluated previously in the SAR.

All safety related equipment located in the TB basement will still perform its safety related function in the event of TB basement flooding. Therefore, the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is not increased. The consequence of malfunction of the safety related equipment located in the TB basement is the same regardless of its location. No physical changes to the equipment were made in conjunction with this UFSAR change, therefore the consequences of a malfunction of equipment important to safety evaluated previously in the SAR is not increased, and the margin of safety as defined in the basis for any Technical Specification is not reduced.

SE 96-14 Core Operating Limits Report (COLR) For Cycle 15

Description and Basis for Change

The COLR is prepared to support the addition of new fuel to the core and the relocation of the existing fuel remaining in the core for the next operating cycle. The COLR contains the thermal limits for the fuel, which are derived from the results of the analysis of the limiting operating transients and accident analyses in the UFSAR. The cycle-specific analysis of these limiting transients and accidents is performed using NRC-approved methods, as described in GE's Standard Application for Reactor Fuel (GESTAR: NEDE-24011-P-A), and the results are presented in the Cycle 15 Supplemental Reload Licensing Submittal (SRLS) for the DAEC. The fuel design used for Cycle 15 was the same fuel design (GE10) as that loaded in previous reloads at the DAEC (i.e., Cycles 11, 12, 13 and 14). This fuel design is licensed by the NRC via GE's topical report, NEDE-31152P, "GE Fuel Bundle Designs".

Summary of Safety Evaluation

The changes in the Cycle 15 thermal limits and addition of fresh fuel do not increase the probability of a previously evaluated accident. The GE10 fuel design meets all requirements for fuel and is a direct replacement of the fuel previously used in the DAEC core. Compliance with the thermal limits for this core design ensure that the fuel design requirements are satisfied during reactor operation in all applicable Operating States in the NSOA. The fuel, proper, is an event initiator only for the Fuel Loading Error (either mislocated or rotated bundle). Because no changes to the fuel loading and verification procedures were made as part of this change, and the GE10 fuel type has the same verification attributes as the existing fuel designs, the probability of a Fuel Loading Error is not increased. Thermal limits were derived using NRC-accepted methods, which demonstrate in the SRLS, that the consequences of the limiting events in UFSAR are within the acceptance criteria for such events. Consequently, the consequences of any previously-analyzed event are not increased, and the margin of safety as defined in the basis for any Technical Specification is not reduced.

Since the GE10 fuel type meets 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 thermal limits provided in the COLR, is not increased from the previously evaluated. Also, the ASME Vessel Overpressure analysis in the SRLS demonstrates that the peak RPV pressure is well within the design allowable. Thus, the probability of a vessel overpressure and subsequent over stressing of the Reactor Coolant Pressure Boundary are not increased from that previously evaluated. Consequently, the probability of occurrence of a malfunction of equipment important to safety previously evaluated is not increased. The consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR is not increased, and the possibility of a new or different type of malfunction is not created. The fuel design criteria for the new fuel have been shown to be satisfied. There is not the possibility of the fuel, by virtue of its design, creating an event that is different from any previously evaluated. Since the only events which are associated with loading of the fuel have been previously analyzed, there is no possibility of creating a new or different type of accident than any previously evaluated in the UFSAR.

SE 96-15 UFSAR Change: Use of BWRVIP-18 Examination Techniques in Lieu of Examination Techniques in IE Bulletin 80-13

Description and Basis for Change

IE Bulletin 80-13 was issued to request licensees to perform (each refueling outage) a visual inspection of the Core Spray spargers and the segment of piping between the inlet nozzle and the vessel shroud. Remote underwater TV examinations are acceptable if adequate resolution (i.e., viewing in situ of 0.001 inch diameter fine wires) can be demonstrated.

The purpose of this UFSAR Change is to allow the use of BWRVIP-18 examination techniques instead of IE Bulletin 80-13 techniques. Since the inspection history of the vessel internals has been very good at the DAEC, the BWRVIP-18 examination requirements in Figure 3-2 visual (VT) flow path for geometry tolerant plants will be followed. A Core Spray (CS) VT-1 (the same examination required by Bulletin 80-13) will be performed on the sparger welds with a VT-3 of the nozzles/nozzle welds and the supports. The nozzle with the missing tack weld will continue to have the CSVT-1 examination performed to verify integrity of the remaining tack weld. Figure 3-1 VT flow path will be followed for the Core Spray piping. An enhanced VT-1 will be performed on all Core Spray piping welds.

Summary of Safety Evaluation

No existing structures, systems, or components are being physically changed. The activity evaluated only changes the type of inspection used to monitor the core spray piping for evidence of crack indications. The alternative inspection will ensure the integrity of the core spray piping. The probability that the core spray piping will crack will not be increased.

The change in inspection method does not result in an increase in the consequences of an accident evaluated previously in the SAR. The examination techniques to be used (in accordance with BWRVIP-18) for the core spray piping are more sensitive than those required by the IE Bulletin 80-13. The resolution is more sensitive and cleaning is required. While the bulletin requires a 100% examination of the Core Spray piping whereas BWRVIP-18 does not, the area which will not be examined is not the area where intergranular stress corrosion cracking (IGSCC) would start. The welds and the associated heat affected zones are the areas of concern when examining for indications of IGSCC, and these areas will be examined more closely by the BWRVIP-18 technique than they have been by the bulletin technique. Therefore, this change does not increase the

probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR.

This change was to the inspection method for monitoring potential cracking in the Core Spray piping. No components required any physical change. The function, operation and reliability of the Core Spray system is not compromised. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR are not increased. This change does not create the possibility of an accident of a different type than previously evaluated in the SAR, and it does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR.

Bulletin 80-13 is not addressed in the Technical Specifications. Since using a different inspection technique will not degrade the condition of the Core Spray system, no margin of safety as defined in the basis for any Technical Specification is reduced.

SE 96-17 Implementation of the Appendix R Re-baseline Program

Description and Basis for Change

This change replaces DAEC's existing Appendix R safe shutdown methodology and its associated analyses with the revised safe shutdown methodology and analyses developed by the Appendix R Re-baseline Project. This revised methodology and subsequent analyses were developed as a result of ongoing interpretation and clarification of Appendix R by the NRC and as a result of difficulties in documentation and maintenance of the existing program.

The Re-baseline Program simplifies the DAEC fire analysis by combining formerly separate Fire Zones together into Fire Areas. This reduces the number of analyses and reduces the number of fire barriers requiring inspection. An additional goal of the Re-baseline Program was to reduce the requirement for raceway fire barrier material. The revised Re-baseline Program is based on the original Appendix R Program as described in the FHA and includes the following major enhancements:

* PC (Process Computer) based relational database instead of text based analysis.

* All facets of the analysis are fully documented and controlled by Project Instructions (PI's) developed as part of the Re-baseline Project. The data contained in the PI's has been transferred to the FHA.

* The improved documentation associated with the PC based database allows for a more complete and thorough review of proposed plant changes.

* The PC based database is directly derived from the DAEC CHAMPS cable and raceway database. The raceway/fire zone/fire area relationship was reviewed and independently verified as part of the re-baseline database development.

* The PC based database better identifies circuit interactions (auxiliary contacts) than the existing text based analysis.

* The PC based database better identifies power supply/cable interfaces than the existing text based analysis.

The re-baseline analysis methodology was developed by IES in cooperation with Bechtel and Vectra. The re-baseline methodology is similar to the original methodology with the following significant exceptions:

* Allows field operator actions to mitigate the consequences of the fire instead of requiring that all of these actions take place within the control room. These field operator actions include re-entry into the fire area if necessary to mitigate the consequences of the fire. In accordance with the re-baseline procedures, all proposed operator actions are analyzed to determine that these actions are prudent and that sufficient time is available to successfully perform these actions.

* Assumes that a Loss Of Offsite Power (LOOP), for non-Alternate Shutdown Capability (ASC) fire areas can only occur concurrent with a fire event when the fire induced damage results in spurious operations or other malfunctions which could result in a LOOP.

In accordance with guidance provided in Appendix R, areas of the plant which rely on alternative or dedicated shutdown systems are still required to assume a LOOP concurrent with the onset of the fire event. The only such system at the DAEC is the (ASC) system which is designed for safe shutdown of the DAEC in the event that a control room evacuation is required as the result of a fire in specific plant areas as determined by the Re-baseline Project Fire Area analyses.

The re-baseline analysis may derive different Fire Area compliance strategies than those contained in the original Appendix R evaluation for a similar analysis area. As with any analysis or calculation, changes in the

assumptions and methodology may result in different solutions to the same problem. Based on the assumptions and methodologies used, both the original and re-baseline solutions, while different, are correct. The differences in compliance strategies between the original and re-baseline analyses for a similar analysis area are not discussed in the re-baseline analysis due to the Fire Zone consolidation and Thermo-lag reduction efforts which were part of the re-baseline project. Each re-baseline analysis is contained in an IES calculation which is independently prepared and approved in accordance with DAEC procedures. The calculation contains all references and assumptions required to allow independent review and understanding of the analysis. The acceptability of the compliance strategy for each area is justified as part of the analysis for that Fire Area.

Various modifications were required by and support the DAEC Re-baseline Analysis. The impact of these modifications on the safe operation of DAEC has been evaluated and are discussed as part of their installing modification.

Summary of Safety Evaluation

The implementation of the Appendix R Re-baseline Program will not increase the probability of occurrence of an accident evaluated previously in the SAR. No new plant systems and or components are added by the re-baseline analysis. The Appendix R Re-baseline Program has re-analyzed which equipment and systems at the DAEC are required to safely shut down the reactor in the event of a fire in any single plant fire area. The revisions to the DAEC safe shutdown analyses and assumptions made as part of the re-baseline analysis are a result of clarifications to Appendix R made by Generic Letter 86-10 and other NRC and industry documents. The justification and bases for all assumptions used in the re-baseline analysis have been documented and evaluated in the PI's developed as part of the re-baseline analyses. These PI's were developed specifically for the re-baseline project. The improved documentation provided by the re-baseline analysis improves the accuracy of the analysis and reduces the possibility for errors or omissions and allows for easier and more complete review of plant changes.

The change which allows field operator manual actions to mitigate the consequences of a fire instead of requiring that all of these actions take place within the control room adds significant flexibility to the analysis. These field operator actions include re-entry into the fire area if necessary to mitigate the consequences of the fire. In accordance with guidance provided in the Re-baseline PI, all operator actions required by specific fire area analysis are reviewed and are considered feasible only if specific

conditions are met. This review assures that adequate time is available and provides assurance that the proposed manual action will not result in an unrecoverable condition or require that the DAEC go through an unacceptable evolution as a result of the proposed manual action.

Design basis fires are not assumed to occur concurrently with non-fire related failures in safety systems, plant accidents or the most severe natural phenomena. Depending on the location and impact of the fire, off-site power may or may not be available. Offsite power is assumed to be available if systems and equipment required to establish and control the flow of offsite power to the DAEC safe shutdown systems and components are not adversely impacted by the fire event. In accordance with guidance provided in Appendix R, areas of the plant which rely on alternative or dedicated shutdown are still required to assume a LOOP concurrent with the onset of the fire event.

The implementation of the Appendix R Re-baseline Program does not impact any accident analysis, equipment or system installed at the DAEC including their design bases and assumptions. Therefore, the change does not affect the consequences of any accident evaluated previously in the SAR, and it does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. It does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and it does not create the possibility of an accident of a different type than any evaluated previously in the SAR. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR is not created by this change. The Appendix R Re-baseline Analyses specifically identifies systems and components important to safe shutdown which may misoperate and evaluates the impact of these fire induced misoperations on the ability to safely shutdown the DAEC in the event of a fire. This change ensures that appropriate equipment is reviewed by the Fire Protection Program and does not affect the consequences of any fire induced or other maloperation. This change reflects the results of the Appendix R Re-baseline Project which evaluated equipment and systems required for safe shutdown in the event of a fire and will not result in a malfunction of a different type than evaluated previously in the SAR.

Fire protection is not discussed in the DAEC Technical Specifications. Therefore, this change does not reduce the margin of safety as defined in the basis of any Technical Specification.

SE 97-01 UFSAR Change: Accident Range Pressure Indication

Description and Basis for Change

This evaluation was performed to resolve a technical issue discovered during review of the UFSAR. Two reactor pressure indicators mounted locally on instrument racks were found not to have maximum pressure indicating pointers as described in UFSAR. Pressure transmitters, indicators and recorders for accident range pressure (0-1500 psig) were installed which provide two fully redundant display and recording channels in the control room, in accordance with post-TMI requirements of Reg Guide 1.97 (type B, category 1), Reg. Guide 1.29--Category 1 and Class 1E electrical. This evaluation demonstrates that the safety related function of the maximum pressure indicating pointers on the pressure gauges on local instrument racks is fulfilled by indication and recording to the control room.

Summary of Safety Evaluation

The functions changed are post-accident monitoring indications. Additional indication is provided in the control room to allow the operators to safely control the plant during normal operations and during transient events. There is no change to the probability of occurrence of an accident as the result of this change. This change will not increase the consequences of any accident previously evaluated. Indication is now located in the control room, aiding the operators to better mitigate the consequences of a transient or accident. This change does not have the potential of increasing the probability of any malfunction which has previously been evaluated. This change will aid operators to limit the consequences of a malfunction of equipment important to safety. It does not have the potential to increase the consequences of a malfunction of equipment important to safety. This is an indication/recording function, which does not have the potential to create the possibility of an accident of a different type than any which has previously been evaluated. No automatic actions are associated with this instrumentation and no actions are taken during operation based on the indications from these instruments. There are two fully redundant indicating/recording channels, which provide a highly reliable indication and record of reactor pressure in the accident range to the operator in the control room. No possibility is created of a malfunction of equipment important to safety of a different type from what has been previously evaluated. This equipment monitors and records reactor pressure vessel pressure on a scale which exceeds the Technical Specification safety limit of 1335 psig. Since there are no automatic actions or operator actions which are driven by this

instrumentation, it does not have the potential to decrease the margin of safety.

SE 97-02 UFSAR Change: Reactor Protection System (RPS) Low Level Temperature Compensation

Description and Basis for Change

This evaluation was performed to resolve the technical issue of a discrepancy between the UFSAR and the as-built configuration of the DAEC. In the discussion of the low reactor water level scram trip setting, the UFSAR stated that temperature-compensating columns are used in the drywell to increase the accuracy of level measurement under normal and post-accident conditions. A drawing in the UFSAR also referred to the scram trip as coming from the heated reference leg instruments. The DAEC has never used temperature-compensating columns for this function. A cold reference leg arrangement is used.

Summary of Safety Evaluation

This change did not increase the probability of occurrence of any accident previously evaluated. Design Documents show that all accident scenarios were considered, with the installed configuration of cold reference leg instrument piping, when determining adequate allowable values and Nominal Trip Setpoints for the required functions. This configuration satisfies the same criteria of accuracy as the system described previously in the UFSAR and exceeds it in diversity and independence by using a separate, different type of instrument from the other low level functions, including the ECCS initiations, the PCIS Groups 1 and 5 isolations and the ATWS-RPT/ARI signal.

This change did not increase the consequences of an accident evaluated previously in the SAR, and it will not increase the consequences of a malfunction of any equipment which was previously evaluated. The installed configuration will perform the safety function of scramming the reactor as well as, or better than, the system previously described in the UFSAR, therefore, the consequences of any accident will not be increased. The ECCS initiation functions and PCIS Groups 1 and 5 functions for low reactor levels are carried out by equipment which is identical to that which was assumed in the UFSAR, so no increase in the consequences of an accident is possible. The high reactor level trip of the High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system is carried out by the cold-leg instruments. Since this is a high-level trip function, the effect of elevated drywell temperatures is non-conservative, but a setpoint calculation shows that full consideration was

given to this accident bias term when the required setpoint was determined.

This change will not increase the probability of occurrence of a malfunction of equipment which was previously evaluated. The installed equipment is highly reliable and fully qualified for nuclear safety related class 1E electrical service. It is fully redundant and installed in accordance with the requirements for such equipment. There is no indication that it is more or less reliable than the equipment assumed in the UFSAR discussion.

The installed system is superior, in that it reduces the consequences of any postulated equipment failure. In the heated reference column system previously described in the UFSAR, an initiating event of a broken reference leg, plus a failure in a Yarway level switch on the opposite side could prevent a scram, PCIS isolations 1-5, ATWS initiation and HPCI or RCIC initiation. It could prevent HPCI and/or RCIC from tripping on high level. The installed design is superior in this respect, as it addresses this, and therefore the installed configuration decreases the probability and severity of an accident due to an equipment malfunction.

This change will not create any possibility of an accident of a different type than previously evaluated, and it will not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated The installed configuration is functionally identical, in input and output, to the one previously described in the UFSAR and therefore does not add any different type of accident from what was analyzed previously.

The change does not reduce the margin of safety as defined in the basis documents. The setpoint methodology and the Design Basis Document (DBD) are specifically written to ensure that all safety margins are maintained at the DAEC by choosing setpoints which consider equipment accuracy and limitation, normal and accident environmental conditions, and the associated Analytical Limits and Safety Limits. Conformance to these documents in evaluating this setpoint is sufficient to maintain the margin of safety as defined in the basis document.

SE 97-03 UFSAR Change: Fire Plan Changes

Description and Basis for Change

The purpose of this evaluation is to evaluate the proposed changes to the DAEC UFSAR resulting from the implementation of the Appendix R Re-baseline Program and a detailed review of the current UFSAR by the

DAEC Fire Protection Engineering Group (FPEG). This evaluation addresses changes to the UFSAR which include the following:

* Removal of redundant material that can be found in other controlled documents.

* Clarifications of existing text.

* Corrections to existing text.

* Replacement of existing text as a result of the Appendix R Re-baseline effort.

The draft Fire Protection DBD, Fire Plan and NRC Fire Protection SER's, including Supplement 1 and the Appendix R SER, were used as a guide in developing the UFSAR change. When other sources which provide greater detail than that provided by the UFSAR provide ready access to details currently contained in the UFSAR the extra detail was removed from the UFSAR and a pointer was provided to that document.

Significant items addressed by this safety evaluation include:

* Removal of the requirement to coat cable routed in tray in the DAEC control room back panel area with fire retardant mastic.

* Use of in-situ testing as opposed to bench testing of smoke detectors.

* Use of non-sealed beam lights for fixed and portable emergency lights.

* Use of means other than Thermo-Lag material to provide separation for redundant cables routed in a single fire area.

* Revision of an UFSAR Table to match information that is also contained in the DAEC Fire Plan.

* Use of "substantial" as opposed to 3-hour rated walls surrounding the Diesel Fire Pump and Diesel Fire Pump Day Tank Rooms.

* Use of seals or locks in lieu of seals and locks to administratively control the positions of valves in the DAEC fire suppression systems.

* Removal of flow restrictors and use of a maximum of 100 feet of hose for local hose stations.

* Use of nozzles which can project a straight stream instead of fog/spray only nozzles.

* Clarification of the number and location of control room low flow booster stations.

* Deletion of the requirement for cable that does not give off toxic smoke and/or fumes when burned.

* Evaluating the acceptability for supplying automatic suppression systems and hose station(s) from the same building feed.

* Re-assigning the Control Building chiller area as a non-Alternate Shutdown Capability (ASC) Fire Area.

* Clarifying the criteria for the sizing of the diesel and electric fire pumps.

* Clarifying the criteria used in the design of fixed suppression systems at the DAEC.

Summary of Safety Evaluation

The UFSAR changes reflect the current status of the DAEC fire protection features and programs. The UFSAR changes evaluated by this safety evaluation cannot, by themselves, increase the probability of an accident evaluated previously in the SAR. Fire is not listed as an abnormal operating transient, design basis accident or special event. A fire is not listed as an entry condition, consequence or as an assumption for any abnormal operating transient, design basis accident or special event evaluated in the SAR. The level of fire protection is in response to commitments listed in the UFSAR, the Fire Protection DBD and the DAEC Fire Plan. The Fire Hazards Analysis and Appendix R analyses required by these commitments ensure that the DAEC can be safely shut down in the event of a design basis fire in any single plant fire area. This analyses ensures that the plant cannot enter or be placed in an unanalyzed or unrecoverable position as a result of or during recovery from a postulated fire event.

This change does not impact any accident analysis, equipment or system installed at the DAEC including their design bases and assumptions.

Therefore, this change cannot increase the probability of occurrence of an accident evaluated previously in the SAR, nor 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 are not increased, and the possibility of an accident of a different type than any evaluated previously in the SAR is not created. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR is not created. This change does not affect the consequences of any accident. Fire Protection is not currently discussed in the DAEC Technical Specifications. This change does not reduce the margin of safety as defined in the basis of any Technical Specification.

SE 97-09 UFSAR Change: Source Range Monitor (SRM) Response During Plant Startup

Description and Basis for Change

The use of the banked position withdrawal sequence (BPWS) necessitates that a dispersed rod withdrawal pattern be used to minimize the effects of a possible control rod drop accident. Although one control rod adjacent to each SRM will be withdrawn by the time that one-quarter of the rods in the core have been fully withdrawn, no requirement or assumption that, prior to achieving criticality, a rod adjacent to each SRM be withdrawn, exists in the design basis of the Source Range Monitor system for the DAEC. Therefore, a change was made to the UFSAR to ensure that it provides a clear description of the SRM system. In addition, the UFSAR was revised to reflect installed neutron sources may be used during the remainder of plant life to obtain the required neutron count rate, and the location of SRM neutron sources "when used" are shown.

Summary of Safety Evaluation

This change has no impact on any accident evaluated previously in the UFSAR, as no actual change in either SRM operation or control rod withdrawal sequence is being made by this change. Specifically, the probability of the "overheating of the fuel barrier" scenario is not increased by this change which improves the description of the system and is consistent with all design and licensing basis of the DAEC. The SRM's are not assumed to mitigate any event in the UFSAR, therefore, this change will not increase the consequences of any such analyzed event.

The SRM subsystem is not safety related. This UFSAR change will not impact the ability of any safety related equipment to perform its safety function. The system, structure, or components of the neutron monitoring

system are not changed in any way. The method of control rod withdrawal to be reflected in the UFSAR (BPWS) is a DAEC commitment which reduces the effects of a control rod drop accident.

Assuming normal operation or any previously-analyzed failure of the SRM system, the fission product barrier (fuel cladding) is not challenged any more severely because of this change, as the SRM's do not mitigate any analyzed event and no change in control rod withdrawal sequence was made. This method is not in conflict with any design basis document or Technical Specification. The SRM system will continue to meet its design basis function, no new failure mode is created and no change in control rod withdrawal sequence was made. Thus, no new accident initiator is introduced by this change.

This change cannot lead to a failure mode of a different type than those already evaluated in the UFSAR. There are no physical changes in plant equipment. The SRM system will continue to meet its design basis function. The method of rod withdrawal to be used is not in conflict with any existing basis requirements and complies with all requirements to minimize control rod worth.

A search of all appropriate basis documents revealed no requirements concerning the state of control rods adjacent to the SRMs at any time during, before, or after criticality is achieved. The safety limit MCPR as discussed in Technical Specifications is not challenged by this change. This change improves the technical accuracy of the UFSAR without changing the margin to safety at the DAEC.

SE 97-13 UFSAR Change: Standby Liquid Control (SBLC) Room Air Temperature

Description and Basis for Change

This evaluation was completed to revise the UFSAR based on actual plant HVAC design configuration. The UFSAR section concerning the SBLC system area temperature was revised to state "maintain the air temperature within the range of 68° to 90° F", instead of 70° to 100° F. The normal range is 68 to 90° F and the area can reach 103° F during a Reactor Water Cleanup (RWCU) line break accident. No actual changes to plant configuration were performed.

The design safety standards for SBLC show SBLC as designed to avoid the unacceptable results for the special event-shutdown without control rods. The safety system auxiliaries required for SBLC are pump power, tank heaters and valve firing circuits, and RWCU Isolation.

Summary of Safety Evaluation

The change in the room area temperatures from 70 to 100° F, to 68 to 90° F will not increase the probability of occurrence of an accident evaluated in the SAR. The slight change in the design range of room temperatures from 70 to 100° F, to 68 to 90° F cannot initiate an accident. The Reactor Building Heating System and Unit Heaters are non-safety related equipment. The electric heaters in the SBLC tank and the heat tracing will maintain the solution temperature above 75° F when the room air temperatures are below 75° F, and room air temperatures will not maintain the solution above 75° F. The tank heater is designed to maintain the temperature at 80° F +/- 5° F with 40° F ambient air temperatures. The capability of the heat tracing to maintain 80° F with a 50° F ambient air temperature was verified by calculation. These temperatures (40° F and 50° F) are far below the design temperature of the room air of 68° F. The SBLC system is required to be operable in the event of a plant power failure. Therefore, the pumps, heaters, valves and controls are powered from the standby AC power supply or the DC power supply in the absence of normal power. Therefore, this change does not increase the probability of occurrence of an accident evaluated previously in the SAR.

The change in range of air temperatures from 70 to 100° F, to 68 to 90° F will not increase the consequences of an accident. The actual solution temperature is maintained at or above 75° F by tank heaters and heat tracing on the suction piping. The lower range air temperature of 68° F ambient air temperature is above the actual saturation temperature (65° F) of the maximum allowed Sodium Pentaborate solution concentration (14.6%). The SBLC solution temperature is monitored by temperature indicators and a low temperature alarm (70° F). The temperature is checked and recorded at least once per day per Technical Specifications. The proper functioning of the heaters is also checked by verifying the tank and piping temperatures are above 75° F. The concentration is checked and recorded at least once per month per Technical Specifications. Therefore, this change does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR, and it does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. Also, the possibility of 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 margin of safety as defined in the basis for any Technical Specification is not reduced.

SE 97-16 UFSAR Change: Chapters 6 and 9 Rewording

Description and Basis for Change

This UFSAR change clarified the wording associated with Secondary Containment, carbon adsorber testing, compressed air systems, reactor building heating, control building HVAC and filtration, and miscellaneous safety related ventilation systems. Changes were made to correct the UFSAR information to match what was already depicted in design documentation, and industry standards. These changes were made for clarification of various UFSAR sections, to make the UFSAR more grammatically correct, and to minimize the possibility of misconstruing the intent of the UFSAR text. This change clarifies system description information. None of the changes deviate from the design intent of the plant. The revised wording does not impact the ability of a safety system to perform its safety function. Specific changes include the following:

* There are five secondary containment automatic initiation signals. The UFSAR previously discussed three signals. Initial plant design consisted of four trip inputs (not three); they were High Drywell Pressure, Low Reactor Pressure Vessel Level, High Radiation Refuel Floor Exhaust, and High Radiation Reactor Building Exhaust. The fifth trip (Offgas Vent Pipe High Radiation) was added. The five signals were already addressed in other UFSAR sections.

* Various editorial clarifications were made. The reference to "Freon" as the evaporator coil cooling media was changed to "refrigerant", since Freon is a trade name of DuPont refrigerants. The use of appropriate refrigerants continues to assure the proper operation of the evaporator coils.

* At one time each Standby Filter Unit (SFU) train had a hot water heating coil installed to supplement the electric heating coil. The water coil was a multipurpose coil which could be used for heating or cooling. However, the coils in the SFU were used for heating only. The "heating-cooling" coils were removed under a Plant Modification and the effect on the plant was evaluated under the modification. The UFSAR has been revised to reflect this change.

* In regard to the Reactor Building Heating and Ventilation Temperature Controller, the UFSAR stated that outside air is tempered to "56-68" ° F. This was in conflict with an UFSAR Figure. The actual controller set-point is 54-72° F. This difference is insignificant and more accurately reflects the actual operation of

the temperature controller. The reset schedule of 54 to 72° F is the original range by design. This range has not changed. This change eliminated the discrepancy between the UFSAR text and figure and makes the UFSAR accurately reflect plant design.

* Another UFSAR change involved removing the reference to "Refrigerant-112" and replacing it with "Refrigerant-11 or equivalent". Current carbon adsorber test standards recommend using R-11 over R-112. Performing testing with Refrigerant-11 or equivalent instead of Refrigerant-112 will continue to ensure that charcoal filter bypass is within allowable limits.

* The UFSAR did suggest that one of the safety functions of the Engineered Safety Features Ventilation System is to protect the safeguards equipment against freezing. Although it is a function of the system to cool and heat, heating is not a safety function. Operating the safeguards equipment generates sufficient heat to avoid freezing. Consequently, heating is not necessary for safe plant operation and is not considered a safety design bases.

* Another change involved removing the discussion of the Secondary Alarm Station (SAS) ventilation. The SAS was removed from the Control Room under a previous modification.

* Previously an UFSAR Table concerning Standby Filter Unit Testing suggested that fan performance be compared against "preoperational balancing characteristics", and that the electric coil meet the "designed preheat requirement". Current Technical Specifications require that the SFU flow be 1000 ± 10%. Required heater output is tested to be at least 22 kW in accordance with the Surveillance Test Procedure. The performance requirements are clearly established and are now identified as "required flow", and "required value" in the UFSAR Table. This does not constitute a change in DAEC's testing philosophy, or operating philosophy. The "preoperational balancing characteristics" and "designed preheat requirement" are the "required values".

Summary of Safety Evaluation

This change does not increase the probability of occurrence of an accident evaluated previously in the SAR. This change clarifies system description information. None of the changes deviate from the present design intent of the plant. The revised wording does not impact the ability of a safety system to perform its safety function.

The input functions and the output functions of the Design Safety Standards were reviewed for Reactor Building Ventilation system, Control Room Heating, Ventilating, and Air Conditioning system, Standby Gas Treatment system, and Equipment Area Cooling system. In all cases, the consequence of each accident evaluated previously is sufficiently extreme so as to envelope the consequence of any potential accident resulting from this change. Therefore, this change does not increase the consequences of an accident evaluated previously in the SAR.

This change does not affect plant drawings, or actual plant configuration in any way. The new UFSAR wording was evaluated against the Design Safety Standards, the Nuclear Safety Criteria for Boiling Water Reactors, the Nuclear Safety Criteria, the Classification of DAEC Structures, Systems, and Components, and the Nuclear Safety Operational Analysis (NSOA). The UFSAR changes remain well within the design criteria established in the documents discussed above. Therefore, this change does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR.

This change does not create the possibility of an accident of a different type than any evaluated previously in the SAR, and it does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. Forty-four events are identified in the NSOA for planned operations. The events were reviewed and found to be sufficiently complete such that they bound any accident that could be caused by the item under evaluation. Therefore, the likelihood of a different accident is not increased.

Design Safety Standards (DSS) were reviewed for the Reactor Building Ventilation System, Control Room Heating, Ventilating and Air Conditioning System, Standby Gas Treatment System, and the Equipment Area Cooling System. The output functions for each of the systems reviewed were analyzed for their potential to malfunction as each relates to the item under evaluation. Based upon this review, the possibility of a malfunction is not increased due to this UFSAR change. This change does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR.

Sections of the Technical Specifications which address topics, functions, and/or systems associated with this change were evaluated. Neither the body, nor the Bases of the sections reviewed were impacted by this UFSAR change. Therefore, this change does not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 97-17 UFSAR Change: Shutdown Outside Control Room

Description and Basis for Change

This evaluation justified changes to the UFSAR to be consistent with current plant procedures. The UFSAR indicated scram capability exists outside the control room through manual tripping of the control rod drive power supplies. Also, an UFSAR table stated that manual scram capability is possible by deenergizing Control Rod Drive (CRD) power supplies and/or manual closure of Main Steam Isolation Valves (MSIV)s by deenergizing power supplies, and tripping the main turbine front standard is a backup method of initiating a reactor scram. However, the guidance in an Abnormal Operating Procedure (AOP) specifies that the Average Power Range Monitoring (APRM) power supplies be tripped, with venting of the scram air header as a backup means of initiating the scram.

The original method described in the SAR for scramming the reactor from outside the control room was to trip the main turbine stop valves at the turbine front standard. This method required operation of logic located within the main control room. Operation of this logic cannot be assured during or after a main control room fire. The SAR change that was developed in support of a modification which installed the remote shutdown panels stated that scram capability exists outside the control room through manual tripping of the control rod drive power supplies. This is accomplished directly by tripping breakers, which results in not only a reactor scram but also a Group 1 (MSIV) Primary Containment Isolation System (PCIS) isolation. However, the intent of the modification was to maximize operator flexibility and minimize needless transients to the reactor. Tripping the breakers for the APRM power supplies also results in a scram but maintains the availability of the turbine and main condenser to remove decay heat and control reactor pressure. The ability to close the MSIVs if necessary exists at the remote shutdown panel. Therefore, the current procedural guidance in the AOP is more appropriate than the method described previously in the SAR. Other means, such as venting the scram air header, also exist that can effect a scram without limiting operator flexibility in dealing with the resulting transient. The changes described do not involve physical changes to the plant.

Summary of Safety Evaluation

The method previously described in the SAR for initiating a reactor scram from outside the control room would have resulted in closure of the MSIVs. Closure of the MSIVs is an analyzed event. However, the method described in the AOP does not result in MSIV closure. Therefore,

the probability of this type of event is reduced. In addition, while tripping the breakers for the CRD power supplies provides a direct means of initiating a reactor scram, the probability of a "hot short" which would prevent a scram from going to completion is greater than by tripping the power range monitor power supplies. The as-built configuration of the alternate shutdown panel and guidance in the AOP provide acceptable means of initiating a reactor shutdown from outside the control room. Therefore, a SAR change is appropriate. Initiating a backup reactor scram by initiating a manual turbine trip from the turbine front standard in the manner previously described in the SAR would have directly resulted in a transient that has been evaluated in the accident analysis. Also, there is no guarantee that this method would have successfully initiated a scram as it required operation of cables and logic within the control room fire area. Venting the scram air header in the method described in the AOP does not result in a turbine trip or other transient such as those described in the SAR. Therefore, this change does not increase the probability of occurrence of an accident evaluated previously in the SAR, and it does not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. This change does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. No physical changes to the plant are made by this change. The UFSAR was revised to be consistent with current plant procedures which support safe shutdown of the plant. Therefore, no new failure modes have been introduced as a result.

This UFSAR change does not increase the consequences of an accident evaluated previously in the SAR. There will be no increase in the radiological consequences of any previously analyzed SAR accident. This change does not degrade or prevent actions described or assumed in an accident discussed in the SAR. This change does not alter any assumptions previously made in evaluating the radiological consequences of an accident, nor will it play a direct role in mitigating the radiological consequences of an accident described in the SAR.

By deenergizing the APRM power supplies instead of deenergizing the CRD power supplies to initiate a scram from outside the control room the severity of the resulting transient to the reactor vessel is reduced while increasing operator flexibility to deal with the event by maintaining the availability of the main turbine and main condenser to remove decay heat. Initiating a backup scram by venting the scram air header also results in a lower severity of the resulting transient compared to initiating a turbine trip at the turbine front standard.

This change did not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The consequences of a failure to scram the reactor after tripping the CRD and MSIV solenoid power supplies or tripping the main turbine are the same or greater than by initiating a reactor scram alone by the methods described in the AOP.

This change did not create the possibility of an accident of a different type than any evaluated previously in the SAR. The failure to scram the reactor by the methods described in the AOP will not affect any systems, structures or components important to safety in a way not previously evaluated. No physical changes to the plant are made by this change. Therefore, no new failure modes have been introduced as a result.

This change provides a more reliable means of initiating reactor scrams from outside the control room without resulting in severe plant transients which would challenge the margin of safety as defined in the Technical Specifications. Therefore, this change does not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 97-18 Affect of Appendix R Re-baseline Program and Thermo-lag Reduction Initiatives

Description and Basis for Change

The purpose of this evaluation was to evaluate the overall effect of plant and program changes resulting from implementation of the Appendix R Re-baseline Project and Thermo-lag reduction initiatives. The Appendix R Re-baseline Project re-reviewed the DAEC position with respect to the criteria contained in 10CFR50 Appendix R and NRC clarification documents such as Generic Letters (GLs) 85-01, 85-09, 86-10 and 86-10 Supplement 1. The Thermo-lag reduction initiative eliminated reliance on Thermo-lag as a fire barrier in all areas of the plant except one. This use of Thermo-lag has been re-evaluated by the Fire Protection Engineering Group (FPEG) and is acceptable. This safety evaluation is in response to questions raised at a Safety Committee meeting. These questions centered on the acceptability of manual actions and the effect of the changes brought about in response to the Re-baseline and Thermo-lag Projects with respect to overall plant safety. This safety evaluation did not look into each modification, program and initiative, in detail, as the impact on plant safety and the safety significance of each individual modification, program and initiative was analyzed by the safety evaluation prepared as an integral part of that work activity.

Summary of Safety Evaluation

The changes 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 SAR. The Re-baseline analysis comprehensively verified and re-evaluated the DAEC position with respect to the requirements of Sections III. G, III. J and III. O of Appendix R to 10CFR50. The results of the Re-baseline Project are a reduction in the number of fire areas and fire barriers required. These fire barriers included barriers made up partially or entirely of Thermo-lag. In addition, the project team identified errors in the original analysis. These errors may have adversely impacted the DAEC's ability to safely shut down the DAEC in the event of a fire.

The conclusions of the Re-baseline Project are that the DAEC can safely shut down for a fire in any single plant Fire Area. The Re-baseline analyses are based on the original analysis methodology. However these analyses utilize more conservative assumptions, are better documented and more detailed than the original analysis. Manual actions required to mitigate the consequences of a fire have been evaluated and determined to not adversely impact the ability to shutdown the DAEC.

Re-baseline manual actions may include repairs as defined by the NRC. In accordance with guidance provided by Appendix R and its associated clarification documents, these repairs are not associated with systems, structures or components required to achieve hot standby conditions. Repairs are only associated with components required to transition from hot standby to cold shutdown and those components that are required to achieve and maintain cold shutdown conditions. Appendix R and its guidance documents allow such repairs. These repairs are described in the Fire Area Analyses for each Fire Area and in an Abnormal Operating Procedure (AOP).

The manual action evaluations considered the availability of emergency lighting, time available, re-entry into the fire area, personnel availability and communication requirements. The result of the manual action evaluation indicates that the proposed manual actions are acceptable and that performance of the proposed manual actions would not place the DAEC into an unanalyzed condition. Therefore, the implementation of the Re-baseline Project and the Thermo-lag reduction initiatives would 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 would not be increased.

The results of the Re-baseline Project are a reduction in the number of fire areas and fire barriers required. The conclusions of the Re-baseline project are that the DAEC can safely shut down for a fire in any single plant Fire Area. The Re-baseline analyses are based on the original analysis methodology. However these analyses are more conservative, better documented and more detailed than the original analysis. Therefore, the implementation of the Thermo-lag reduction initiatives and the Re-baseline Project would not create the possibility of an accident of a different type than any evaluated previously in the SAR, and it would not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR.

The re-baseline project resulted in a net reduction in the number of raceway fire wraps required. These wraps were originally constructed of Thermo-lag. Recent analyses indicate that the derating factors of Thermo-lag protected raceway were not correct and that additional derating may have been required. By removing the Thermo-lag, the raceway is returned to its previously analyzed condition and the probability of a cable failure within a protected raceway are reduced. The conclusions of the Re-baseline Project are that the DAEC can safely shut down for a fire in any single plant Fire Area. The Re-baseline analyses are based on the original analysis methodology. However these analyses are better documented and more detailed than the original analysis. Manual actions, including repairs, required to mitigate the consequences of a fire have been evaluated and determined to not adversely impact the ability to shutdown the DAEC and that performance of the proposed manual actions would not result in an equipment malfunction or place the DAEC into an unanalyzed condition. Therefore, the implementation of the Re-baseline Project and the Thermo-lag reduction initiatives would not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR and it would not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR.

Fire Protection features at the DAEC are not addressed by the Technical Specifications. The implementation of the Re-baseline project and the Thermo-lag reduction initiatives would not impact the margin of safety as defined in the basis for any Technical Specification.

SE 97-19 UFSAR Change: Well Water (WW) and Reactor Building Closed Cooling Water (RBCCW)

Description and Basis for Change

The purpose of this change was to update the UFSAR to reflect several changes resulting from a previous plant modification, and to correct typographical errors which existed.

The DAEC WW system consists of four independent wells. Three had low capacity pumps with 750 gpm capacity and one had a 1650 gpm capacity. After the installation of a new "B" well, the "B" well capacity has increased to 1200 gpm. Therefore, the UFSAR has been revised to reflect this change. Now the "B" or "D" pumps are capable of supplying all the loads with no other well water pumps in service. DAEC now has two high capacity pumps and two low capacity pumps.

A note was added to the UFSAR to indicate that the gpm values next to the equipment supplied by WW are nominal only and were used for system design. Equipment important to safety is supplied by Emergency Service Water (ESW) should WW become unavailable.

Concerning RBCCW instrumentation requirements, the UFSAR stated that the temperature and pressure of the cooling water (RBCCW) are indicated in the control room. However, the control room has had only a pressure indicator for RBCCW since the inception of the plant. The inplant operator monitors RBCCW temperature on a routine basis. As an additional safeguard measure the high alarm point for the RBCCW heat exchangers outlet temperature is set at 115° F, and will alarm in control room.

The UFSAR section concerning RBCCW surge tank automatic filling has been revised to indicate the RBCCW surge tank is filled manually rather than automatically. Manual filling provides the operators with positive control of RBCCW surge tank makeup. RBCCW surge tank level is periodically monitored. Normally the tank may need to be filled once or twice in a cycle. The tank low level alarm is set at two feet which allows the operator enough time to fill it.

The UFSAR was also revised to indicate use of a mobile demineralizer to makeup demineralized water for plant use, instead of using plant original equipment. The mobile unit has been specifically designed to produce water acceptable for use within the nuclear power industry. The plant typically makes water periodically to maintain adequate inventory in the storage tank(s).

Summary of Safety Evaluation

The changes evaluated in this safety evaluation are only to change the UFSAR. There are no accidents related to WW or RBCCW which have been evaluated in the SAR. All containment isolation functions for WW and RBCCW remain unaffected. These changes do not effect any equipment important to safety. Therefore, these changes do not increase the probability of occurrence of an accident evaluated previously in the SAR, and they do not increase the consequences of an accident evaluated previously in the SAR, nor do they increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The changes do not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR, and they do not create the possibility of an accident of a different type than any evaluated previously in the SAR. The possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR is not created. There are no bases for any Technical Specifications associated with the WW or RBCCW systems. Therefore, the changes do not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 97-20 UFSAR Change: Neutron Monitoring System (NMS) Logic

Description and Basis for Change

The UFSAR incorrectly stated that the NMS logics are part of the Reactor Protection System (RPS). This was in conflict with the design basis for the NMS and RPS. The UFSAR was revised to reflect that the NMS channels and logics are considered part of the NMS. The UFSAR was also revised to reflect this change in the system classification of the NMS logics.

Summary of Safety Evaluation

The change in system classification of the NMS logics from RPS to NMS is one of interpretation. No physical changes are made to the facility. Therefore, the probability of occurrence of an accident evaluated previously in the SAR is unchanged. No physical changes were made to the facility. Therefore, the consequences of an accident evaluated previously in the SAR are unchanged and the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR is unchanged. The consequences of a malfunction of equipment important to safety evaluated previously in the SAR are unchanged and the possibility of an accident of a different type than any evaluated previously in the SAR is not created. 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 number of operable IRM and APRM channels, as required by the Technical Specifications for plant operation, is unaffected by the change in system classification. There is no analytic limit associated with the IRM upscale trip. The analytic limits for the APRMs are not affected by this change, nor are the nominal trip setpoints which represent the margin of safety to those analytic limits. Also, the flow-biased scram function of the APRM channels in RUN mode is unchanged. Therefore, no margin of safety as defined in the basis for any Technical Specification is reduced.

SE 97-23 UFSAR Change: Jockey Fire Pump Accumulator

Description and Basis for Change

An UFSAR change has been made which removed the Fire Protection system jockey fire pump accumulator from the DAEC UFSAR.
The accumulator was valved out of service by a Design Document Change (DDC). The function of the accumulator, or surge tank, was to assist the jockey or keep-fill pump in maintaining fire main pressure to prevent cycling of the main fire pumps and to eliminate water hammer or other transients. The accumulator was installed to assist the jockey pump and reduce cycling of the jockey pump. Improvements to the jockey pump system and improved leak tightness of the fire water system has rendered the surge tank of marginal additional value to the system.

Summary of Safety Evaluation

Fire is not an entry condition, basis or an assumption for any accident previously evaluated in the SAR. The safety significance of removal of the Fire Protection system accumulator from service was previously evaluated by a DDC. The Fire Protection system accumulator is not required by the Branch Technical Position (BTP) or the National Fire Protection Association (NFPA) 20. The surge tank (accumulator) provides marginal additional value to the system. Observation of the system indicates acceptable system performance without the accumulator. The duty cycle of the jockey pump with the accumulator removed from service is acceptable, therefore a revision to the UFSAR to remove the reference to the accumulator is acceptable. The valving out of the accumulator and the associated UFSAR change will not increase the probability of occurrence of an accident evaluated previously in the SAR, and it will not increase the consequences of an accident evaluated previously in the SAR. It will not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the SAR. The valving out of the accumulator and the associated UFSAR

change will not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This change does not create the possibility of an accident of a different type than any evaluated previously in the SAR, and it does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. Fire Protection features at the DAEC are not addressed by the Technical Specifications. The removal of the accumulator from service and the associated UFSAR change do not impact the margin of safety as defined in the basis for any Technical Specification.

SE 97-24 Change To INPO Accredited Licensed Operator Training Program

Description and Basis for Change

This evaluation was performed to determine the impact of substituting an accredited training program for Licensed Operator initial and requalification training. Per NRC Generic Letter 87-07, prior to implementing the option of substituting an accredited training program for initial and requalification training programs previously approved by the NRC, written notification to the NRC is required, and it is necessary to certify that the substitute training program is both accredited and based upon a systems approach to training. After a thorough review of NRC docketed correspondence, neither the NRC nor IES has been able to find such written notification. In 1986, DAEC elected to exercise the option to substitute an Institute of Nuclear Power Operations (INPO) accredited Licensed Operator Training Program in lieu of the alternate requirements of 10CFR55.31(a) and 10CFR55.59(c). This substitute training program has been developed and administered using a systematic approach to training and has been accredited by INPO since 1986.

No plant systems or components are affected by this change. The DAEC Updated Final Safety Analysis Report (UFSAR) was previously updated per 10CFR50.71(e) in 1990 to include recognition that the DAEC Operator Training Program was accredited by INPO. The UFSAR was revised again in 1995 to note that the DAEC Operator Training Program was based on a systematic approach to training accredited by the National Academy for Nuclear Training.

Summary of Safety Evaluation

The change to an INPO-accredited Operator Training Program is approved by the NRC per 10 CFR 55.31(a), 55.59(c) and Generic Letter 87-07. This change does not alter the scope or reduce the effectiveness of the Operator Training Program. As a result, the probability of occurrence of "operator

errors" as initiating or contributing actions to the accidents evaluated in the UFSAR is not increased, and this change does not increase the consequences of an accident evaluated previously in the SAR. This change does not affect any structures, systems or components important to safety evaluated in the UFSAR. Consequently, this change does not increase the probability of occurrence of any related equipment malfunctions, nor does it increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. This change does not create the possibility of an accident of a different type than any evaluated previously in the SAR, and it does 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 does not reduce the margin of safety as defined in the basis for any Technical Specification.

SE 97-25 UFSAR Change: Vessel Head Flange Leakage Detection and Collection System

Description and Basis for Change

This evaluation was performed to support the resolution of the UFSAR description for the Vessel Head Flange Leakage Detection and Collection System being different from the actual plant configuration.

The previous UFSAR description contained reference to two remote-solenoid-operated valves, a collection volume of one fourth of a gallon, and a level detector, none of which were installed at the DAEC during construction. The pressure switch, correctly described previously in the UFSAR, is installed and is connected to a control room annunciator window to make operators aware of the failure of the inner metallic gasket. Two locally-operated manual valves are supplied instead of the remotely operated valves previously described. In the installed configuration, there is no provision for relieving the pressure in the leak detection system while at power.

Summary of Safety Evaluation

This change has no effect on the probability of occurrence of any accident which has previously been evaluated in the SAR. Neither the system as described previously in the UFSAR nor the system as it is actually installed will cause any automatic actions to occur. The only safety function associated with this system is as a passive boundary to reactor coolant. The probability of a catastrophic failure of the reactor pressure boundary is not increased. Since there are fewer parts and connections, it is actually decreased. The system did not have any impact on the

consequences of any accident previously evaluated, and therefore will not increase the consequences of any accident previously evaluated in the SAR. Since this system still provides operators with a warning of the failure of the first head flange metallic gasket, and because it has fewer components subject to reactor pressure, there is no scenario for which the installed configuration increases the probability of occurrence of a malfunction of any equipment important to safety which has previously been evaluated in the SAR. Because this system has not had any impact on the consequences of any of the malfunctions of equipment important to safety which were previously evaluated, and it still does not have any impact, it will not increase the consequences of any malfunction of equipment previously evaluated in the SAR.

Since the only safety function associated with this system is that of passive reactor pressure boundary, and no penetration is made of the primary containment, the only type of accident which may apply to this system is the small break LOCA inside containment, which is not a different type of accident. It has been analyzed previously, so this change does not create the possibility of an accident of a different type than any evaluated previously in the SAR. The only type of malfunction possible which is important to safety is a rupture of the pressure boundary, which has been analyzed extensively. Therefore, this change does not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR. The only Technical Specification which is impacted by this system deals with Reactor Coolant leakage into Primary Containment. This change does not impact the expected leakage into the Drywell Equipment Drain Sump or the Drywell Floor Drain Sump. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

SE 97-27 UFSAR Change: Core Spray System Locked Closed Valves

Description and Basis for Change

The purpose of this UFSAR change was to clarify the portion of the containment integrity and valve arrangement description which stated, "A vent line is provided between the two shutoff valves that can be used to measure leakage through the inside check valve or the inboard shutoff valve. To ensure containment integrity, the vent line is normally closed with two valves (at least one of which is locked closed) and a pipe cap." This UFSAR change removed "(at least one of which is locked closed)" from this statement.

This statement referred to the 3/4" lines which tap off between the Core Spray Inject Motor-Operated Valves. These lines are outside of the

primary containment isolation and reactor coolant pressure boundaries. The subject valves have never been included and are currently not included in the locked closed valve list and are currently not locked closed. Per plant drawings, the inboard vent valves show a "NC" (Normally Closed) designation, which is the normal position for the subject valves. The isolation valves which provide primary containment isolation and reactor coolant pressure boundaries are the inboard inject line check valves and the inboard inject valves.

Summary of Safety Evaluation

This change does not increase the probability of occurrence of an accident previously evaluated in the SAR. A failure of the valves to be closed can be considered to be equivalent to a break in the subject piping and has been evaluated (or is within the boundary of other evaluated breaks) in the SAR. The probability of the valves not being closed is not considered to be credible. The consequences of an accident previously evaluated in the SAR are not increased because the installed line and subject valves are normally closed (and will remain so, as required per a plant drawing, an Operating Instruction, as well as other plant procedures) such that they have no effect on the performance of the Core Spray system. The installed line and subject valves are outside of the primary containment isolation and reactor coolant pressure boundaries such that they have no effect on the containment isolation or reactor coolant pressure boundary functions. Thus, the design basis for the containment isolation and reactor coolant pressure boundary functions are not affected and the consequences of an accident previously evaluated in the SAR is not increased.

The probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is not increased because, although the positions of these valves may contribute to a malfunction of the Core Spray system due to misdirecting water from the reactor vessel, the closed positions of the valves are adequately controlled, such that the failure of the valves to be closed is not considered to be credible. This change does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the SAR. The possibility for an accident of a different type than any evaluated previously in the SAR is not created because the maximum potential adverse effect of not locking the valves is that the subject valves are left open. However, procedures are in place such that the failure of the valves to be closed is not considered to be credible.

The possibility for a malfunction of equipment important to safety of a different type than any evaluated previously in the SAR is not created. The installed line and subject valves are outside of the primary

containment isolation and reactor coolant pressure boundaries and they have no effect on the containment isolation or reactor coolant pressure boundary functions. Thus, the design basis for the containment isolation and reactor coolant pressure boundary functions are not affected. The margin of safety is not affected because the closed positions of the valves are adequately controlled. This will ensure that the Core Spray system can meet all of its design and licensing basis requirements. The design basis and licensing basis for the containment isolation and reactor coolant pressure boundary functions are not affected and the margin of safety remains unaffected.

SE 97-29 UFSAR Change: Switchyard Hiawatha Line Disconnect

Description and Basis for Change

The DAEC Main Generator supplies power to the auxiliary transformer and the DAEC switchyard through the main transformer. The main transformer feeds the 161KV "West Bus" via the "I" breaker and the "East Bus" via the "H" breaker and "G" breaker. The Hiawatha line is fed from the "H" and "G" breakers. The previous configuration did not allow maintenance on the Hiawatha line without opening both the "H" and "G" breakers, disconnecting the line, and reconfiguring the protective metering, to allow reclosure of the "H" breaker. This required extensive maintenance and increased operating time with only one generator output breaker closed, the "I" breaker.

It is best operating practice to minimize the operating time with only one generator output breaker closed. The addition of the Hiawatha line disconnect improves plant availability by reducing the risk of operating with only one main generator output breaker closed when the Hiawatha line is being removed from service for maintenance. By adding the disconnect on the Hiawatha line side of the "H" breaker, once opened, both the "H" and "G" breakers can be reclosed and the protective metering will be intact. By installing the disconnect, the Hiawatha line may be removed from service for extended periods of time with minimal impact on plant maintenance and operations. This evaluation was performed in support of revising the UFSAR switchyard schematic to include the disconnect.

Summary of Safety Evaluation

This modification is non-safety related and provides increased maintenance opportunities and operational flexibility. The switchyard disconnect switches are not described in the UFSAR. The installation of the disconnect switch does not increase the probability of a plant transient.

It allows the "H" generator output breaker to be reclosed once the Hiawatha line has been isolated to prevent a single failure of the "I" generator output breaker from tripping the DAEC off-line. The generator, main transformer, new disconnect, Hiawatha line, generator output breakers, and operation of the new component have no affect on any of the accidents evaluated in the UFSAR. Therefore, this change does not increase the probability of occurrence of an accident evaluated previously in the SAR, and it does not increase the consequences of an accident evaluated previously in the SAR. The act of removing the Hiawatha line from service is simplified by the addition of the disconnect, hence reducing the potential for an error or malfunction. This change does not increase the probability of occurrence of an equipment malfunction. The addition of the disconnect switch for the Hiawatha line does not affect the operation of the DAEC switchyard to transmit power to the IES grid. The disconnect addition will not increase the consequences of an equipment malfunction, and it will not create an accident of a different type than any evaluated previously in the SAR. 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 disconnect switch and its associated switchyard equipment are not addressed in Technical Specifications, hence this change does not reduce any margin of safety as defined in the basis for any Technical Specification.

Section C - Experiments

This section has been prepared in accordance with the requirements of 10 CFR Section 50.59(b). No experiments were conducted during the period beginning October 1, 1995 and ending March 1, 1997.

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, 1995 and ending March 1, 1997 did not alter our commitment to the NRC guidelines contained in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

Revision No.	Description of Change
31	Minor non-technical changes were made, such as title changes. Plan was reformatted. Organizational changes were incorporated and responsibilities were clarified. Added insurance notification guidelines for impaired active fire protection systems. Clarified purpose of referenced Fire Protection administrative procedures.
32	Deleted a fire brigade roster containing personnel information.

Section E - Commitment Changes

The information contained in this section identifies and briefly describes various commitment changes which occurred during the period beginning October 1, 1995 and ending March 1, 1997. The changes described were evaluated and are being reported per the Nuclear Energy Institute's "Guideline For Managing NRC Commitments", dated December 19, 1995.

C 96-004 In NG-89-0613, "Response to Notice of Violation Transmitted with Inspection Report 88-022", dated February 23, 1989, DAEC committed to having a continuing education program to provide annual 10CFR50.59 training to the technical staff. Since this commitment was made it has been determined that annual requalification is unnecessary and excessively burdensome, considering the relatively small number of evaluations performed as compared to the large number of personnel in the training program. Therefore, this commitment has been revised. Individuals performing and reviewing safety evaluations still receive initial 10CFR50.59 training and supervisors are responsible for determining the need for refresher training.

C 96-005 In NG-87-3176, "Response to NRC Inspection Report 87-016", dated August 21, 1987, in response to a Notice of Violation concerning a fire watch not being maintained for a half hour after completion of welding activities, DAEC committed to the Fire Marshall meeting with contractor personnel before commencement of hot work activities, and to have the Fire Marshall monitor fire watch activities to effectively communicate the need for strict procedural compliance to all workers and to take immediate corrective action as required to prevent future incidents. DAEC also committed to having the Fire Marshall summarize his comments, actions, and observations in a monthly fire inspection report and forward them to higher levels of management.

This commitment has been revised because DAEC no longer has contract administrators, the Fire Marshall is not able to monitor all fire watch activities during periods of peak activity, and there is no need for a monthly Fire Marshall inspection report.

Currently the Fire Marshall regularly communicates to plant management through the Fire Protection Engineering Supervisor. Level I firewatches who will be supervising hot work activities are trained and required to demonstrate they understand hot work requirements and their individual responsibilities prior to performing the job of firewatch. Work supervisors are required to complete the precautions section on the Hot Work Permit which directs that the firewatch be provided during and for 30 minutes after the completion of the hot work. The Fire Marshall periodically visits

plant locations where hot work is being performed to assure procedural requirements are met.

The revised process is more effective since the responsible supervisor filling out the checklist for work will know that a 30 minute firewatch is required and will direct personnel accordingly. The revised process will continue to assure appropriate firewatch activities through a checklist and training.

C 96-006 In Licensee Event Report 92-002, "Missed Fire Watch Due to Procedural Inadequacies", dated February 25, 1992, DAEC committed to revising certain procedures to require a firewatch when vent fans are secured and the Turbine Building Sump Room door is blocked open to prevent hydrogen buildup.

The condition requiring the door to be open during performance of the procedures no longer exists. Therefore the door is no longer opened to perform the procedures and a firewatch is no longer required. The procedures have been revised to delete the requirement for a firewatch.

C 96-010 In NG-86-4125, "Request for Modification of Commitments to NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", dated November 21, 1986, DAEC committed to performing a PT (dye-penetrant) examination of the Control Rod Drive Return Line (CRDRL) nozzle once every four cycles. This was not required by NUREG-0619. This commitment appears to have been made due to misinterpretation of the NUREG. The NUREG required a final PT following modifications, not a repetitive PT exam. Therefore, the commitment to perform a PT examination of the CRDRL nozzle once every four cycles is withdrawn.