SECTION A - PLANT DESIGN CHANGES

This section contains brief descriptions of and reasons for plant design changes completed during the calendar year 1989 and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR Part 50.59(b).

The basis for inclusion of a Design Change Package (DCP) in this report is closure of the package at the Duane Arnold Energy Center (DAEC) in the calendar year 1989. It is noted that portions of some DCPs listed were partially closed in previous years.

In addition, various Minor Modifications (MMs) were completed during the calendar year 1989. MMs are not included in this report since, by definition, they do not change the facility as described in the Final Safety Analysis Report. Each MM was screened to ensure that no unreviewed safety questions were involved.

DCP 1122 Turbine Generator Bearing Fire Protection

Basis for Change: In response to a recommendation made by American Nuclear Insurers to install fire protection equipment for the turbine generator bearings and unguarded oil lines beneath the turbine lagging, a CO2 Fire Suppression System to protect the Exciter housing and a Pre-Action Turbine-Generator Main Bearing Spray System were installed.

CO2 Fire Suppression System

Description of Change: The CO2 Fire Suppression System uses the pre-existing ten ton CARDOX tank and CO2 Vaporizer as the source of CO2. An additional pipeline was installed from the CARDOX System CO2 Vaporizer manifold to the Main Generator purge line to provide a flowpath to the vicinity of the Exciter housing. An additional line was installed from the Main Generator purge line to a pre-existing branch line in the Exciter housing. The new pipe installed in the vicinity of the CARDOX tank is supported by the free-standing seismic category I platform constructed under a previous design change. Four smoke detectors were also installed in the Exciter's ventilation exhaust airstream to provide a Control Room alarm if smoke particulates are present.

Summary of Safety Evaluation: The Exciter CO2 Fire Suppression System is independent of the Cable Spreading Room CO2 Fire Suppression System. The two systems share a common tank and riser pipe. Also, the systems have separate isolation valves for both the vapor pilot supply and the common riser spool. The common riser component does not change the CO2 flowpath or make it more complex. Since the Exciter CO2 Fire Suppression System is initiated manually, no new inadvertent Generator-Turbine-Reactor trips have been created. The new CO2 pipe crosses above a safety-related conduit carrying cables relied upon to provide safe shutdown capability from outside the Control Room for compliance with 10 CFR 50, Appendix R. This modification does not affect compliance with 10 CFR 50, Appendix R, since systems installed to ensure post-fire safe-shutdown capability need not be designed to meet seismic, single failure or other design basis accident criteria. This modification does not adversely impact any fire protection requirements or equipment needed to shutdown the plant safely. In the event that the Exciter CO2 Fire Suppression System is initiated and CO2 usage lowers the CARDOX tank level to less than the Technical Specifications require, a fire watch would immediately be initiated in the Cable Spreading Room as required by Technical Specifications. In addition, if the free standing platform above the CARDOX System were to fail and cause the portion of the Cable Spreading Room CO2 Fire Suppression System to become inoperative, the plant could still be safely shutdown. A fire watch would be initiated in the Cable Spreading Room as required by Technical Specifications. Even if a fire existed in the Cable Spreading Room concurrently with the collapse of the platform, safe shutdown could still be assured. Of the two Class 1E divisions of safety-related cable, only Division II is present in the Cable Spreading Room. Division I cable does not run through the Cable Spreading Room. It is routed directly into the Control Room from the Reactor Building providing redundant safe-shutdown capability. Remote shutdown capability is also available. All replaced or added riser pipe and fasteners that are common to both CO2 Systems were procured non-safety related to ensure compliance with applicable ASTM specifications. This modification did not involve an unreviewed safety question or Technical Specification change.

Pre-Action Turbine-Generator Main Bearing Spray System

Description of Change: The Pre-Action Bearing Spray System was expanded to include all Turbine-Generator main bearings. The system uses an electrical heat detection system with the detectors set to open a deluge valve solenoid at approximately 225°F. This allows water to partially charge the supply piping until a spray nozzle fusible link melts through in the affected area. Fusible link melt-through temperatures are 360°F and 286°F for the the high pressure turbine and balance of the system, respectively.

Summary of Safety Evaluation: The addition of the Pre-Action Spray System to protect the Turbine-Generator main bearings does not impact equipment relied upon to shutdown the reactor safely. One water supply pipe, feeding Generator bearing No. 8 branch line, is above and within a solid cone angle of 30 degrees (vertical axis) from a conduit carrying cables relied upon to provide safe shutdown capability from outside the Control Room for compliance with 10 CFR 50, Appendix R. This modification does not affect compliance with 10 CFR 50, Appendix R since systems installed to ensure post-fire safe-shutdown capability need not be designed to meet seismic, single failure or other design basis accident criteria. This modification does not adversely impact any fire protection requirements or equipment needed to shutdown the plant safely. Although the system is designed to actuate automatically, the possibility of inadvertent actuation causing water impingement on the main turbine is remote because both a spurious electrical signal and an open spray nozzle would have to occur. This modification did not involve an unreviewed safety question or a Technical Specification change.

DCP 1315 Low-Level Radwaste Processing and Storage Facility (LLRPSF)

Description and Basis for Change: The Low-Level Radioactive Waste Policy Amendments Act of 1986 mandated that, after January 1, 1986, existing burial sites could refuse to accept waste from sources outside of their region. Therefore, Iowa Electric built a LLRPSF under this DCP. Generic Letter (GL) 81-38 provides guidance for onsite storage of low-level radioactive waste (LLRW). This GL indicates that proposed increases in storage capacity for LLRW are to be evaluated under the provisions of 10 CFR 50.59. The additional storage capacity may be provided if 1) existing license conditions do not prohibit increased storage, 2) no unreviewed safety questions exist, and 3) the increased storage capacity does not exceed the expected radioactive waste generated for 5 years. The facility operating license and Technical Specifications for the DAEC do not prohibit increased onsite radioactive waste storage capacity. In addition, the proposed storage does not exceed the expected amount of radioactive waste generated at the DAEC for 5 years; the storage portion of the LLRPSF is designed to provide an interim onsite storage of up to 2.5 years for dewatered/solidified resins and dry active waste (DAW) produced at the DAEC until they can be shipped to a permanent disposal facility. Concerns over seismic qualification of the LLRPSF prohibit long-term storage of liquid radioactive waste inside of the facility. The LLRPSF will not be used for long-term storage of radioactive liquid until this issue is resolved; no unreviewed safety questions exist regarding current use of this facility. The processing portion of the LLRPSF is designed to process DAW, oil, laundry, contaminated tools, and used respirators. The LLRPSF is designed to protect against the release of radioactive substances to the environment.

Summary of Safety Evaluation: The LLRPSF is designed in accordance with the same criteria (seismic, tornado) as the existing radwaste building. The shielding configuration of the facility is designed to limit the offsite dose from onsite storage of resins and DAW to less than the limits of 40 CFR 190. The ventilation system provides the capability to filter exhaust through particulate air filters after an accident. A radiation effluent monitor is provided in the exhaust stack to monitor activity levels and is interconnected to the ventilation system to provide an isolation signal if activity levels exceed the limits imposed by 10 CFR 50, Appendix I. Liquid drainage systems are routed to the existing radwaste building for processing. A conservative accident analysis was performed to verify that a liquid release to the environment will not exceed the accident dose rates defined in 10 CFR 100. The storage portion of the LLRPSF is designed to protect against the effects of the maximum probable flood. This modification did not involve an unreviewed safety question. Technical Specifications were revised to include the additional radiation monitor.

DCP 1318 Breathing Air Enclosure

Description and Basis for Change: This modification provided for the construction of an enclosure to house existing Breathing Air Equipment (compressors, purifiers, air receivers, and air/lube oil coolers) adjacent to the Reactor Building railroad airlock. Breathing Air Equipment was reconfigured to allow construction of the enclosure. New piping, electrical conduit, lighting and power for system operation, trouble alarms in the Control Room and additional air filters were also provided.

Summary of Safety Evaluation: The Breathing Air System is not a safety-related system. The design basis of the system as stated in the UFSAR was not changed by this modification. The design intent of the Reactor Building railroad airlock penetration to provide for secondary containment integrity was not changed nor were any new penetrations added. Relocation and installation of the six air receivers does not affect the integrity of the new enclosure or the railroad airlock wall. One additional pipe hanger was installed to support the increased loading associated with the newly installed air filters. This modification provided for the permanent installation of the Breathing Air Equipment. The new enclosure protects the Breathing Air Equipment from the elements enhancing the reliability of the system. Failure of the Breathing Air Equipment would have no effect on the integrity of secondary containment or the ability to shutdown the plant safely. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1346 Miscellaneous Appendix R/Fire Protection Modifications

Description and Basis for Change: The modifications performed under this DCP resolve action items identified by: (1) the Appendix R Mock Audit Inspection Report performed for the DAEC; (2) the Inadvertent Actuation of Fire Suppression Systems Evaluation performed for the DAEC; and (3) the Underwriter's Laboratories Fire Door Inspection Report.

Emergency lighting packs with eight hour capability were added to provide adequate lighting in areas required to be entered by operations personnel during shutdown operations from outside the Control Room. Two solenoid valves that could have been affected by the actuation of a single sprinkler were replaced with NEMA 4 (watertight) rated valves to preclude inadvertent loss of power to control room supply fans. Fusible links were added to the closing mechanisms of various fire doors. Discrepancies found in as-built hardware in some fire doors were corrected by revising documentation to reflect the as-built hardware conditions. One non-U.L.-rated double swinging door was removed since it was not required for any purpose and was determined to be a personnel hazard due to the velocity with which it closes. These modifications were not safety-related nor did they effect the function or operability of safety-related systems or components.

Summary of Safety Evaluation: The addition of battery-backed emergency lighting improved the plant lighting system by providing additional lighting where required to illuminate adequately the access and egress paths between Remote Shutdown Panels and the Diesel Generator rooms. The additional lighting fixtures placed near the Hydraulic Control Units provide sufficient lighting for operations personnel to confirm a reactor scram after evacuating the Control Room. The emergency lighting added by this DCP does not effect the operability of any equipment or the seismic design of any structure; no hazard would be presented should any portion of these lights break loose and fall from its installed location. Battery packs and light units that could endanger safety-related components were seismically supported. The NEMA 4 (watertight) valves used to replace the pre-existing solenoid valves are identical to the valves that they replaced and did not alter the function or operation of the solenoid valves. The NEMA 4 rating assures proper operation of these solenoid valves during actuation of a fire suppression system. Removal of the non-fire rated door has no effect on fire protection capability or security since another fire-rated door is located in the same corridor. The existing fire-rated door is monitored by Security and provides the required fire barrier in the corridor. The addition of fusible links on both sides of three fire doors identified by the U.L. Inspection increases the assurance that these doors will close in the event of a fire. This modification did not involve an unreviewed safety question or a Technical Specification change.

DCP 1366 Permanent Hydrogen Water Chemistry System

Description and Basis for Change: A Hydrogen Water Chemistry (HWC) System was installed under this DCP to control intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel piping and components and to provide a source of hydrogen for Main Generator stator cooling. Control of IGSCC is accomplished by injecting hydrogen into the reactor feedwater. Under a neutron flux, the injected hydrogen combines with dissolved oxygen, effectively reducing the concentration of oxygen in the reactor coolant. This results in a reduction of the electro-chemical potential (ECP) of the coolant to levels below that required to favor the occurrence of IGSCC. In support of the HWC System installation, a hydrogen and oxygen supply system and a process monitoring and control system were also added. A feedwater hydrogen injection system provides the method for introducing hydrogen into the reactor coolant. An offgas oxygen injection system was installed to ensure recombination of surplus injected hydrogen to preclude the possibility of a hydrogen explosion in the offgas system. The capability to inject oxygen into the condensate system is also provided to maintain dissolved oxygen concentrations greater than 20 ppb, ensuring that carbon steel corrosion rates are not increased.

Summary of Safety Evaluation: The modifications required to implement the HWC System are not safety-related. Modifications performed by this DCP did not change the function or degrade the operability of any safety-related systems or components. The piping installed by this DCP is located primarily in the Turbine Building and yard areas. The size and location of the newly-installed piping systems preclude any potential damage to safety-related systems and equipment due to falling objects. None of the required piping systems connects directly to safety-related piping systems or equipment. Applicable design, procurement, installation and testing standards at least as stringent as those previously used were adhered to. Inadvertent opening of valves will not cause a loss of boundary because check valves are installed near all plant interface points to prevent reverse flow. The potential for fires or explosions due to uncontrolled release of hydrogen or oxygen was addressed and found to be no greater than previously evaluated. All process control devices are "fail-safe" and loss of power or control signals will result in an automatic hydrogen system shutdown. The potential fire hazard associated with routing of additional electrical cable, isolation of HWC System trip signals from safety-related relays and the potential for damage of safety-related equipment due to seismic events were evaluated and no adverse effects were identified. Poolside and destructive examination of fuel bundle components at another BWR exposed to HWC conditions for 18 months indicate that the effect of increased concentrations of hydrogen on corrosion rate, hydrogen pickup, and reactivity are not significant. In addition, experience with Canadian reactors and PWRs indicate that increased dissolved hydrogen concentrations will not increase the probability or consequence of fuel cladding failure. Radiation levels in the Drywell near the recirculation piping and components will actually decrease due to increased N-16 carry-over. Outside the Drywell, the carry-over of N-16 causes radiation levels to increase primarily in the Steam Tunnel and in some Turbine Building areas. It was determined that equipment exposed to the increased radiation levels over the remainder of the life of the plant will not be degraded beyond acceptable limits. Also, because of the automatic system shutdown capability, accident dose rates will not be affected. HWC pre-implementation test results indicate that dose rates will increase by a factor of two in the vicinity of main steam lines and turbine standard floor areas. Based on surveys conducted to identify high dose rate areas, no additional shielding or redesignation of radiation areas is required. The formation of combustible gas mixtures in the Torus due to lifting relief valves was evaluated. Under these conditions, the hydrogen flow rate to the Torus would increase slightly. However, oxygen blowdown will be decreased, and the inerted containment eliminates the possibility of forming combustible gas mixtures in the Torus. The possibility of reaching combustible gas concentrations in sumps is extremely remote. In addition, the HWC System design provides for automatic hydrogen injection shutdown if high hydrogen concentrations are detected in the vicinity of the only two sumps where combustible gas concentrations are possible. The potential for safety and explosion hazards associated with the new hydrogen piping, hydrogen storage facility and Control Room habitability in the event of a gas leak was reviewed. The new hydrogen piping contains less hydrogen than that previously existing in the Turbine Building for Main Generator stator cooling. In addition, the HWC hydrogen supply is equipped with three independent safety shutdown devices. The new hydrogen piping was determined not to create any Control Room habitability or fire and explosion hazard. The new hydrogen storage facility is located approximately 230 feet from the nearest safety-related structure which exceeds the recommended separation distance of 110 feet. The oxygen storage facility (1,500 gal LOX tank) is located in the yard adjacent to the Turbine Building wall. This location meets the separation criteria contained in NFPA-50 for proximity to combustibles. The nearest combustibles are the main and auxiliary transformers, also located along the Turbine Building wall. In addition to meeting established separation criteria, a concrete wall separates the LOX tank from each transformer. The effect of increased oxygen entering safety-related air intakes was also considered and adequate separation was assured. Technical Specifications require automatic initiation of a reactor scram at three times the normal background radiation level in the main steamlines. The main steamline radiation level increase due to hydrogen injection at the maximum ten standard cubic feet per minute will raise the main steamline radiation level to only two times the normal background radiation level. Therefore, no change to Technical Specifications was required. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1371 <u>Compressed Air System Modification</u>

<u>Description and Basis for Change</u>: This modification provided a more reliable supply of compressed air for instrument and plant service air requirements. The primary change adds three new air compressors and maintains the older air compressors as standby compressors. A new building was constructed approximately 100 ft. from the Turbine Building to house the new air compressors and associated support systems. A common air header was routed underground from the compressor building to the Turbine Building through an existing penetration in the Turbine Building south wall. The new header joins with the existing header upstream of the air receivers. In addition to the new air compressors, a cooling system was installed to support compressor operation. Electrical power to run the new air compressors and cooling system is obtained

from transformers installed for the new Low-Level Radwaste Processing and Storage Facility.

Summary of Safety Evaluation: The compressed air system is not required for and does not affect the operation of safety-related equipment. All containment isolation valves and dampers have Seismic Category I air accumulators in the event of a complete loss of compressed air. Air-operated isolation dampers in the Standby Gas Treatment System and Control Building Ventilation System share a Seismic Category I compressed air supply that will be available in the event of a loss of the normal compressed air supply. All other safety-related valves fail to the fail-safe position upon loss of air. The possibility for a reactor scram on low reactor water level due to loss of air to the feedwater regulating valves is reduced since this modification is an enhancement to the reliability of the compressed air system. The location of the building that houses the new air compressors has no impact on plant safety. Fire detectors were installed in the new building and provide an alarm in the main Control Room. All new piping for the system is designed and installed to Seismic Class II and was not routed near any Seismic Class I piping. Power for the new compressor system was obtained from non-essential sources and segregation of essential and non-essential cables was maintained. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1378 250 VDC Battery Replacement

Description and Basis for Change: Age-related degradation necessitated the replacement of sixteen cells in the 250 VDC Battery System. Replacement of these sixteen cells was accomplished under this DCP. The cells that were replaced are of the lead-antimony type, are obsolete and are no longer manufactured. The replacement cells are of the lead-calcium type. Lead-calcium cells are preferred over lead-antimony because they require less maintenance, have a longer life expectancy, and do not show signs of positive plate corrosion. There are 120 individual cells in the 250 VDC battery. Following this modification, 104 remaining lead-antimony cells and 16 new lead-calcium cells make up the 250 VDC battery system.

Summary of Safety Evaluation: An evaluation was performed to determine the effect of the two different types of cells on the operation of the 250 VDC battery. This combination of cells performs the same functions with the same connected load; meets the voltage and amp-hour capacity requirements of the 250 VDC Battery System; will receive the necessary maintenance and surveillance testing; is seismically installed and rated for Class 1E service. The new cells are of larger capacity and are judged more reliable than the cells that they replaced. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1398 Deparallelling of Plant Process Computer

Description and Basis for Change: This change was in response to a commitment to update the plant process computer (PPC) hardware and software to more modern technology. Under a previous design change, a new PPC was installed in the Data Acquisition Center. The new PPC was run in parallel with the old PPC to ensure proper operation prior to dismantling the old PPC. This modification removed the older, obsolete PPC from service and replaced it with the newer improved PPC. Also, this modification replaced the old Rod Worth Minimizer (RWM) display with the newer RWM display on Reactor Control Panel 1C05 and connected it into the Reactor Manual Control System Circuitry to provide control rod blocks when required. The old RWM function was replaced by the new RWM under a previously-completed design change.

Summary of Safety Evaluation: This modification did not alter the function of any safety system. The objective of the PPC is to provide quick and accurate determination of core thermal performance including proximity to core thermal limits defined in Technical Specifications. The objective of the RWM is to supplement procedural requirements for control rod manipulation during reactor startup and shutdown. The new PPC and RWM display meet these objectives. In addition, the functionality and reliability of the upgraded PPC is improved through use of optical isolation and transformer coupling isolation devices. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1401 HPCI/RCIC Room Cooling

Description and Basis for Change: Temperatures in both the High Pressure Coolant Injection (HPCI) room and the Reactor Core Isolation Cooling (RCIC) room were found to be higher than their design maximum normal-operating temperature of 104°F. Poor air circulation in the HPCI room and high humidity and inadequate ventilation in the RCIC room were also a concern. This DCP installed an additional chiller in the yard area, two new air conditioning units, one in the HPCI room and one in the RCIC room, and sufficient piping to provide supply and return to and from the new air conditioning units. Placement of the new unit in the HPCI room required construction of a new platform. Coolant circulated in the system is a 60% water/40% ethylene glycol mixture. The ventilation air flows provided to the HPCI room and Southeast Corner Room were returned to their original design values, a new exhaust fan was added to the RCIC room and the temperature initiation setpoints for the engineered safeguards heating and ventilating systems existing in the HPCI and RCIC rooms were increased to prevent fans from operating unnecessarily and adding to the room heat load. The temperature initiation setpoints changed by this modification are not associated with the HPCI and RCIC Steam Leak Detection system setpoints controlled by Technical Specifications.

Summary of Safety Evaluation: Ventilation flow to the HPCI room and Southeast Corner Room has not been reduced below the minimum flow requirements stated in the FSAR. The addition of an exhaust fan in the RCIC room to exhaust air to the Torus area ensures that flow from lesser to greater contamination potential, as required by the FSAR, is maintained. The modified temperature setpoint for initiation of the engineered safeguards cooling fans in the HPCI and RCIC rooms does not affect the ability of the coolers to maintain room temperatures below the design value of 104°F during HPCI and RCIC system operation. Existing HPCI/RCIC room penetrations were utilized for routing of new piping. The new chiller unit and associated cooling units are self-contained and do not directly interface with other systems. The new equipment installed in the HPCI and RCIC rooms is seismically mounted for protection of safety-related systems that are present in the rooms. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1403 Feedwater Startup Control Valve

Description and Basis for Change: The Main Feedwater Control Valves are not designed to accurately control flow within the low and narrow range required during reactor startup conditions. At least one reactor scram has been directly attributed to difficulties experienced while using these valves to control flow during reactor startup. In addition, low flow conditions in these valves cause rapid erosion of valve seating surfaces, releasing cobalt impurities into the reactor water, increasing valve leakage and requiring frequent valve repair. This modification provided for the installation of a Feedwater Startup Control Valve that is sized and designed specifically for low flow plant startup conditions. In support of the new Feedwater Startup Control Valve, new piping, two manual isolation valves and one motor-operated isolation valve were added. The modification is configured to allow bypass of feedwater flow around the Main Feedwater Control Valves during reactor startup. To facilitate this modification, two core drills were made in the Turbine Building floor and a single variable pipe support connected to overhead structural steel was required. In addition, a new platform was constructed to allow easy access to the Main and Startup Feedwater Control Valves for maintenance and operation. Manual or automatic position control of the new Startup Feedwater Control Valve is provided in the Control Room from a new manual/auto control station using pre-existing reactor level control signals.

Summary of Safety Evaluation: The new Startup Feedwater Control Valve is capable of accurately controlling feedwater flow over the range required during reactor startup. This capability reduces the probability that reactor water level will exceed acceptable limits resulting in a feed pump trip or a reactor scram during startup; due to reactivity spikes from a cold water injection. The new piping and components are designed to meet the original requirements listed in the FSAR and therefore the probability of occurrence of a piping failure is not increased. The effect of a feedwater flow control failure in the direction of increased flow was reviewed. Since the modification provides for flow in parallel with and at a much lower capacity than the Main Feedwater Control Valves, a failure of the new Startup Feedwater Control Valve, in a direction that increases flow, is bounded by the previously-analyzed condition of failure of the feedwater controller. In addition, should feedwater control valves fail in the full-open position, the flow to the reactor will not be any greater than any occurrence previously analyzed. Failure of the Startup Feedwater Control Valve will not affect the operation of the feedwater check valves that provide the primary containment isolation function associated with the feedwater system. This modification did not involve an unreviewed safety question or Technical Specification change.

DCP 1413 <u>4A Feedwater Heater Replacement</u>

<u>Description and Basis for Change:</u> Eddy current testing performed during a refueling outage revealed a high degree of tube damage to the 4A Low Pressure Feedwater Heater. This DCP replaced the damaged heater with a new heater of improved design. Design improvements incorporated into the new heater include relocating reactor feedwater pump seal water lines to the heater shell, providing impingement plates for protection of the tube bundle and replacing the last twenty feet of reactor feedwater pump seal water piping adjacent to the heater with a material known for its resistance to erosion.

<u>Summary of Safety Evaluation:</u> Applicable accident scenarios addressed in the FSAR include loss of a feedwater heater, loss of feedwater flow and a steam line break outside of containment. Since the new heater is designed to meet the same thermal requirements, provide the same heat transfer capability, and handle a more stringent set of abnormal operating conditions than the heater that it replaced, a loss of feedwater heating is no more probable than for the system that it replaced. The new heater is designed to accommodate the same flow as the old heater and all welds were tested to ensure compliance with the applicable ANSI requirements, therefore the probability of the occurrence of a loss of feedwater flow or a steam line break has not been increased. No new failure modes have been postulated. This modification does not affect any system related to the safe shutdown of the plant or any other safety system. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1416 Drain Off Piping Addition for TCV 6924A and B

Description and Basis for Change: Temperature Control Valves (TCVs) control condensate and bypass flow for the Control Building chillers. Each valve requires a constant bleed-off flow from the upper diaphragm for proper operation. This bleed-off flow has been routed to the storm drain line at the south end of the Reactor Building via a 1.5" rubber hose. This modification installed hard drain piping from the TCVs to the HVAC Hx/Chiller Area sprinkler drain line.

Summary of Safety Evaluation: The piping was seismically supported due to Seismic II over I concerns. The new drain line is sealed to eliminate the possibility of contaminating the storm drains in the event of an airborne release or radioactive spill in the Reactor Building. In the unlikely event that the new drain line breaks, water from the Emergency Service Water System would drain into the second floor of the Reactor Building and eventually into the Reactor Building floor drain sumps. The potential internal flooding rate is within the capacity of the Reactor Building floor drain system therefore no internal flooding hazard exists. The new closed drain line reduces the amount of water processed through the Radwaste System and decreases the possibility of leaking of contaminated fluid reaching the clean storm drain system. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1419 Sewage Treatment Plant

Description and Basis for Change: During the 1985 refueling outage, the increased number of persons at the site caused the capacity of the DAEC sewage treatment facility to be exceeded and the limit stated in our National Pollutant Discharge Elimination System effluent permit was exceeded. As a result, Iowa Electric committed to the Iowa Department of Natural Resources that a new sewage treatment plant would be in operation prior to the 1988 refueling outage. This Design Change Package provided for construction of a new sewage treatment facility, renovation of the old sewage treatment plant and tie-ins to existing plant electrical, water and sanitary systems.

Summary of Safety Evaluation: The sewage treatment plant does not effect the nuclear operation of the DAEC. The use or failure of the sewage treatment plant cannot cause or help mitigate any accident as defined by the FSAR. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1421 PPC Enhancements

Description and Basis for Change: This modification provided for the capability to monitor main condenser drain temperatures on the Plant Process Computer (PPC). To facilitate this capability, data acquisition hardware including fiber optic modems and fiber optic cable was installed connecting main condenser drain line temperature elements to the PPC. In addition, a new recorder was placed in the Control Room back panel area to provide Control Room operators with the ability to monitor Control Rod Drive (CRD) temperatures. CRD temperatures were made available by replacing the older temperature recorder located at a local instrument panel with a Remote Transmit Unit (RTU) which digitally transmits CRD temperature data to the new CRD temperature recorder. The new temperature recorder will be capable of providing output signals to the communications multiplexer in the Dperator's Common Console for future use on the PPC.

Summary of Safety Evaluation: The equipment installed to make condenser drain temperatures available on the PPC and CRD temperature monitoring capability available in the Control Room does not provide any accident initiation or mitigation signals to equipment important-to-safety. Electrical isolation of this equipment from equipment important-to-safety ensures that a malfunction of the PPC or CRD RTU cannot cause spurious activation of equipment important-to-safety. No new penetrations were made and the new CRD Temperature Recorder was mounted on a non-seismic panel. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1424 Control Rod Drive (CRD) Pump Bypass

Description and Basis for Change: During refueling operations, it is desirable to provide some small amount of continuous flow through the Control Rod Drive Mechanisms (CRDMs) to keep them clean and free of debris. In the past, this small amount of flow was provided through the CRD Pumps which allowed for unfiltered particles to build up on the wear rings, causing premature failure of the pumps. The CRD Pump discharge filters prevent these small particles from reaching the CRDMs. This modification provided a means to bypass the CRD pumps during refueling operations and thereby prevent particle buildup. Bypass flow is now provided via a service condensate line fitted with two service connections and another service connection on a pressure instrument line in the discharge header of the CRD pumps. Flow to the CRDMs is provided by connecting a hose between each of these fittings. New piping and appropriate service connections were installed on both the CRD System and Service Condensate System piping to facilitate this modification.

In addition, a 15 second time delay relay was installed in the CRD Pump low suction pressure trip circuitry to prevent spurious CRD

Pump trips due to pressure fluctuations. This modification was recommended by the BWROG Scram Frequency Reduction Program.

Summary of Safety Evaluation: Any postulated failure of the added bypass line piping or CRD instrument line will not adversely affect the ability of the reactor to scram or the condensate system to supply necessary flow. The bypass piping is supported so that it will not affect the operation of other equipment during a seismic event. Approved construction practices were used in the installation of the bypass piping and service connections to this piping. This modification is not relevant in the evaluation of an accident or malfunction of equipment important-to-safety since it will not impair the ability to safely shutdown the plant.

The purpose of the CRD Pump trip on low suction pressure is not a function required to prevent or mitigate any accident or malfunction evaluated in the FSAR. The addition of this time delay does not adversely affect the trip function and minimizes the probability of a spurious CRD Pump trip. The time delayed trip function does not affect any component or system important-to-safety and does not reduce the effectiveness of the system to protect against low CRD Pump suction pressure.

The portion of the CRD System needed to support the reactor scram function does not include the CRD pumps, therefore the scram function is not affected by this modification. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1426 Recirculation Pump Vibration Monitoring

Description and Basis for Change: General Electric and Byron-Jackson recommended increased vibration monitoring of the recirculation pumps to detect abnormalities that indicate the need for maintenance. Scheduled maintenance based on vibration monitoring reduces equipment failures and the high cost associated with equipment downtime. Seven accelerometers and nine proximeters were installed on each recirculation pump to provide signals for analysis.

Summary of Safety Evaluation: This change physically modified the recirculation pumps which are important-to-safety. The design function of each pump was unchanged by the addition of the sensors, sensor mounting brackets and mounting blocks. This design change did not affect the operability of the pumps or the reactor coolant pressure boundary. The seismic capability of the pumps was unaffected by this change. No new failure modes were introduced by this modification. Failure of the vibration monitoring system will not affect the operation of any system required for plant safety. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1428 Cooling Tower Modifications

Description and Basis for Change: During periods of high temperature and high humidity, the cooling towers at the DAEC were incapable of supporting plant operation at full generating capacity. This resulted in annual generation losses of approximately 35,000 MW-hours, most of which occurred during periods of peak demand (between June and September). This modification provided for the addition of two new mechanical draft cells, similar to the existing cells, at the north end of both cooling towers, allowing recovery of a substantial portion of the lost generation capacity. Accompanying the cooling tower modifications, a new deluge system was installed to protect the new cells from fire, four new post indicating valves were added to the fire main, and the deluge valve houses were relocated to make room for the new cells. Each of the new cells is equipped with a 200 hp motor and fan powered from existing nonessential load centers.

Summary of Safety Evaluation: The cooling towers at the DAEC are not safety-related, however they are relied upon to provide a heat sink necessary to maintain main condenser vacuum, thereby preventing a main turbine trip and a subsequent reactor pressure increase. The new cells were constructed to meet the same structural criteria as the existing cells. Besides the potential for indirectly causing a reactor pressure increase, the cooling towers are not involved in any accident scenario described in the UFSAR, nor are they utilized to mitigate the consequences of any evaluated events. The capability of the fire protection system in the vicinity of the cooling towers is unchanged by this modification. No increase in fire protection system capacity is required since only one deluge system is assumed to require operation at any given time. The non-essential load centers used to provide power to the new fan motors have adequate capacity to power the new loads. Therefore, the likelihood of a loss of heat rejection capability due to overloading of these non-essential load centers is not increased. Since main condenser vacuum will be higher, use of the new cells will increase the margin to a turbine trip due to high condenser backpressure. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1430 Replacement of Breakers 1B3401 and 1B4401

Description and Basis for Change: The design of the power supplies for Low-Pressure Coolant Injection (LPCI) System injection valves and the Reactor Recirculation System isolation valves was found to be inadequate if a loss of one division of 125 VDC power is considered in conjunction with a Loss of Offsite Power and concurrent Loss of Coolant Accident (LOOP/LOCA). The AC power supply for these valves consists of Motor Control Centers (MCCs) 1B34A and 1B44A that are tied together in a "swing bus" arrangement. Air circuit breakers (ACBs) 52-3401 and 52-4401 with

125 VDC powered controls are used to feed this bus from Division I and Division II power, respectively. An interlock is provided that prevents the opened ACB from closing on loss of AC power until the closed ACB has opened. In an event of a loss of 125 VDC control power associated with the closed ACB, this ACB would fail to trip open causing a loss of AC power to the swing bus. This scenario would leave only a single division of the Core Spray System available to provide low-pressure water to the core. LPCI would be unavailable as the injection valves would be without power until the opened ACB is closed manually. ACBs 52-3401 and 52-4401 were replaced with new breakers that contain a DC undervoltage trip device. This undervoltage device ensures that that the affected ACB will trip upon loss of 125 VDC control power allowing the swing bus to automatically transfer to the other division of AC power. A time delay is included with the undervoltage device to ensure that adequate time has elapsed following a LOOP to allow the respective Emergency Diesel Generator to start and load. Additional concerns regarding the effect of a swing bus fault resulting in the loss of either MCC 1B34 or 1B44 were considered. The DAEC design provides power to valves associated with the LPCI System from MCCs 1B34A and 1B34B and valves associated with the Core Spray System from MCCs 1B34 and 1B44; therefore the existing ACBs and Motor Controlled Breakers (MCBs) are required to separate the swing bus from 1B34 and 1B44. Adequate protection was provided by improving the breaker coordination between the swing bus maintenance isolation breakers 52-3402 and 52-4402 and load center breakers 52-303 and 52-403. Improved coordination was provided by replacing MCBs 52-3402 and 52-4402 with new breakers containing solid-state overcurrent protection devices that meet the established coordination criteria. MCBs 52-303 and 52-403 are located in series with ACBs 52-3401 and 52-3402, and ACBs 52-4401 and 52-4402, respectively, in the feeders to the swing bus. This DCP also modified the control circuitry associated with ACBs 52-3401 and 52-303 to allow operation if a shutdown from outside the Control Room is required concurrent with a loss of 125 VDC control power.

<u>Summary of Safety Evaluation</u>: The replacement ACBs, MCBs and cables installed by the design change were procured as safety-related and are qualified for operation in their intended environment. Separate evaluations determined that seismic, fire protection, cable ampacity and cable fault current carrying capabilities as a result of these modifications meet applicable requirements. These modifications did not change the function of any system but did restore the plant to its originally intended design. System performance was enhanced by ensuring that a sufficient number of Emergency Core Cooling Systems (ECCS) will be available in the event of a loss of a single division of DC power. The Alternate Shutdown Capability System (ASCS) modifications relied upon in the event of a shutdown outside the Control Room do not change the failure modes, function, or method of operation of any system when transfer switches are in the

"NORMAL" position. The ASCS modifications enhance the ability to carry out the emergency operations required following a Control Room evacuation by providing a reliable source of DC control power to ACBs 52-3401 and 52-303. The modifications performed ensure compliance with 10 CFR 50.46 and 10 CFR 50, Appendix K, by restoring the validity of the assumption that a loss of a single division of DC power will not result in a significant reduction in ECCS capability. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1432 <u>Control Bldg. - Switchgear Rooms Heating, Ventilating and Air Conditioning (HVAC) Modifications</u>

<u>Description and Basis for Change:</u> This modification provided for the installation of additional ductwork in the Essential Switchgear Rooms to improve air flow distribution in the vicinity of the three new static inverters installed under a previous design change. In each of the Essential Switchgear Rooms, a pre-fabricated duct "T" section was installed in the existing cooling air supply just upstream of the room's diffusers. Seismically supported rigid horizontal duct branches were installed. These rigid duct branches are not safety-related, however, they are seismically supported to protect safety-related equipment in the room. The added ductwork materials were fabricated, procured and installed as non-safety-related equipment.

<u>Summary of Safety Evaluation:</u> The installation of the new ductwork does not affect the total cooling air flow to the Essential Switchgear or Battery Rooms. The increased room temperatures associated with the installation of the new inverters is determined to reduce the rated life of the 125 VDC and 250 VDC batteries, however the performance of each battery system is increased due to the higher temperature. Technical Specifications require testing the batteries at regular intervals to monitor for degradation. The new ductwork is seismically supported and routed to ensure that it will not impact the existing ductwork or any safety-related equipment. Installation of the new ductwork does not affect the function of the Essential Switchgear Rooms HVAC Systems. A failure of any portion of the new ductwork would not reduce the switchgear room's airflow. The switchgear room temperature will remain below the qualification temperature of the new inverters. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1436 <u>Instrument Air Dryer</u>

<u>Description and Basis for Change:</u> Two instrument air dryers were installed in the Instrument and Service Air System at the DAEC. The primary air dryer, installed under a previous design change, is a fully automatic heat-less type dryer with a maximum flow capability of 500 scfm. The other air dryer was originally

installed equipment and was designed to operate automatically through an eight hour cycle with a capacity of only 200 scfm. The original air dryer did not have sufficient capacity to meet plant demand when used as a back-up. As a result, the air was not dried properly and a limited amount of desiccant entered the system. This modification replaced the originally installed air dryer with a new dryer identical to the other fully automatic heat-less dryer. In addition, various other improvements were made to the system to improve resultant air quality and reliability.

Summary of Safety Evaluation: Total failure of the compressed air system will not affect the operation of safety-related systems. Plant safety functions are protected by accumulators or independent compressed air supplies. The increased capacity and reliability of the Instrument and Service Air System results in a dryer air supply to air operated components which has reduced the probability of equipment malfunction caused by moisture in the air. No new air system failure modes are postulated with this modification. This modification did not adversely affect the performance of any equipment that initiates a safety action or is required to mitigate an accident. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1441 Replacement of an Intermediate Range Monitor (IRM) Cable and Installation of Spare IRM Cables

Description and Basis for Change: This modification was performed to replace a defective Intermediate Range Monitor (IRM) cable. The conduits and junction boxes through which the defective cable was run contained several other IRM cables. These cable runs were filled such that attempting to pull a new cable may have damaged adjacent cables. New conduit was installed to provide for routing of the new and several spare IRM cables.

Summary of Safety Evaluation: This modification made no functional changes to the IRM System or to any safety-related system. New conduit and cables were installed to applicable specifications. The new IRM cables were procured to meet existing standards. Only environmentally qualified connectors were used to terminate the cables. The new conduit was installed to maintain required separation between components and since the cable was routed only through conduit and junction boxes the Fire Hazards Analysis was not adversely impacted. This modification did not involve an unreviewed safety question or a change to Technical Specifications.

DCP 1452 Emergency Service Water (ESW) Air/Vent Valve Modification

Basis and Description of Change: Each ESW pump discharge line contained an air/vent valve located close to the pump. These air/vent valves were replaced with blind flanges by this design change. The purpose of each air/vent valve was to provide an air discharge path from the pump casing and system piping when the pump started to preclude degradation of heat exchanger performance due to air intrusion. In addition, these valves served as vacuum relief devices for the discharge piping when the ESW pumps were secured, allowing water in the ESW discharge piping to drain back through the pump to the wet pit.

Summary of Safety Evaluation: The ESW System piping is safety-related. The function of the ESW System is to provide cooling water to all emergency equipment except for the RHR heat exchangers. The system consists of two independent trains, each with the capacity to meet the design basis requirements for emergency operation. The seismic evaluation was not adversely affected because replacement of the air/vent valves with the blind flanges reduced the weight and loading on the piping in this area. The air pumped downstream due to removal of the air/vent valve function is small compared to the amount of air normally contained in the system; any air in the system is transported through the system quickly precluding adverse effects on the heat transfer capabilities of the various heat exchangers and ensuring adequate water flow for cooling purposes. The vacuum relief function could be removed since the pressure on the ESW pump seals due to the residual column of water is negligible compared to the operating system pressure. The ESW System is an open loop system, containing high point vents and discharges to the cooling towers and storm sewers. Therefore, the possibility of water hammer in the system was not increased by replacing the air/vent valves with blind flanges. The ESW flowrate to downstream safety-related equipment remains as designed. The reliability and availability of the ESW System has been improved by eliminating the possibility of failure of the air/vent valves. This modification did not involve an unreviewed safety question or Technical Specification change.

During 1989, various procedures as described in the Safety Analysis Report were revised and updated. All changes were reviewed against 10 CFR 50.59 by the DAEC Operations Committee. Two safety evaluations were written to support changes to plant procedures (one Operating Procedure (OP) and one Surveillance Test Procedure (STP)) during 1989. Summaries of these procedure changes and their safety evaluations are provided below. No changes were made that involved unreviewed safety questions or changes to Technical Specifications.

All Special Test Procedures (SpTPs) performed in 1989 were reviewed by the DAEC Operations Committee. No unreviewed safety questions were found to exist. Summaries of these special tests and their safety evaluations are also found below.

TEST/PROCEDURE	TITLE/DESCRIPTION
OP-25	HPCI and RCIC Turbine Overspeed Trip Test

This procedure change reactivated Operations Procedure No. 25 (OP-25) which verifies the operability of the HPCI and RCIC overspeed trip tappet assembly. The vendor recommended that this verification be performed following each surveillance test of the systems until a permanent modification to the trip tappet assembly is developed. OP-25 has Operations personnel physically verify the ability of the tappet assembly to move freely by first closing the respective steam supply valve, then manually tripping the turbine overspeed trip lever and visually verifying free tappet movement. During the time that the tappet mechanism is actuated, the respective system is not available for automatic operation without operator action. For the HPCI System, this operator action may be taken from the Control Room. For the RCIC System, local operator action is required.

Summary of Safety Evaluation: The purpose of this procedure is to detect a system condition that could degrade system availability. Binding of the trip tappet assembly could prevent the turbine overspeed protection feature from performing its intended function and could cause damage to the turbine if an overspeed condition did develop, such that the system would be unavailable for further injection. The short duration of the test and the availability of the operator to promptly reset the trip mechanism in the event that a valid initiation signal is generated during performance of this procedure ensures that the ability of the HPCI and RCIC systems to perform their design functions is maintained. Technical Specifications do not require testing of the HPCI and RCIC overspeed trip function to ensure operability. This surveillance did not involve an unreviewed safety question or a change to Technical Specifications.

STP-42A016 Reactor Water Cleanup System Leak Detection Instrument Functional Test/Calibration

Test switches in the Steam Leak Detection System (SLDS) used during functional testing and calibration surveillance testing were jumpered out of the SLDS circuitry by a temporary modification to preclude the occurrence of spurious isolation signals experienced following a recent refueling outage. As a result of this temporary modification, monthly functional testing of the HPCI and RCIC SLDS, and annual calibration of the Reactor Water Cleanup (RWCU) SLDS requires lifting leads at the associated temperature switches to adequately complete each test. A design change will be implemented during the next refueling outage to permanently modify the system so that lifting of leads will not be necessary to accomplish testing. A safety evaluation was performed to document that, although lifting leads in the SLDS to perform testing is not in agreement with the FSAR, an unreviewed safety question did not exist.

Summary of Safety Evaluation: The SLDS is designed to detect small steam leaks in the vicinity of high temperature/high pressure systems. Lifting leads to perform functional testing and calibration cannot initiate a steam leak requiring actuation of the SLDS. When performing this surveillance, the isolation function provided by the SLDS is bypassed. After relanding the leads, the isolation signal is verified to be cleared and temperature readings are taken to verify that existing conditions are consistent with those at the beginning of the test. The isolation function is then placed back in service. Should the repeated lifting and relanding of leads result in damage to the leads, system isolation could occur, which is the consistent with the compensatory action required by the DAEC Technical Specifications in the event of SLDS inoperability. The RWCU and RCIC Systems are not required to operate during a design basis event. Isolation of HPCI during a small break LOCA does not increase the consequences associated with this event because of the redundancy provide by the Automatic Depressurization System and low pressure Emergency Core Cooling Systems. Performance of this STP did not involve an unreviewed safety question or a change to Technical Specifications.

SpTP No. 156 In-Reactor Stress Corrosion Monitoring

This special test made use of the Stress Corrosion Monitoring (SCM) System and associated in-core test assemblies installed by another design change to 1) provide data to support the determination of the optimum hydrogen injection rate for protection of reactor internals and recirculation piping; 2) provide data to correlate in-pipe recirculation line measurements with off-line measurements taken from the recirculation sample line; and 3) explore the means for lifetime evaluation, verification and control of environmental conditions within the reactor vessel and recirculation piping to reduce and/or arrest Intergranular Stress Corrosion Cracking and Irradiation Assisted Stress Corrosion Cracking. This test was similar in scope to the DAEC Hydrogen Water Chemistry Pre-Implementation Test performed in 1986 (SpTP No. 128).

Summary of Safety Evaluation: The safety issues concerning elevated hydrogen and oxygen injection rates have been previously addressed in the safety evaluation written for SpTP No. 128. Elevated hydrogen and oxygen flowrates were required for performance of this test. The higher flowrate was maintained with hydrogen system pressure within its normal range and an increase in oxygen system pressure from 150 psig to 350 psig. The effect of the increased oxygen system pressure on the integrity of the oxygen system was evaluated and was determined to be acceptable. To facilitate the high hydrogen flowrates required for the test, the excess flow check valves and automatic high hydrogen flow trip were bypassed. Although these isolation functions were bypassed, sufficient capability was maintained to isolate the hydrogen system in the event of a line break. All oxygen system isolation functions remained operable during the test with the exception of the excess flow check valves. The amount of hydrogen stored onsite during the test did not exceed that amount normally maintained onsite. Reactor water sample flows were obtained from the recirculation loop sample line, RWCU sample line and the Post Accident Sample System Loop "B" Jet Pump sample line. Automatic isolation functions associated with the sample flowpaths were unaffected by this test. Increased radiation levels were expected during performance of this test. Data obtained during the Hydrogen Water Chemistry Pre-Implementation Test was used to control and/or restrict access to certain plant areas using existing Health Physics procedures.

A temporary increase in the Main Steam Line High Radiation Scram and Isolation setpoints does not adversely impact any accident or transient evaluated in the FSAR. At power levels greater than 20%, a control rod drop accident will not result in fuel damage. Main Steam Line High Radiation monitor setpoint changes were implemented only at power levels greater than 20% and were consistent with and performed in accordance with Technical Specifications. Performance of this Special Test did not involve an unreviewed safety question or a change to Technical Specifications.

SpTP No. 157 Feedwater Heater Level Set Test

Because of recent problems associated with vibration induced failures of feedwater heater tubes, it was recommended that the liquid level be raised in the feedwater heaters. This action is expected to preclude the occurrence of steam entrainment in the drain cooler zone of the heater, eliminating the root cause of the vibration damage. This Special Test was performed to determine the optimum water level for feedwater heater operation.

Summary of Safety Evaluation: Analysis of a loss of feedwater heater event in the FSAR assumes a reduction in feedwater temperature of 100°F. Operating data indicates that the loss of one heater would result in at most a 50°F reduction in feedwater temperature. Since the level of only one heater was adjusted at any one time during the test, the analysis is bounding. Plant conditions during the test were more conservative than those analyzed in the FSAR for a turbine trip without bypass. Therefore, the FSAR analysis bounds the severity of this event during the test. Feedwater heater levels were not adjusted to levels needed to cause either of the above events by this test. No new failure modes were introduced and no modifications were made to feedwater heater level controls to accomplish the actions required by the test. Performance of this special test did not involve an unreviewed safety question of a change to Technical Specifications.

SpTP No. 159 <u>Effect of Standby Filter Unit Flow on Control Building Pressure</u>

This test was performed to determine the effect of increasing the fresh air supply to the Control Building HVAC System, when it is in isolation mode, on Control Building pressure. The test results were used to determine if Control Building modifications were required to improve positive pressure conditions in the Control Building during isolation. The test required the simultaneous operation of both Standby Filter Units (SFUs) to provide the required increase in fresh air supply.

<u>Summary of Safety Evaluation:</u> The safety issues associated with simultaneous operation of both SFUs were addressed. At no time during the performance of this test was either of the two SFUs made inoperable. If during the test, one SFU was found to be inoperable, the second SFU would remain fully operational and capable of performing its design function. Damage to SFU fans, filters and dampers should not occur because the flowrate through each system will not exceed the normal maximum flowrate. If for any reason SFU flow was too high, an alarm would actuate in the Control Room to alert plant operators and the test would be aborted to prevent equipment damage. Operating both SFUs simultaneously does not constitute a unique situation since initially during the normal isolation mode both units operate automatically until one is manually placed in standby mode. If both SFUs continued to operate simultaneously following a radiation alarm, the resultant increase in doses received by Control Room personnel would remain significantly below the guidelines established by General Design Criterion (GDC) 19 of Appendix A to 10 CFR 50. The ductwork, as designed, is capable of withstanding any increase in pressure resulting from the increased flow conditions. Performance of this special test did not result in an unreviewed safety question or a change to Technical Specifications.

SpTP No. 161 Secondary Containment Integrity Testing with a Smoke Generator

The purpose of this test was to evaluate the secondary containment boundary for leakage using a smoke generator. The Standby Gas Treatment System was isolated during the test to ensure that future operability of this system was not threatened.

Summary of Safety Evaluation: This test was performed during a period of time when secondary containment integrity was not required and therefore any potential effect of the smoke on secondary containment integrity was eliminated. The effect of smoke on equipment and personnel within the secondary containment was evaluated and was determined not to be harmful due to the small concentrations present. Performance of this special test did not involve an unreviewed safety question or a change to Technical Specifications.

SECTION C - EXPERIMENTS

This section has been prepared in accordance with the requirements of 10 CFR Part 50.59(b). No experiments were conducted during calendar year 1989.

SECTION D - SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES

This section contains information concerning relief valve and safety valve failures and challenges for calendar year 1989 in accordance with the requirements of Technical Specifications 6.11.1.e. Note that any instance in which a main steam relief or safety valve was manually cycled open, for surveillance testing or other reasons, is included for your information. There were no safety valve failures or challenges during 1989. There were three relief valve challenges and one relief valve failure during 1989. These events are summarized below:

Date	Event Description
January 24, 1989	Relief valve PSV-4402 was opened and closed during satisfactory completion of a post-maintenance test.
February 2, 1989	Multiple high steamline radiation signals were received during a test of the Hydrogen Water Chemistry (HWC) System at 100% power when the increased hydrogen injection rate required by the test caused an automatic reactor scram and reactor vessel isolation. The Safety Relief Valves (PSV-4400, PSV-4401 and PSV-4407) automatically operated as designed to mitigate the resultant reactor pressure increase. This event was reported to the NRC by Licensee Event Report (LER) 89-003.
March 5, 1989	While the plant was operating at 100% power and calibration of the main steam line high radiation monitoring system was being conducted, one Main Steam Isolation Valve (MSIV) inadvertently closed due to a failed DC solenoid. Closure of this MSIV resulted in high flow (140%) in the remaining main steam lines which caused an automatic reactor vessel isolation followed by a reactor scram. The subsequent pressure transient was controlled by automatic operation of pressure relief valves PSV-4400, PSV-4401, PSV-4402 and PSV-4407 in relief mode followed by automatic operation of PSV-4401 and PSV-4407 in Lo-Lo Set mode, per design. This event was reported to the NRC by LER 89-008.
August 26, 1989	The plant was operating at 100% power during power/load unbalance testing of the main turbine control circuitry when a failed generator current transformer caused the main turbine to trip resulting in a reactor scram and pressure transient. Three safety relief valves (PSV-4400, PSV-4401, and PSV-4407) automatically opened as designed to mitigate the pressure transient. This event was reported to the NRC by LER 89-011.
November 24-29, 1989	During reactor startup from a maintenance outage, PSV-4402 momentarily lifted at a pressure below its normal setpoint. During troubleshooting, PSV-4402 lifted two additional times below its normal setpoint. The reactor was shutdown to repair PSV-4402. The pilot valve was replaced and PSV-4402 was opened and closed during satisfactory completion of post-maintenance testing.

SECTION E - FIRE PLAN CHANGES

The information contained in this section identifies, briefly describes and provides assurance that changes made to the DAEC Fire Plan during the period May 15, 1986 (the date of the most recent NRC inspection to assess and review Fire Protection Program requirements at the DAEC) to December 31, 1989 did not alter our commitment to the NRC guidelines contained in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance." In the future, this section will provide this assurance for changes made to the Fire Plan during the period covered by this report.

Revision No.	Description of Change
19	This revision added a responsibility for the Fire Marshal to review all modifications associated with fire protection systems and/or plans, added a requirement for certification of the staff specialist in fire protection to be certified by the State of Iowa, addressed the XL-3 System alarm functions in the Control Room and included Area Fire Plans for the new Low-Level Radwaste Processing and Storage Facility and Data Aquisition Center. This revision did not change or violate any commitment made pertaining to the Fire Protection Program at the DAEC.
20	This revision incorporated the Fire Suppression Water Status Board in the Control Room, defined the authority and instructions for use of the status board, provided a definition of "Fire Protection Impairment" and implemented guidance to operations personnel and the Fire Marshal concerning planned and unplanned impairments to fire protection systems. This revision did not change or violate any commitment made pertaining to the Fire Protection Program at the DAEC.
21	This revision modified two Area Fire Plans to reflect actual plant equipment configurations. This revision did not change or violate any commitment made pertaining to the Fire Protection Program at the DAEC.
22	This revision incorporated guidance for the disposition of fire drill records, participation of off-site agencies in fire drills and radiation training for fire brigade members. This revision did not change or violate any commitment made pertaining to the Fire Protection Program at the DAEC.
23	This revision divided the Fire Plan into two volumes. Volume I contains administrative controls and Volume II contains fire strategies and Area Fire Plans. It also placed signature blocks for additional managerial approval of the Fire Plan, defined the authorities and responsibilities of the Plant Services Superintendent, Fire Protection Coordinator and Operations Department within the Fire Protection Program, and provided additional administrative information. This revision did not change or violate any commitment made pertaining to the Fire Protection Program at the DAEC.