Iowa Electric Light and Power Company

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NG-88-0424

Dr. Thomas Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
1987 Annual Report of Facility Changes, Tests, Experiments, and Safety and Relief Valve Failures and Challenges
File: A-118e

Dear Dr. Murley:

In accordance with the requirements of Appendix A to Operating License DPR-49, 10 CFR Part 50.59(b), and NUREG-0737 (Item II.K.3.3), please find enclosed the subject report covering the calendar year 1987.

Very truly yours,

William C. Rothert
Manager, Nuclear Division

WCR/DJM/pjv*

cc: D. Mienke
L. Liu
L. Root
R. McGaughy
J. R. Hall (NRC-NRR)
A. Bert Davis (Region III)
NRC Resident Office
Commitment Control 870021

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SECTION A - PLANT DESIGN CHANGES

This section contains brief descriptions of and reasons for plant design changes completed during the calendar year 1987 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 of interest. It is noted that portions of some DCPs listed were partially closed in previous years.

DCP No. 803 — Addition of Chemical Treatment Equipment to the Circulating Water System

Description and Basis for Change: A significant reduction in the heat transfer efficiency of the main condenser and an increase in the corrosion rate of the main condenser and the general service water heat exchanger surfaces were caused by (1) a build up of scale due to high calcium concentrations in the circulating water, (2) a thick mat of biogrowth which could not be adequately controlled by chlorination of the circulating water system, and (3) a build up of silt in areas of low flow velocities. To solve these problems tanks, pumps, valving and controls for the addition of (1) a biocide, (2) a dispersant, and (3) a stabilizer that stops the crystaline growth of calcium carbonate were installed.

Summary of Safety Evaluation: This change was not safety-related. The installed equipment does not interface with any safety-related system. Additionally, reduction of the corrosion rate improved the reliability and heat transfer characteristics of the circulating water system. All piping installed meets ANSI B31.1 power plant piping requirements. The chemical storage tanks are mounted on a concrete pad with a spill protection dike and 2" drain valve. No unreviewed safety questions existed.

DCP No. 1033 — Radwaste Solidification Process Piping

Description and Basis for Change: All radwaste shipped by the DAEC for burial must meet the burial site requirements outlined in NRC Information Notice 80-24. Beginning July 1, 1981, the burial sites required that all spent resin which exceed activity levels of 1 μCi/gram must be solidified to insure no leakage after burial. Under this DCP, piping was installed to deliver resin slurry from the process piping in the centrifuge room to the solidification equipment. This new piping also provides the capability to sample the resin mixture.

Summary of Safety Evaluation: This DCP was not safety-related and does not affect the safe shutdown of the reactor. The new piping meets the NRC criteria concerning radwaste piping, Regulatory Guide 1.143, except the methods used for seismic qualification were consistent with original plant design and not upgraded to Regulatory Guide 1.143 standards. Because this system ties into ASME Section III, Class 3 and ANSI B31.7, Class 3 piping, the interface points have been modified to conform to the original construction code. No unreviewed safety questions existed.

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DCP No. 1065 Drywell Radiation Monitors Finstrip Heaters and Improved Detectors

Description and Basis for Change: The drywell radiation monitor detectors had a history of premature failures due to moisture intrusion. The air sampling system radiation monitors provide an early indication of leaks in the drywell or torus and serve as an alternate to the drywell leak detection system. The solution to this problem was to seal the detectors to prevent moisture damage, and to heat and insulate the detector assemblies to prevent condensate from forming around the detectors by the addition of circuit-protected finstrip heaters and fire retardant insulation.

Summary of Safety Evaluation: This change was not safety-related. The detectors do not interface with any safety-related components, nor do they initiate any automatic protective functions. The air sampling piping for the drywell radiation monitors is isolated at the drywell penetration during a Group III Isolation. The modification did not alter the operation or design function of the system. Additionally, it increased the availability and reliability of the system such that the margin of safety defined by Technical Specification (T.S.) 3.6.C.2. is improved. No unreviewed safety questions existed.

DCP No. 1086 Replace Radiation Element/Radiation Indication Monitor (RE/RIM) 6101 A/B & 7606 A/B

Description and Basis for Change: DCP 1086 corrected a number of problems associated with this equipment, including seismic qualification inadequacies, poor equipment reliability and availability due to adverse weather conditions, and difficulties in performing the required calibration of the equipment. The solutions to these problems were, as follows, (1) new seismically-qualified instruments for RE/RIM 6101 A/B and 7606 A/B were procured and installed, (2) RE/RIM 6101 A/B were relocated from an outdoor location to an indoor location downstream of the standby filter unit (SFU) pre-heat coils, which is in the normal air flow path, (3) temperature inputs to the SFU lockout relay were removed, but the existing low temperature alarm was retained, and (4) an access hole (for the insertion of a calibration source) was provided to better facilitate calibration.

Summary of Safety Evaluation: This modification was safety-related. It did not change the original design intent or function of RE/RIM 6101 A/B and 7606 A/B. The new instruments were seismically qualified in accordance with IEEE-344. Relocating the detectors downstream of the preheat coils eliminated their exposure to rain and winter weather. Upgrading the instruments provides more reliable operation and more accurate readings because the upgraded equipment was better qualified to operate during and after a seismic event. Habitability of the control room is not affected by the removal of the temperature inputs that initiate auto startup of the SFUs because the alarm function was retained and the operator will have sufficient time to manually start the SFUs upon receipt of an alarm. No unreviewed safety questions existed.

DCP No. 1131 Radwaste Cuno Filters Piping Installation

Description and Basis for Change: The piping to/from the Cuno filters from the radwaste surge pump had the potential for leakage of radioactive fluid which could have resulted in potential equipment and personnel contamination. Additionally, that piping returned filtered liquid to the radwaste surge tank. This design change replaced the piping with permanent steel pipe and added cross-tie piping to return filtered water to the radwaste collector tank, the floor drain collector tank, or the chemical waste tank for added flexibility and efficiency in the processing of liquids.

Summary of Safety Evaluation: This change reduced personnel radiation exposure and reduced the likelihood of leaks or spillage of contaminated fluid, while adding operational flexibility to the radwaste system and produced no conditions not previously analyzed in the FSAR. No changes in plant technical specifications were required. No unreviewed safety questions existed.

DCP No. 1149 Modification to Diesel Generator Air Start System

Description and Basis for Change: Various problems to the diesel generator air start system existed before DCP 1149. These problems were: (1) leakage of system air via the relief valves due to relief valve chatter. The relief valve chatter was caused by too small of differential pressure between the diesel air start system operating pressure and the relief valve setpoint; (2) pressure switch failure due skid vibrations and air pulsations from the reciprocating compressors; (3) unregulated battery chargers for the diesel starting air compressor batteries caused premature failure of the batteries; (4) air leakage from the diesel air start regulator located on the crank case of the emergency diesel generators (due to repeated disconnection and reconnection of the copper air supply tubing to the oil booster tanks during testing of the system); and (5) particulate build up on a solenoid valve downstream of the air filter. Additionally, small amounts of corrosion were detected in the assembly, indicating the presence of moisture. DCP 1149 made the following modifications: (1) the maximum operating pressure of the diesel generator starting air system was lowered from 240 psig to 225 psig, new relief valves were installed and the low pressure alarm switch setpoints (PS-3232A&B) and (PS-3233A&B) were reset from 200 and 225 psig to 175 and 200 psig respectively; (2) pressure switches PS-3234A&B were relocated away from their respective compressor skids; (3) the unregulated battery chargers were replaced with regulated battery chargers; (4) instrument valves were installed upstream of the oil booster tanks for emergency diesel generators 1G-21 and 1G-31 to preclude air leakage; and (5) new air filters, housings and flanges were installed. Additionally, moisture drain valves and flexible tubing were installed in the air lines of the diesel generator air start system. The flexible tubing (which was installed from the air compressors to the rigid piping of the diesel air start system) prevents damage to the rigid piping due to vibration of the air compressor skids.

DCP No. 1149 (Continued)

Summary of Safety Evaluation: This modification was safety-related. The reliability of the diesel generator air start system was increased by lowering the maximum operating pressure and installing superior relief valves and regulated battery chargers. The original design intent of the diesel generator air start system was not affected by DCP 1149. The capacity of the system still meets the original design requirements described in section 8.3.1.1.2 of the Updated FSAR. Installation of instrument valves upstream of the oil booster tanks eliminated the necessity to disassemble and reassemble the piping during system testing. No unreviewed safety questions existed.

DCP No. 1163 Personnel Monitoring Stations

Description and Basis for Change: Radiation protection procedures require that all personnel exiting contaminated areas survey themselves for radioactive contamination. The purpose of DCR 1163 was to install six additional personnel monitoring stations in the turbine and reactor buildings. Permanent walls and roofs were constructed to provide shielding to reduce the area background radiation levels. The walls were constructed of high density grout-filled concrete blocks with the roof of reinforced concrete on steel deck. Access to each station is through an open doorway.

Summary of Safety Evaluation: The added station structures do not house any safety-related items and are not essential to plant operation. The design of each station structure ensures that a possible, but unlikely collapse of the stations as a result of a seismic event would not impact the integrity of the surrounding building structure or any safety-related system components. Each structure will retain its integrity under load during Design Basis Earthquake (DBE) conditions. The enclosures had a negligible affect on the structural integrity of the existing buildings. No unreviewed safety questions existed.

DCP No. 1180 Turbine Lube Oil Conditioner 1T-39

Description and Basis for Change: The existing Turbine Lube Oil Purification system did not clean the oil adequately and was a frequent maintenance item. To improve the quality of the lube oil, a larger more efficient lube oil conditioner was installed. The existing system was retained as a backup. The new conditioner can be operated alone or in parallel with the existing system.

Summary of Safety Evaluation: This modification was not safety-related. High quality turbine lube oil is essential for turbine and generator protection. The new system was installed in accordance with ANSI B.31.1, "Power Piping Code." The probability of a turbine trip due to the failure of the Turbine Lube Oil System is reduced, because the system is more reliable and has improved conditioning capabilities. Additionally, curbing was added to the lube oil tank area to accommodate the additional fluid capacity of the new system in the event of fluid leakage or spillage. The additional capacity was analyzed for its impact on the Fire Hazards Analysis and was found acceptable. No unreviewed safety questions existed.

DCP No. 1185 Acid Pump Replacement & pH Control Modifications

Description and Basis for Change: The sulfuric acid feed system is intended to control the acidity of the Circulating Water System. The system that existed before DCP 1185 was a gravity feed arrangement that performed erratically and had permitted water acidity to become high enough to cause corrosion damage to equipment in the Circulating Water System. DCP 1185 installed acid feed pumps and improved controls to allow more precise control of the introduction of acid into the circulating water to maintain optimum pH. Additionally, DCP 1185 upgraded the sulfuric acid feed system piping to stainless steel. Stainless steel is better suited for handling acid because of its excellent resistance to corrosion. DCP 1185 also added a catch basin for the sulfuric acid tank and a safety shower/eyewash station.

Summary of Safety Evaluation: DCP 1185 was not safety-related and does not interface with any safety-related equipment. The installation of corrosion resistant piping and better pH control increases the reliability of the Acid Feed System. The installation of a catch basin for the sulfuric acid tank and a safety shower/eyewash station improves personnel safety. The remote location of the tank and pumps (outside the pumphouse) assures that safety-related equipment will not be affected in the event of a piping failure. The Circulating Water System is protected from low pH by interlocks that shut off the pumps and isolate the Acid Feed System should the Circulating Water System pH fall below 6. High pH alarms are provided although high pH is not damaging to equipment. No unreviewed safety questions existed.

DCP No. 1237 Replacement of Pressure Indicating Switch (PIS) 3887

Description and Basis for Change: Pressure indicating switches (PIS 3887 and 3901) monitored line pressure at the Collector Filter (1F-207) inlet in the Liquid Radwaste Treatment System. Before DCP 1237, jumpers were installed in control room panel 1C96 to bypass both pressure inlet switches and indications because of previous concerns with the pressure switches either becoming inoperable or losing calibration. Design Engineering concluded that the pressure switches were not needed for the operation of this filter. Therefore, the pressure switches and indicators were eliminated.

Summary of Safety Evaluation: The removal of these switches did not adversely affect the operation of the liquid radwaste system. No other control functions for the radwaste system were adversely affected by this modification. This change did not alter system operation as described or implicit in the Final Safety Analysis Report. No unreviewed safety questions existed.

DCP No. 1238 Replace Recirculation System Flow Transmitters

Description and Basis for Change: Flow transmitters FT-4631 A-D and FT-4632 A-D for the reactor recirculation system had a history of zero shift drifting that caused spurious half-scrams and could have resulted in incorrect biasing of the scram setpoint. The existing transmitters were no longer manufactured and were replaced with suitable substitutes.

DCP No. 1238 (continued)

Summary of Safety Evaluation: This modification was safety-related. Evaluation of the DAEC's temperature profile for the specific locations and applications in the specific loops, showed that the new transmitters were suitable replacements for the existing transmitters. The transmitters were purchased as Class 1E units and were qualified to IEEE-323 and IEEE-3[illegible] standards and no new failure modes were introduced. Seismic calculations were performed to attest to the seismic mounting of the new transmitters. The instrument racks are Seismic Category I and have been seismically qualified (see FSAR Section 3.10.1.1). No unreviewed safety questions existed.

DCP No. 1245

Access Control Modifications

Description and Basis for Change: DCP 1245 authorized the modification of specific areas of the ground floor of the Administration Building. The modifications primarily involved the Access Control function of the building and resulted in more efficient traffic flow in and out of the reactor building. New office and storage space was created, allowing more effective utilization of Health Physics personnel. The capability of processing contaminated personnel and equipment was also improved, as was emergency equipment access and storage. These improvements were recommended in NRC Appraisal Audit Report Number 50-331/80-21. Additionally, this modification resulted in a significant improvement in housekeeping and appearance of the administration building. The scope of DCP 1245 included modifications to the existing architecture and plumbing as well as Heating, Ventilating and Air Conditioning (HVAC) and electrical systems of the administration building. The modification was essentially a reutilization of existing space and no changes were made in the overall structural integrity of the building.

Summary of Safety Evaluation: This modification was not safety-related and did not affect any safety-related equipment or functions of the plant. The installation, use or failure of any modification made by DCP 1245 did not increase the probability or the consequences of an FSAR-evaluated accident, nor was the possibility of a different type of accident created. The modifications did not affect the existing structural integrity of the building and existing margins of safety remain unaffected. The work was divided into two phases to maintain the access control function of the building at all times. No unreviewed safety questions existed.

DCP No. 1262

Neutron Monitoring System and Rod Block Module Power Supplies and Shared LPRM Count Circuits

Description and Basis for Change: DCP 1262 formalized Emergency DCP 1262 (EDCP 1262), which implemented the recommendations of General Electric Service Information Letters (SILs)-312, -313, and -290. This modification updated the Average Power Range Monitor (APRM) electronic modules to conform with revised General Electric service/maintenance procedures. These changes precluded the operational occurrences described in the SILs.

DCP No. 1262 (continued)

Summary of Safety Evaluation: This modifiction was safety-related. It was performed while the plant was shutdown. No changes to the DAEC Technical Specifications were required. This change involved implementing manufacturer's recommended engineering service changes to improve the performance of the existing equipment and eliminate conditions described in the SILs, which could lead to unnecessary scrams. The design intent or function of the APRMs was not affected by this DCP. The existing equipment remained fully qualified to perform in any adverse environment to which it may be subjected. This modification enhanced the reliability of the Neutron Monitoring System because the probability of a single APRM power supply failure causing an unnecessary scram is reduced. No unreviewed safety questions existed.

DCP No. 1274 Modify Radiation Monitors (RM) 1997/4767

Description and Basis for Change: The modification was required to meet radiological Technical Specifications implemented at the DAEC in accordance with Appendix I of 10 CFR Part 50. Radiation Monitor RM-1997 was modified in order to monitor the required lower limit of detection (LLD). This modification relocated RM-1997, whose function is to monitor the effluents from the Residual Heat Removal and Emergency Service Water Systems (RHRSW/ESW), from its previous location in the High Pressure Coolant Injection (HPCI) room. The background radiation count in the HPCI room varies with reactor power level and with HPCI system operation. This would have made it necessary to recalibrate the detector's trip setpoints every time reactor power was changed. Also, whenever the HPCI system is initiated, the increased background radiation levels in the HPCI room would have caused RM-1997 to alarm needlessly. The radiation transmitter was also relocated to a position closer to the new location of RM-1997 to ensure that electrical "noise" would not hamper the transmission of the signal from the radiation element to the radiation transmitter. Further evaluation demonstrated that RM-4767 did not require modification.

Summary of Safety Evaluation: DCP 1274 was not safety-related. The original design function of RM-1997 was not changed. Since the alarm function associated with RM-1997 provides no automatic functions and only serves to inform the operator that there is a high level of activity in the RHRSW/ESW water, the additional time (approximately 15 seconds) it takes for the operator to receive the signal from RM-1997 (due to the monitors being relocated approximately 90 feet downstream from its original location) is not a limiting time concern. The required piping and electrical hardware were constructed, installed and tested to the same specifications and quality level as the original system. The ability of RM-1997 to monitor release of radioactive material in liquid effluents was not reduced. No unreviewed safety questions existed.

DCP No. 1295 Diesel Generator Jacket Coolant Heater Relief Valve Removal

Description and Basis for Change: The seat in the relief valve on the jacket coolant heater of diesel generator (1G-31) was leaking. Repairs had been unsuccessful and replacement valves were not available because the vendor had discontinued that particular

DCP No. 1295 (continued)

model. Emergency DCP (EDCP) 1295 removed the relief valve on the jacket coolant heater for 1G-31. DCP 1295 formalized EDCP 1295 and performed the identical modification on the other diesel (1G-21) by removing its jacket coolant heater relief valve. Additionally, flexible tubing was installed from the 3/4" overflow line on the expansion tank to the floor drain to preclude water from the expansion tank spilling onto electrical equipment.

Summary of Safety Evaluation: This modification was safety-related. This modification was performed when the plant was in the cold shutdown condition. The safety-related functions of the relief valves were to act as a system pressure boundary and to provide overpressure protection for the jacket coolant heaters, (ASME Section VIII coded vessels). Failure to perform either function could result in a loss of jacket coolant and eventual overheating of the diesel. However, upon evaluation, there was no possibility of heater vessel overpressurization since the outlet of the heater is connected (via unvalved, unobstructed 1-1/2" and 3" hard piping) to the jacket coolant expansion tank. The expansion tank has a vent to atmosphere and a 3/4" overflow line, both unvalved. Therefore, the relief valves were unnecessary for overpressure protection and could fail open and therefore increase the probability of loss of jacket coolant. No unreviewed safety questions existed.

DCP No. 1313 Offgas Condenser (1E-223) Lifting Lug

Description and Basis for Change: The tubing in the offgas condenser (1E-223) is required to be inspected periodically. In order to make the inspection it is necessary to remove the condenser head. A lifting ring is located on the top of the condenser head to facilitate its removal. However, there was no lifting lug located directly above the condenser head. The solution to the problem was to fabricate and install the needed lifting lug.

Summary of Safety Evaluation: The installation of the lifting lug for the offgas condenser head was not safety-related. The lifting lug will not interact with any safety-related items during a seismic event. Since the offgas condenser head weighs less than 1000 pounds, it is not classified as a heavy load per NUREG-0612. This head is removed only during cold shutdown, therefore the consequences of a load drop will not affect offsite doses via the offgas system flow path. No unreviewed safety questions existed.

DCP No. 1319 Separate Reactor Pressure Scram Switches

Description and Basis for Change: There are four pressure switches in the high pressure reactor scram logic circuitry of the Reactor Protection System. These switches are arranged into two logic channels, each including two pressure switches (i.e., "one out of two twice" logic). Before this modification, the four pressure switches were physically located in close proximity to each other. Since normal reactor operating pressure is marginally close to the scram setpoint, inadvertent "bumping" of the pressure switches or the instrument racks on which they were mounted could

DCP No. 1319 (continued)

have caused an inadvertent scram. This concern is described in GE Service Information Letter (SIL) 143. DCP 1319 physically relocated two of the four previously mentioned pressure switches to preclude inadvertant scrams.

Additionally, each of the four pressure switches mentioned above had an associated pressure switch that actuated an alarm when reactor pressure was within 10 psig of the scram setpoint. This alarm function is not safety-related. The alarm circuits for the alarm pressure switches were wired such that they were not divisionally separated. This DCP eliminated two of the alarm pressure switches and rewired the two remaining switches so that one alarm pressure switch was associated with each scram logic channel and they were divisionally separated.

All six of the pressure switches were replaced with switches procured from a different vendor because the original equipment manufacturer no longer supplied a qualified replacement.

Summary of Safety Evaluation: This DCP was safety-related. The replacement pressure switches met the same specifications as the original switches and were qualified as Class 1E equipment. Physically separating the pressure switches associated with the scram logic will significantly reduce the possibility of inadvertent scrams due to physically "bumping" the pressure switches. Rewiring the alarm circuits will improve divisional separation of redundant circuits and thereby increase the reliability of the alarm function (although it is not a safety function). The replacement pressure switches will introduce no failure modes other than those previously considered and are at least as reliable as the original equipment. The high pressure reactor scram setpoints were not changed as a result of these modifications. No unreviewed safety questions existed.

DCP No. 1320

Hydrogen Water Chemistry

Description and Basis for Change: Oxygen suppression in reactor primary coolant systems through a Hydrogen Water Chemistry (HWC) program has proven to be effective in mitigating inter-granular stess corrosion cracking (IGSCC) of piping. DCP 1320 provided the piping modifications required to perform the HWC mini-test. These modifications consisted of installation of injection and sample taps. Each tap was installed with isolation valves which were accessible during plant operation. DCP 1320 did not address the actual test or any consequences of running the test. (See Special Test Procedure 128 in the 1986 Annual Report of Facility Changes).

Summary of Safety Evaluation: This modification was not safety-related. The taps were not installed within the primary coolant pressure boundaries of any safety-related systems. The failure of the affected system piping due to the installation of the taps is not more probable than previously evaluated, because of a strict adherence to codes for design, procurement, installation and testing which are as stringent or more stringent than those previously evaluated and the consequences were

DCP No. 1320 (continued) unchanged as the taps were smaller in diameter than the piping to which they were attached. No unreviewed safety questions existed.

DCP No. 1323 Seismic Pipe Support Mods

Description and Basis for Change: The deadweight, thermal and earthquake loads at seismic supports were reanalyzed in accordance to NRC Bulletin 79-14. Part of the reanalysis involved determining whether each support was adequate under the new loads. These support calculations showed that all supports were adequate. However, at some of the supports, the new loads created component reactions, which exceeded the support's design rating. Additionally, an evaluation of Inservice Inspection (ISI) reports concluded that the scram discharge volume supports were insufficient for the calculated seismic loads. This DCP modified the affected supports to restore the recommended factor of safety.

Summary of Safety Evaluation: This modification was safety-related. The plant was in the cold shutdown condition during this modification. This modification did not change the design intent or function of these supports. This modification utilized the improved accuracy of the calculated loads to restore the pipe supports to their recommended factor of safety. No changes to the DAEC Technical Specifications were required. This modification ensures that the structural integrity of the scram discharge volume pressure boundary will be maintained during and after a seismic event. The loads being transferred to the reactor building and drywell concrete were evaluated in accordance with the design of Seismic Category I structures and found to be acceptable. No unreviewed safety questions existed.

DCP No. 1325 Testable Check Valve Modifications

Description and Basis for Change: The disk for CV-2002 did not seat properly with the valve actuator installed and it was recommended the actuator be permanently disconnected for both Residual Heat Removal (RHR) testable check valves (CV-1906, CV-2002). These testable check valves had also experienced numerous problems with their magnetic disk position indicating switches. Further engineering evaluation concluded that the testable check valve actuators in all Emergency Core Cooling Systems (ECCSs) and their position indicators could be removed. This DCP removed the actuators and position indication for testable check valves CV-1906, 2002, 2118, 2138, 2313 and 2513.

Summary of Safety Evaluation: This modification was safety-related. Each of these testable check valves has a free floating disk. The actuators and position indicators were not required in order for the valves to perform their safety functions. Furthermore, the actuators and position indicators were not a part of the primary containment isolation boundary. This modification did not affect the ability of any associated systems to provide their intended safety functions. No unreviewed safety questions existed.

DCP No. 1328 Standby Gas Treatment System (SGTS) Heater Indicating Lights

Description and Basis for Change: Before DCP 1328, if the maintenance service switch to the SGTS heater was placed in the off position, the SGTS heater (on/off) indicating lights in control room panels 1C170 and 1C171 could indicate that the heater was on, when actually it was off. This was because the indicating lights were powered by heater control circuit rather than providing actual heater status. To eliminate this problem, DCP 1328 added auxiliary contacts to the control circuits for the SGTS electric heaters. This modification resulted in a improved system by providing positive indication of the SGTS heater condition (on/off). The old system simply indicated that the system was calling for the heaters to be on without feedback regarding actual status.

Summary of Safety Evaluation: This modification was safety-related. It did not alter the function or operation of any safe shutdown system. The seismic integrity of control room panels 1C170 and 1C171 was not affected. No unreviewed safety questions existed.

DCP No. 1330 Power Feeds for Data Aquisition Center (DAC)

Description and Basis for Change: Two independent sources of power were needed to provide a noninterruptable power supply to the equipment in the DAC and Technical Support Center (TSC). In order to meet this need, two "primary" sources and one alternate power source were installed, thus providing three power feeds to the DAC and the TSC. The three load centers supplying power are load center 1B6, the Guard Facility (Security Building) transformer, and the radwaste load center.

Summary of Safety Evaluation: DCP 1330 furnished power for the DAC and TSC. The power feeds for the DAC do not supply any safety systems. The power feeds do not adversely affect any buses which supply safety systems. Equipment powered by these power feeds is not required for plant operation; nor is it required in order to make decisions that will affect plant operations. The reliability of the Emergency Response Organization equipment in the TSC is increased, because the power feeds provide an alternate source of power. The routing of cables did not impact the safety function of any safety system. Obtaining power from load center 1B6, the Guard Facility transformer, and the radwaste load center did not adversely impact any other plant safety functions. Since these load centers and transformer are not electrically connected to any safety systems, no unreviewed safety questions existed.

DCP No. 1331B Modifications To The High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems

Description and Basis for Change: Due to problems with severe cavitation during RCIC system operability testing, globe valve MO-2517 the RCIC system pump discharge test bypass valve, was replaced with a special drag valve. The replacement valve has a

DCP No. 1331B (continued)

thicker stainless steel body and special internals designed to take high pressure drops without cavitation. Additionally, changes made by this DCP involved adding position indication for valves MO-2515 and CV-2315 in the control room, repowering the control circuit for CV-2315, providing a back-up source of air for operation of CV-2315 in the event that its normal air supply is unavailable, and replacing restricting orifice FO-2519 with a nonflow restricting spectacle flange, as the newly installed valve no longer required a restricting orifice. The control circuit for CV-2315 is being powered from a different source because it previously received its power from an AC lighting panel, which is not powered by an essential bus.

Summary of Safety Evaluation: This design change was safety related. Changing the RCIC system pump discharge test bypass valve (MO-2515) from a globe valve to a drag valve does not adversely affect the system quality or any safety-related functions because the new valves, piping, indicators, and flanges were designed, procured, installed and tested to the original or later construction codes. The replacement valve and operator meets the same operability requirements as the original valve. Seismic concerns were addressed and were found satisfactory. No unreviewed safety questions existed.

DCP No. 1332

Diesel Oil Pumps Flowmeters

Description and Basis For Change: The ASME Code, Section XI, Paragraph IWP-3400, requires inservice testing of pumps 1P-44A and 1P-44B every 3 months. Testing includes the measuring of pump discharge flowrate. The flowrate instrument accuracy is required to be within ±2% of full scale and the full scale range must be three times the reference value or less. With the configuration that existed before this DCP, the flowrate could not be measured within the required accuracy. The flowrate was calculated by timing the filling of the day tank and estimating the quantity based on level indicator readings. To improve the accuracy and reliability of the flowrate measurement, DCP 1332 installed a direct-reading inline flowmeter (rotameter) on the discharge piping of each pump near the diesel fuel oil storage tank 1T-35. Each flowmeter was installed in an existing nonsafety-related bypass line which ran from a pressure instrument tap line back to tank 1T-35. The 3/4 inch isolation globe valve at the inlet end of this bypass line was replaced with a 1-1/2 inch gate valve to improve flow characteristics. An isolation valve was also added at the outlet end of the bypass line where it returns to tank 1T-35.

Summary of Safety Evaluation: This modification was safety-related. The operation of equipment important to safety remains unchanged. The bypass lines up to the isolation valves are required to maintain integrity during a seismic event. The new piping arrangement was reviewed using the DAEC small bore pipe design criteria. It was judged that the short span of piping and inline components were adequately supported by the connections at the tank cover, the existing 1-1/2 inch pipe support, and an added support at the rotameter. The pressure and flow conditions in the

DCP No. 1332 (continued) transfer pump discharge lines are unchanged because existing tees were utilized where the bypass lines connect to the discharge lines, and no new pressure drop was created. No unreviewed safety questions existed.

DCP No. 1333 Replacement of Temperature Indicator (TI-1006)

Description and Basis for Change: This modification increased the range of TI-1006 (located on a main steam line) from 475-575°F to 0-600°F. A wider temperature range on TI-1006 was necessary to allow operations personnel to use TI-1006 during main steam line warming procedures which restrict main steam line warming to 100°F/hr. Also, new transmitters and a selector switch were installed to accommodate the wider range.

Summary of Safety Evaluation: This modification was not safety-related. It provides control room operators with additional temperature information during warmup of the main steam lines. The function and operation of TI-1006 remain unchanged, except the range was increased to 0-600°F. The installation of the new temperature transmitters, selector switch, and temperature indicator did not alter the function or operability of existing safe shutdown systems. The TI was procured and installed to original design specifications. No unreviewed safety questions existed.

DCP No. 1334 Feedwater (FW) Valves Position Indicator

Description and Basis for Change: This modification replaced the position indicators for the feedwater regulating valves. The valve controllers did not provide positive indication of the actual valve position, only an electrical indication of where the valve was supposed to be. The solution was to install valve positioner-transmitter assemblies on the existing feedwater regulating valves, General Electric Co. Model 180 position indicators and two new power supplies for the new instrument loops on control room panel 1C-06. This provides accurate, positive indication of the actual valve position.

Summary of Safety Evaluation: This modification was not safety-related. It provides more accurate valve position indication and does not adversely impact either the feedwater control system or the regulating valves themselves. The new equipment was procured and installed to the same standards as the existing feedwater equipment. No unreviewed safety questions existed.

DCP No. 1338 Rod Worth Minimizer (RWM)

Description and Basis for Change: DCP 1338 made the following changes: (1) the replacement of the Rod Worth Minimizer (RWM); (2) the replacement of the control room operators' console; (3) the installation of conduit and cables from the plant process computer equipment to the Data Acquisition Center computer room,

DCP No. 1338 (continued)

(the terminations of these cables will be made by DCP 1339); and (4) the installation of circuits from the Traversing Incore Probe (TIP) chassis to equipment to be installed by DCP 1339 (Plant Process Computer).

Summary of Safety Evaluation: The rod worth minimizer is not a safety system. The modifications implemented by DCP 1338 did not affect the design intent or function of the RWM system or add any new interfaces with safety systems. The overall reliability of the RWM system was increased by the installation of the new, microprocessor-based RWM. No unreviewed safety questions existed.

DCP No. 1341

Drywell/Steam Tunnel Heating, Ventilation and Air Conditioning (HVAC) Modifications

Description and Basis for Change: This was an IELP initiated change intended to reduce the ambient temperatures in the drywell and steam tunnel. The high temperatures inside the drywell were due to both higher-than-anticipated heat loads and air stratification. Hot spots in the steam tunnel were due to inadequate air flow distribution. DCP 1341 included the installation of two cooling units in the upper drywell, new ductwork in the steam tunnel, and the replacement of existing drywell metal reflective insulation with NUKON-fiberglass blanket-type insulation.

Summary of Safety Evaluation: These modifications were not safety-related. They were performed when the plant was in the cold shutdown condition. All modifications meet or exceed the design criteria used in the original plant design or more stringent criteria. Seismic analysis was performed on all affected structures, including piping, ductwork, piping supports, and ductwork supports and was found satisfactory. DCP 1341 involved the installation of non-safety-related components in the drywell and steam tunnel areas. These modifications do not alter the function of any safety-related system within these areas. DAEC surveillance requirements remain unchanged. The new NUKON insulation was reviewed against NUREG 0897, Revision 1, Regulatory Guide 1.36, Generic Letter 85-22, Regulatory Guide 1.82 and was found to conform to their specific criteria. No unreviewed safety questions existed.

DCP No. 1342

Switchyard Breaker Control, Indication and Metering

Description and Basis for Change: DCP 1342 provided sources of electrical power for new buildings constructed at the site (Low Level Radwaste Processing and Storage Facility, Air Compressor Building and Data Aquisition Center) by (1) the relocation of the indicating lights for switchyard breakers "R", "S" and "T" to provide space for the installation of three, new breaker control switches with indicating lights in control room panel 1C08, (2) the installation of three GE model SBM breaker control switches and six indicating lights with associated wiring at the panel 1C08

DCP No. 1342 (continued)

in the control room to control new breakers "LQ", "GB" and "GC" installed in the switchyard, (3) revision of the mimic display on Panel 1C08, (4) the installation of watt and VAR meters for the 161/36KV switchyard transformer and the necessary wiring from the new transformer meters to the Plant Process Computer (PPC), (5) rewiring of the station power totalizer (the metering device that measures total site electrical power consumption) in panel 1C31, and (6) the installation of necessary cables between the DAEC control room, PPC, and switchyard facilities.

Summary of Safety Evaluation: This modification was not safety-related. It did not affect normal plant operation or operation of systems related to plant safety. Equipment mountings were analyzed to ensure that their integrity will be maintained during a design basis earthquake (DBE). All cables were procured as class 1E and installed to ensure electrical and physical isolation from existing safety-related equipment was maintained. All devices were installed in compliance with the requirements set forth by the Detailed Control Room Design Review (DCRDR) to preclude human engineering deficiencies. System reliability was not adversely affected. DAEC Technical Specifications and surveillance requirements were not affected. No unreviewed safety questions existed.

DCP No. 1343 ECCS Pump Sequencing On The Essential Buses

Description and Basis for Change: During plant operation the normal electrical line-up has the essential (vital) buses powered from the startup transformer and the non-essential (non-vital) buses powered from the auxiliary transformer. If the auxiliary transformer is not available it is permissible to power both the essential and non-essential buses with the startup transformer only. This condition has been analyzed (FSAR Section 8.2) and found to be acceptable. To reduce load on the startup transformer when the auxiliary transformer was unavailable, the essential buses were powered from the standby transformer with the non-essential buses powered from the startup transformer. Under the original pump start design, the Core Spray (CS) and Residual Heat Removal (RHR) pumps, on receipt of a Loss of Coolant Accident (LOCA) signal, will start simultaneously if the output voltage of either the standby or startup transformer drops below 65% of nominal voltage and there are no degraded bus voltages, (92.5% of nominal bus voltage), regardless of the power source energizing the essential buses. A Loss-Of-Offsite-Power (LOOP) signal is provided to the CS and RHR pump start circuitry which causes the pumps to sequence onto their respective diesel generators at five second intervals. Sequencing is required to prevent simultaneous high pump starting currents from possibly stalling the diesel generator engines rendering them inoperable. The LOOP signal is sent only if the output voltage of both the standby and startup transformers drop to 65% or less of nominal voltage or if a degraded bus voltage condition occurs. Therefore, if the essential loads are powered by the standby transformer and a simultaneous LOCA and standby transformer failure occur, the

DCP No. 1343 (continued)

output voltage of the startup transformer would prevent the LOOP detection logic from sending a LOOP signal to the pump start circuitry which, in turn, would cause all CS and RHR pumps to load to their respective diesel generators simultaneously. This design deficiency was reported in LER 85-36.

This design change was made to permanently correct the problem described above by modifying the automatic CS and RHR pump-start circuitry so that the pumps sequence on at all times following a LOCA signal regardless of the status of offsite power.

Summary of Safety Evaluation: This modification was safety-related. It eliminated a possible malfunction (simultaneous pump loading to the diesel generators during a partial LOOP) which had not been previously analyzed the FSAR. This modification remained within the plant's design bases and is enveloped by the LOOP-LOCA analysis in the DAEC FSAR. No unreviewed safety questions existed.

DCP No. 1347 Modification of Vent Filter 1V-TF-6

Description and Basis for Change: The door permitting access to the vent filter for the condensate demineralizer backwash receiving tank had a tendency to become detached during backwash operation. It was determined that the tank vent duct and filter were undersized for their intended function, resulting in a sharp increase in pressure across the filter when the air inside the tank was forced out by the backwashing action. Additionally, the tank transfer pump handswitch was a spring return selector switch and had to be held in place by an operator during the entire pump operating period. The solution to the problem was to replace the existing 6-inch tank vent, 6-inch ductwork, 4-square-foot filter, and 10-inch backdraft damper with an 18-inch tank vent, 18-inch ductwork, 12-square-foot filter unit, and an 18-inch backdraft damper. Also, the spring-return selector switch was replaced with a maintained-contact selector switch.

Summary of Safety Evaluation: This modification was not safety-related. DCP 1347 involved replacement of nonsafety-related components in the condensate demineralizer backwash system. Installation of a new tank vent, vent filter, ductwork, damper, and handswitch did not alter the function or method of operation of any safety system. This change assures compliance with the original design basis for the aforementioned equipment and improves the process conditions by reducing the pressure in the system. The sizes of the new ductwork and filter were determined by calculating airflow rates and pressures during the backwash operation. The integrity of system was not affected by these changes. No unreviewed safety questions existed.

DCP No. 1350 Reactor Building Elevator

Description and Basis for Change: The reactor building elevator was modified to provide fast and reliable transportation for injured personnel. Specifically, a key switch and signal light

DCP No. 1350 (continued)

was installed on all five floors to permit the elevator to be called to the designated floor, cancelling all other calls and bypassing non-emergency floor calls. When the elevator arrives, it remains with the doors open for a set period of time to permit the elevator to be placed on emergency service. If it is not placed on emergency service, the doors will close and the elevator automatically returns to normal operation. The signal lights are illuminated while the elevator is responding to a priority call and extinguished when the elevator has been placed on emergency service or returned to normal service.

Summary of Safety Evaluation: This modification was not safety-related, nor did it impact any safety-related equipment. No structural modifications were made to the elevator. No unreviewed safety questions existed.

DCP No. 1352 Nitrogen Purge Panel Location

Description and Basis for Change: Nitrogen temperature must be monitored to detect incomplete nitrogen vaporization that could induce thermal shock to nitrogen purge piping systems and result in the formation of cracks in the primary containment, as discussed in NRC Information Notice 85-99. Before DCP 1352, detection of incomplete nitrogen vaporization was accomplished by relying on full-time operator monitoring of nitrogen temperature during purge system operation. DCP 1352 improved the operability of the primary containment nitrogen purge system by (1) providing automatic system isolation in the event of low nitrogen temperature downstream of the nitrogen vaporizer, (2) relocating the controls for the system to the area where the control valves are located in the offgas recombiner fan room and (3) adding a local manual controller. Implementation of these modifications resolved the concerns of Information Notice 85-99 by preventing cold nitrogen from coming in contact with the containment pressure boundary.

Summary of Safety Evaluation: DCP 1352 was safety-related. The design intent and function of the primary containment nitrogen purge system remains unchanged by DCP 1352. Operability of the primary containment nitrogen purge system was improved by adding automatic isolation and relocating system controls. The possiblity of damage to equipment was reduced because the modification allowed an operator to more readily recognize and correct an unacceptable situation. No changes to the DAEC Technical Specifications were required. No unreviewed safety questions existed.

DCP No. 1353 Anticipated Transients Without Scram (ATWS) Modifications

Description and Basis for Change: DCP 1353 brought the DAEC into compliance with 10 CFR Part 50.62 (The Anticipated Transient Without Scram (ATWS) Rule) by installing the following ATWS modifications: (1) the Recirculation Pump Trip (RPT) logic (that existed before this DCP) was replaced with a logic which initiates both RPT and Alternate Rod Injection (ARI) on either low-low reactor vessel level or high reactor system pressure. The new

DCP No. 1353 (continued)

RPT/ARI logic consists of two separately-powered trip systems, with each trip system receiving inputs from two existing reactor pressure vessel (RPV) level switches. If two-out-of-two logic system trip of either parameter occurs in either trip system, both recirculation pumps will trip and energize one ARI solenoid valve per division, which will vent the scram valve pilot air header, causing the control rods to be inserted, (2) the scram valve pilot air header was modified to install additional solenoid valves to vent the header on an ARI initiation, (3) the standby liquid control (SLC) pump control circuitry was modified to add dual-pump operation capability from the control room. The keylocked control switch HS-2613 (that existed before this DCP) was replaced with a two-position keylock switch that will start both pumps, and (4) instrumentation to indicate SLC flow both locally and in the control room was added.

Summary of Safety Evaluation: While this modification was not required to be safety-related, many of the changes were made safety-related due to their interfaces with existing safety-related equipment. Changes to the DAEC FSAR and Technical Specifications were not required but were considered advantageous to improve clarity after the systems were added and/or modified. The revised DAEC Technical Specifications and surveillance requirements maintain the original SLC system design margins. The frequency of inspection and the consequences of failure of inspection or test of components remain unchanged. The method of operation of equipment important to safety either remained unchanged or the modifications enhance system performance and/or reliability. This design change enhanced the effectiveness of the SLC and RPT systems. These systems serve as a backup to the safety-related Reactor Protection System (RPS). Additionally, this change added the ARI system, which is independent and diverse of the reactor protection system. The ARI modification increased the reliability of the reactor trip function, by adding an alternate means of rod insertion. The ARI/RPT logic test switches are seismically qualified with the exception of the keylocked operator. The basic switch assembly is seismically qualified. The ARI solenoid valves were procured as safety-related and are seismically qualified. The increase in SLC system flowrate due to two-pump operation increased the safety margin by bringing the reactor subcritical in less time than during one pump operation. The SLC system was evaluated for two-pump operation and was found not to be adversely affected by either pump syncronization, system relief valve chattering or excessive pump vibration concerns. The flow indication added to the SLC system does not affect plant safety, because the flow sensor does not penetrate the system piping and the flow instruments are non-safety-related devices performing an indication function only. Seismic evaluation determined that the addition of the flow instrumentation has no adverse affect on the SLC system and that the seismic integrity of the SLC piping is maintained. All equipment installed (by DCP 1353) is environmentaly qualified for its intended service environment and will withstand temperature, pressure, humidity, and radiation levels associated with design basis accidents in its installed locations. No unreviewed safety questions existed.

DCP No. 1355 Modification of Remote Shutdown Equipment

Description and Basis for Change: The modifications performed by DCP 1355 in conjunction with the modifications previously performed by DCPs 1203 (Addition of Alternate Shutdown Capability), 1204 (Addition of Automatic Fire Suppression Equipment), 1205 (Addition of Fire Detection Systems), 1206 (Fire Barriers for Electrical Equipment), and 1207 (Fire Barriers) are required to bring the DAEC into compliance with 10 CFR Part 50, Appendix R Section III.G and the exemptions thereto which have been granted by the NRC. As discussed in the NRC Safety Evaluation Report (SER) of January 6, 1983, the alternate shutdown capability system (ASCS) installed by DCP 1203 meets the requirements set forth in 10 CFR Part 50, Appendix R Sections III.G.3 and III.L. The addition of alternate fusing by DCP 1355 is required by NRC Information Notice 85-09. DCP 1355 added backup fuses to all control circuits for equipment required for shutdown outside of the control room. The addition of backup fuses ensures that the reactor can be brought to cold shutdown from the ASCS without replacement of control fuses which may be destroyed by a fire in the main control room panels. DCP 1355 also added switches to isolate and transfer control to ASCS of three, loop A Residual Heat Removal (RHR) valves and the degraded voltage detection circuit on bus 1A4 to prevent spurious actuation due to a control room fire. Additionally, DCP 1355 modified several ASCS circuits to improve operability and maintainability.

Summary of Safety Evaluation: These modifications were performed when the plant was in the cold shutdown condition. This change was safety-related. This design change adds redundant fuses for safety-related components required for safe shutdown from outside of the control room and modifies the control circuitry for individual devices required for alternate shutdown. The modified design resolves concerns identified in Information Notices 85-09 and 85-20. The design change reduces the number of operator actions that will be required to accomplish a safe shutdown outside of the control room, improving the safety of the alternate shutdown process by reducing the chance of operator error. The ASCS modifications are qualified to the same standard as that of the existing design, so the probability of random failure after installation of the modifications will be no higher than that of the existing design (upon upgrade per DCP 1390). The design of the alternate shutdown capability system modifications did not in any way alter the function or method of operation of any safe shutdown system. The modifications enhanced the ASCS by reducing operator actions involved in fuse replacement, providing assurance that possible spurious events are avoided, and simplifying the operator steps required by Emergency Operating Procedures. Safety-related panels and equipment added by DCP 1355 are qualified to operate in their post-accident environment. The parameters considered in the environmental qualification are temperature, humidity, pressure, and radiation. The ASCS panels were reviewed for human factors considerations. The Remote Shutdown Fuse Panels and all interconnecting cabling added by DCP 1355 has been routed in accordance with the requirements of

DCP No. 1355 (continued)

Appendix R to 10 CFR Part 50 as well as existing DAEC separation criteria. Therefore, no additional conduits and/or cables require fire protection as a result of DCP 1355. No unreviewed safety questions existed.

DCP No. 1357

HPCI/RCIC Steam Trap Modification

Description and Basis for Change: The main steam line drain traps for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) steam supply lines were relocated to improve system operation. Before this DCP, the traps were mounted vertically with an upward direction of flow; which was contrary to manufacturer's recommendations. Additionally, these traps were located at points well above the lowest portions of their respective drain lines, thus making complete condensate removal impossible. Also, the inlet and outlet piping to the drain trap bypass valve on the RCIC steam supply line drain pot was replaced with 1/2" schedule-160 piping that conforms with plant design specifications. The piping that existed before this DCP was 1/2" weld-overlayed schedule 80 pipe. Additionally, a 1/2" globe valve was installed in the bypass line to replace original 1" gate valve.

Summary of Safety Evaluation: This modification was safety-related. The plant was in the cold shutdown condition during construction. This modification did not change the design intent or function of the drain pots for either the HPCI or RCIC systems. The condensate removal ability of the system was improved by the relocation of the drain traps. Seismic analysis of the changes to the HPCI and RCIC system showed there was no reduction in the ability of the HPCI or RCIC drain piping to withstand the effects of a design basis earthquake. The integrity of the system was not affected by these changes. No unreviewed safety questions existed.

DCP No. 1360

Elimination of Spurious Actuation of HPCI/RCIC System Leak Detection System

Description and Basis for Change: Before Emergency DCP 1360 (EDCP 1360), the High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) steam leak detection theromcouple trip units could have spiked whenever their power supplies were energized or when actuating the "read" switch in order to take data during surveillance testing. This could have introduced a false isolation signal. EDCP 1360 installed four new time delay relays into the isolation circuits of the HPCI/RCIC steam leak detection thermocouple trip units. The HPCI steam leak detection time delay relays were seismically qualified but the documentation to support seismic qualification of the RCIC steam leak detection time delay relays was unavailable. Therefore, the relays in the RCIC system were installed as non-safety related until qualified relays could be procured (see LER 86-013). The purpose of DCP 1360 was to formalize EDCP 1360 and change time delay relays in the RCIC steam leak detection system from non-seismically qualified relays to seismically qualified relays.

DCP No. 1360 (continued)

Summary of Safety Evaluation: This modification was safety-related. No changes to the DAEC Technical Specifications were required. This modification eliminates inadvertent system isolations and unnecessary safety system challenges due to spurious steam leak detection system (SLDS) signals. The one second time delay introduced into HPCI/RCIC system SLDS will not cause the small breaks to become more limiting than the larger breaks for either loss of coolant inventory or radiological consequences and would not prevent the system from responding to a real event within the necessary time. The original design intent or function of SLDS was not changed. No unreviewed safety questions existed.

DCP No. 1361 Recombiner Vault Access Hatch

Description and Basis for Change: The installation of an access hatch into the offgas system recombiner vault "B" was an IELP-intitiated change. Previously, access was obtained through a ventilation duct which did not permit easy access or removal of an injured person. Access through the roof hatch plugs is only possible during an outage, because their removal would violate secondary containment. The access hatch was installed in the wall between the Heating and Ventilating (H & V) room and recombiner vault B. Additionally, a platform in the H & V room was replaced with a larger one to accommodate personnel entering the access hatch.

Summary of Safety Evaluation: This modification was not safety-related. This modification was performed when the plant was in the cold shutdown condition. These changes did not alter the function or method of operation of any safety system. The new access hatch was placed high in the shielding wall of recombiner vault B to reduce the possibility of radiation streaming. Structural analysis of the affected walls determined that the new access openings would not adversely impact their integrity. The access door is locked to control access and the key is controlled by health physics personnel. The affected walls are not fire barriers. The new access point is through a nonseismic portion of the wall and no seismic category II/I concerns existed. No unreviewed safety questions existed.

DCP No. 1362 Firemain to DAEC Training Center

Description and Basis for Change: In order to provide fire protection for the DAEC Training Center, it was necessary to obtain water from the existing DAEC fire loop header. The fire main leading to the DAEC Training Center was connected to the DAEC fire loop header with appropriate isolation valves.

Summary of Safety Evaluation: This modification was not safety-related. The fire main installed under this DCP does not provide protection to safety-related systems. Any accident or malfunction of this installation does not in any way affect safety-related equipment or impair the ability of the plant to be shutdown safely. The fire main leading to the Training Center can be isolated from the DAEC fire loop header should any breaks or leaks occur. The installed fire pumps have adequate capacity to meet the requirements for a worst case scenario. No unreviewed safety questions existed.

DCP No. 1363 Modifications to Area Radiation Monitor System

Description and Basis for Change: Area radiation monitors (ARMs) 9155, 9156, and 9157 were modified to increase their range, eliminating excess maintenance and facilitating accurate calibration. This was accomplished by replacing the existing detectors of range 10^2-10^6 mR/hr with detectors of range 1-10^6 mR/hr. A detector of this type is currently in use in the offgas system carbon vault in ARM 4138, and it had a satisfactory maintenance history. The power supply of refuel floor ARMs 9163 and 9178 was exchanged with the power supply of control rod drive (CRD) module ARMs 9168 and 9169 so all four of the refuel floor ARMs are no longer on the same power supply in panel 1C11.

Summary of Safety Evaluation: This modification was not safety-related. Normal plant operation or the operation of safety systems were not affected by this change. The modifications made by DCP 1363 only increased the range of three ARMs and exchanged the existing power supplies of four others. These changes did not alter the design or function of these monitors or create any interfaces with safety-related systems or components. No changes to the DAEC Technical Specifications and surveillance requirements were required. Additionally, system reliability was increased by both the new, more easily calibrated range of the detectors and the new power supply arrangement. No unreviewed safety questions existed.

DCP No. 1364 Modify Control Logic of Motor Operated Valves (MOVs) 1936 and 1937

Description and Basis for Change: Before this DCP, if shutdown cooling isolation at 135 psig occurred while the Residual Heat Removal (RHR) system was being flushed to radwaste through RHR drain valves (MOV-1936 and MOV-1937), the reactor vessel would have been isolated by the closure of containment isolation valves (MOV-1908 and MOV-1909) and the remaining RHR system water inventory could possibly have been drained to the radwaste system because the drainage rate of flow was beyond the capacity of the keep fill system. If the RHR system was not refilled prior to reopening of the containment isolation valves, the RHR piping would have been filled with reactor water. This could have resulted in low water level scram and/or water hammer to RHR system piping and valves. This design change used spare contacts of the 135 psig interlock relay (K50) in the closure logic of MOV-1937 in conjunction with an additional two-position handswitch (HS1937A) in Panel 1C03 in the control room to prevent the aforementioned situation. The change causes closure of MOV-1937 if the 135 psig signal occurs during the RHR system flush mode. The handswitch allows opening of MOV-1937 (even if the 135 psig isolation signal is present) for drainage to radwaste in torus cooling mode during normal power operation.

Summary of Safety Evaluation: This modification was safety-related. It was performed when the plant was in the cold shutdown condition. The probability of an accident was decreased, because the valve will close automatically when a 135 psig isolation signal is received instead of having to be closed manually by an operator. This change adds an additional

DCP No. 1363 (continued) isolation signal for MOV-1937. All other existing permissive signals to the control valve remain unchanged. The elimination of low water level scram signals and/or water hammer increases the margin of reactor safety in the RHR shutdown cooling mode. No unreviewed safety questions existed.

DCP No. 1368 Drywell Stabilizer Connections

Description and Basis Changes: During the last refueling outage, several Lubrite plate washers used for the bolted flange connection of the drywell stabilizers were found broken or missing in different locations. These stabilizers span between the biological shield and the primary containment in the drywell. The horizontal stabilizer truss is designed to transfer seismic loads or seismic-plus-jet-impingement loads into the Reactor Building. These loads act on the reactor, biological shield and drywell. These washers were made of Hi-tin bronze (ASTM B22-C91300) a brittle, low-strength material. The washers were made with several 3/16" holes that hold graphite material for lubrication. The 3/16" diameter holes effectively reduced the cross sectional area of the washer by approximately 60% and, therefore, created stress concentration areas which caused fatigue crack initiation and growth failure. Lubrite is not the most suitable material for washers in the flanged sliding connections. In order to eliminate the problem of washer failure, the Lubrite plate washers were replaced with hardened steel washers of larger diameter and thickness, made of ASTM F-436 material. Additionally the bolts and nuts were replaced with bolts and nuts of the same material as the new washers, and the bolt-to-nut connection was changed to a double nut arrangement rather than a tack weld.

Summary of Safety Evaluation: This modification was safety-related. It was performed when the plant was in the cold shutdown condition. The load-carrying flange plates, fasteners, and stabilizer struts are considered safety-related. A nuclear-grade lubricant was applied to the washer to allow free vertical slip of the flange. The possibility of fatigue damage is reduced with the use of a solid, high-strength plate washer. This modification did not change the design intent or function of the drywell strut assembly but served to improve the existing design to ensure the plate washers can maintain their design function without damage. No unreviewed safety questions existed.

DCP No. 1370 Low Pressure Turbine "B" Rotor Replacement

Description and Basis for Change: Examination by General Electric (GE) of the "B" low pressure turbine rotor revealed stress corrosion cracks in keyways between the shaft and shrunk-fit turbine wheels. Therefore, the potential existed for the wheel material to fracture upon reaching a critical crack depth, leading to the occurrence of turbine missiles. The solution was to replace the rotor with a monoblock-type rotor where the wheels are integral to the shaft. This design eliminates the high stress area of the keyways (the source of stress corrosion cracking), thereby reducing the probability of turbine rotor failure.

DCP No. 1370 (continued)

Summary of Safety Evaluation: This modification was not safety-related. The plant was in cold shutdown condition during this modification. This modification did not change the design intent or function of "B" low pressure turbine . Nor did it increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The monoblock has been designed by G.E. to be functionally a "like-for-like" replacement for the shrunk-fit wheel rotor. Elimination of keyway-induced stress corrosion cracking with the monoblock design reduces the probability of occurrence of turbine missles and increases the margin of safety. No unreviewed safety questions existed.

DCP No. 1373 Steam Jet Air Ejector (SJAE) Condensate Pumps Power Supply Modifications

Description and Basis for Changes: The original control transformer that supplies control power for the operation of the SJAE condensate pumps failed in April 1986, which resulted in the loss of both condensate return pumps and a rapid decrease in condenser vacuum. Both pumps were rendered inoperable because they shared a common power supply. The solution was to install separate power supplies, starters and transformers for each pump. The modification relocated this equipment outside of the high radiation area to provide easier access for repairs and to reduce radiation exposure. Also, a more reliable indication of water level in the SJAE condensate return tank (1T136) was needed for operators in the event of possible system failure or inadequate system capacity. The solution was to install a high/high level alarm in 1T-136 to augment the existing high level alarm.

Summary of Safety Evaluation: The plant was in the cold shutdown condition when this modification was performed. This modification was not safety-related and it did not affect any safety-related equipment. This modification did not change the design intent or function of the SJAE Condensate Return System. This change increased the reliability of the SJAE Condensate Return System by providing independent control of the condensate return pumps. No unreviewed safety questions existed.

DCP No. 1376 Torus Vacuum Breaker T-Seal Modification

Description and Basis for Change: This change, initiated by NRC Generic Letter 84-09, permits monitoring potential oxygen inleakage from the T-Ring seal system for the isolation valves in the torus-to-reactor-building vacuum breaker lines. A calculation was performed to determine the maximum-expected air leakage rate through the T-ring seals. The results indicated that leakage through the T-ring seals is proportional to the differential pressure drop across the upstream components in the instrument air line. Therefore, inleakage through the T-ring seals can be monitored by measuring the differential pressure across components

DCP No. 1376 (continued)

in the air supply line to the T-ring seals. DCP 1376 installed differential pressure indicators in the air lines of the downstream isolation valves (control valves CV-4304 and CV-4305) for torus vacuum breakers V-43-169 and V-43-168. A connection was provided in the air line for the T-ring seals to allow temporary connection of a nitrogen supply to the seal if air leakage through the seal exceeds the calculated limit. Isolation valves were provided for the capped connections to allow connection of the nitrogen supply without bleeding down the air system. Accumulators 1T-105A and 1T-105B were connected to the air supply lines downstream of check valves V-43-116 and V-43-117. This configuration prevents the accumulators from acting as backup air supplies to solenoid valves SV-4304 and SV-4305 in the event of long-term loss of air, as originally intended. Therefore, check valves V-43-116 and V-43-117 were removed to restore the air system to a configuration consistent with the original design intent.

Summary of Safety Evaluation: This modification was safety-related. It was performed when the plant was in the cold shutdown condition. This change restored the reactor building vacuum breaker system accumulators to their originally intended design function. The probability of oxygen inleakage into the primary containment was reduced by the addition of differential pressure indicators and their daily surveillance requirement. The differential pressure indicators were seismically qualified by the manufacturer and they were installed to seismic category I requirements to maintain the integrity of the air line during a seismic event. The addition of the differential pressure indicators did not change the method of operation or safety function of the T-ring seals. No unreviewed safety questions existed.

DCP No. 1377

IRM, SRM and PRM Modifications

Description and Basis for Change: Before DCP 1377, the Intermediate Range Monitors (IRMs), Source Range Monitors (SRMs), and Process Radiation Monitors (PRMs) did not provide indication of system failure upon loss of negative voltage, as required by Section 7.6.1.5.7 of the FSAR. DCP 1377 was performed in response to General Electric Service Information Letter 445 to provide the required indications. No new equipment was installed. Termination points were moved on the IRMs and the SRMs to insure that the system logic functions as required by the FSAR. Existing trip unit relays were used for the PRMs to provide the required trips. This modification was performed on all six IRM channels, all four SRM channels and nine PRMs.

Summary of Safety Evaluation: This modification was safety-related. It was performed when the plant was in the cold shutdown condition. Post installation modification testing was performed to ensure that the wiring changes did not change any

DCP No. 1377 (continued)

other system logic or functions and that all instruments perform as designed. The total response time of the Reactor Protection System (RPS) was verified to be less than the maximum allowable response time of 50 milliseconds (DAEC Technical Specifications, Section 3.1) upon completion of the modification. No unreviewed safety questions existed.

DCP No. 1378

250 VDC Battery Replacement

Description and Basis for Change: The 250 VDC battery system is required to support the High Pressure Coolant Injection (HPCI) system electrical functions, provide electrical power to five primary containment isolation motor-operated valves, provide electrical power to selected instrumentation through an uninterruptible AC power supply and provide emergency electrical power to selected main turbine and recirculation MG set lubrication oil pumps to prevent bearing damage. The HPCI and primary containment isolation functions are safety-related functions. The main turbine and recirculation MG set bearing lubrication oil pumps do not perform safety functions and do not interface with any safety systems. The uninterruptible AC power supply provides power to accident monitoring instrumentation but does not interface with engineered safety features. A new battery was installed by this DCP, because age-related degradation of the 250 VDC Battery had shortened its useful life. Annunciators and instruments associated with the 250 VDC system were evaluated and modified as required to comply with human factors guidelines and current NRC guidelines. Additionally, space within the 250 VDC battery room was used to install a 48 VDC battery for other systems and these cells may be used as replacements in the 250 VDC and 125 VDC battery systems if needed.

Summary of Safety Evaluation: This change was safety-related. It was performed when the plant was in the cold shutdown condition. This modification replaced the degraded 250 VDC battery with a functional equivalent. An analysis was performed to verify the time history load curve for the 250 VDC system. The replacement battery rating provides a substantial margin above the system performance requirements. The replacement battery meets standards IEEE 450-'80, 535-'79, 484-'81 and 485-'80 for capacity and aging requirements, which exceed the 1971 standards used to design the old battery system. The replacement battery and rack were constructed to meet IEEE 323-'74 and 344-'75 standards for seismic and environmental qualification. The design intent or function of the 250 VDC system was not affected by this DCP. Additionally, replacement of the 250 VDC battery did not adversely affect the existing battery chargers and the battery room ventilation systems' operability requirements. No unreviewed safety questions existed.

DCP No. 1379 Reactor Specimen Re-Installation

Description and Basis for Change: Reactor specimen samples provide metallurgical information on the reactor vessel material for engineering evaluation. The purpose of this evaluation is to ensure the reactor vessel material has adequate metallurgical properties to continue to perform its design function after continued exposure to reactor operating conditions. The reactor vessel specimen samples are required to be removed per the removal schedule described in the FSAR. General Electric (GE) provided reconstituted specimen samples to install in the reactor vessel prior to the beginning of cycle 9 operation. GE provided the specimen holder, installation tool, and personnel to install the specimens. GE fabricated the reconstituted specimens from samples that had been removed during the cycle 7/8 refueling outage. Therefore, the irradiation effects of operating cycle 8 are not included in these reconstituted reactor specimen samples. The reconstituted specimen set contains Charpy V-notch specimens, tensile specimens, and flux wire specimens. Since ASME has removed the requirement, no weld heat-affected zone metal tensile specimens were included.

Summary of Safety Evaluation: This modification was safety- related. This DCP was performed when the plant was in the cold shutdown condition. The reactor vessel or reactor vessel internals were not altered as a result of this modification. This modification did not change the design intent or function of the reactor specimens; nor did it change the design intent or function of any other safety-related system. No unreviewed safety questions existed.

DCP No. 1380 Shroud Head Bolt Replacement

Description and Basis for Change: General Electric (GE) Service Information Letter (SIL) 433 dated February 7, 1986, notified owners of Shroud Head Bolt (SHB) cracking concerns. The crack formation in the SHB was determined by GE to be caused by crevice-initiated Intergranular Stress Corrosion Cracking (IGSCC) in the area of the Ni-Cr-Fe Alloy 600 portion of the bolt formed where the 304 stainless steel sleeve is welded to the bolt shaft. Accordingly, Iowa Electric contracted GE to perform ultrasonic (UT) examinations on all 26 shroud head bolts during the 1987 refuel outage. Thirteen SHBs were found to be unacceptable. Eleven were replaced with a new type of SHB which does not have a crevice area and two SHBs were replaced with new SHBs of the original design. The thirteen remaining SHBs and the two new SHBs of the original design installed by DCP 1380 were analyzed and were found to be acceptable for continued use until they can be replaced.

Summary of Safety Evaluation: The SHBs are non-safety-related. This modification was performed when the plant was in the cold shutdown condition. This modification did not alter the design intent or function of the SHB or the shroud head. The new design of SHB improves the reliability of the SHBs by eliminating the SHB cracking concerns. This modification did not alter any other equipment and did not alter the reactor vessel internals.

DCP No. 1380 (continued)

Additionally, GE performed a safety evaluation for the continuing operation of the DAEC with the remaining SHBs of the original design and determined that such operation does not pose a safety concern. The new SHB design is expected to perform its design function for the remaining life of the DAEC. No unreviewed safety questions existed.

DCP No. 1381 Motor Operate Valve (MO) 1908 Motor Replacement

Description and Basis for Change: The purpose of DCP 1381 was to formalize Emergency Design Change Package 1381 (EDCP 1381). EDCP 1381 was initiated to replace the failed motor for MO-1908. MO-1908 is the Residual Heat Removal (RHR) system shutdown cooling mode inboard pump suction isolation valve. The root cause of the motor failure was over-torquing of the valve motor caused by normal torque requirements to unseat the valve being larger than expected in the original design. The required torque exceeded the motor's locked rotor torque design of 40 ft-lbs. The failed 40 ft-lbs motor was replaced with a new 60 ft-lbs motor. Ampere readings taken after installation of the new motor indicate that the new motor's locked rotor torque capacity exceeds the unseating torque requirements of the valve.

Summary of Safety Evaluation: This modification was safety-related. The motor replacement occurred while the plant was in the shutdown condition. This modification did not change the design intent or function of the RHR system. The original seismic calculations were verified using the weight of the new motor and were found to remain valid. No modifications to the electrical circuitry were required. The replacement motor's torque is sufficient to properly operate the valve without overstressing the disc and stem assembly. No unreviewed safety questions existed.

DCP No. 1382A Residual Heat Removal Service Water (RHRSW) Water Chemistry

Description and Basis for Change: This modification made a change to control biological fouling and corrosion on the tube side of the Residual Heat Removal (RHR) heat exchangers. This modification involved installation of sodium hypochlorite injection piping in the RHRSW/Emergency Service Water (ESW) wet pits. This piping is connected to the existing General Service Water (GSW) system to provide dilution flow. Sodium hypochlorite injection piping and valves were installed between the discharge of the circulation water chemistry injection pumps and the stilling basin.

Summary of Safety Evaluation: This modification was not safety-related. This modification did not affect the method of operation or safety function of the RHRSW/ESW system. The sodium hypochlorite injection system has an interlock that protects the piping by limiting the total chlorine concentration to a maximum value of 0.05 ppm which is similar to the existing Circulating Water and GSW chlorination systems. Seismic requirements were evaluated and determined not to be adversely effected, because the new piping is not safety-related and does not tie into any seismically designed piping. However, the new piping has been seismically supported and routed to maintain pipe integrity; thus preventing any impact on the RHRSW and ESW pumps. The modifications were installed to the same quality requirements

DCP No. 1382A (continued) and construction codes as the existing systems. All fire protection requirements for pipes penetrating fire barriers were met. No unreviewed safety questions existed.

DCP No. 1385 Decommission Head Spray

Description and Basis for Change: The flanged joints in the head spray line above the well seal were a source of drywell leakage and had required extensive maintenance during the critical time of vessel reassembly and startup. The solution was to (1) disconnect the removable head spray line at the well seal and vessel nozzle flanged joints; (2) remove the piping, valves and associated supports; (3) install blind flanges at the well seal and vessel nozzle; (4) close and de-energize motor-operated valves MO-1900 and MO-1901.

Summary of Safety Evaluation: This modification was safety-related. The plant was in the cold shutdown condition during this modification. The integrity of the reactor coolant pressure boundary and the primary containment remained unchanged by this design change. No credit is taken for performance of the head spray mode of the Residual Heat Removal (RHR) system in any accident analysis. The blind flanges were installed in accordance with applicable piping code requirements (ANSI B-31.7, Class 1). The closure and de-energization of MO-1900 and MO-1901 assures that the isolation function is met. The head spray piping external to the drywell was not affected by this design change. No unreviewed safety questions existed.

DCP No. 1387 General Service Water (GSW) Isolation Valves

Description and Basis for Change: This modification consisted of installing two 6-inch gate valves in the General Service Water (GSW) supply and return lines that serve both reactor feed pump motors and lube oil coolers and both Electro-Hydraulic Control (EHC) system oil coolers. The purpose of these valves is to permit isolation of this particular GSW cooling loop for maintenance purposes without shutting down the entire GSW system.

Summary of Safety Evaluation: This modification was not safety-related. This modification was performed while the plant was in the cold shutdown condition. The installed valves will remain open at all times while the plant is in operation. The only time the valves will be closed is during plant shutdown when maintenance is required downstream of the valves. All pipe supports were designed to the original construction code for this line (ANSI B-31.1). Structural failure of this line during a Design Basis Earthquake (DBE) will not imperil any safety-related equipment. This change has not affected the design intent or function of the GSW system. No unreviewed safety questions existed.

DCP No. 1388 Jet Pump Sensing Line Modifiction/Hold Down Beam Replacement

Description and Basis for Change: Vibration has caused fatigue failure of jet pump sensing lines at two foreign plants. These failures occurred at the support bracket where the sensing line is attached to the jet pump. The resonant vibration and subsequent failure is a result of the natural frequency of the sensing line coinciding with the passing frequency of the recirculation pump vanes. The problem occurs when the ratio of sensing line natural frequency to vane passing frequency approaches unity. General Electric performed a generic study of the jet pump sensing lines for a number of plants including the DAEC. The jet pump vibration frequencies used were obtained during pre-operational and startup testing at the DAEC. This evaluation concluded that the DAEC was vulnerable to this concern. This design modification installed a clamp and beam assembly that is attached to the lower support bracket where the sensing line is attached to the jet pump diffuser. The clamp and beam reinforced the lower support bracket and changed the natural frequency of the sensing line. General Electric provided and installed the clamps on the 12 jet pumps which were determined to be most vulnerable to this concern.

Additionally, DCP 1388 involved the replacement of the jet pump hold-down beams. Jet pump hold-down beam cracking has occurred at several operating BWRs (see NRC Bulletin 80-07). The beam assembly holds the jet pump inlet mixer section to the jet pump diffuser. Failure of this beam will result in significant degradation of the performance of the recirculation system. The cracks in the beams are caused by Intergranular Stress Corrosion Cracking (IGSCC). General Electric performed calculations to estimate the crack initiation and propagation rates based on available material property data and the calculated stresses during failure. Due to the fact that BWR-4 beams are slightly thicker and have lower operating stress levels than the beams for other designs, the crack initiation was not expected for over 10 years. New General Electric heat-treated beams were installed to replace the present beams. The new beam design is expected to last for more than forty (40) years. These estimates are based on laboratory tests conducted with specimens which have metallurgical characteristics identical to the original beams. The physical size and function of the new beams are the same as the replaced beams. General Electric provided and installed hold down beams on all 16 jet pumps at the DAEC.

Summary of Safety Evaluation: This modification was safety-related. It was performed when the plant was in the cold shutdown condition. The new-hold down beams perform the same function as the ones that were replaced. The enhanced material properties of the new beams reduce the likelihood of jet pump hold-beam failure and jet pump malfunction. The installation of jet pump sensing line clamps reduce the likelihood of sensing line failure due to resonant vibration by changing its natural frequency. The clamp design is adequate to withstand operational and seismic excitation. No unreviewed safety questions existed.

DCP No. 1389 Weld Overlay Flapping on Recirculation System

Description and Basis for Change: This DCP evaluated and controlled the flapping process on eight weld overlays on the recirculation system. The weld overlays were originally completed under DCP 1310. However, new Electric Power Research Institute (EPRI) standards for ultrasonic examinations require a smoother surface on the weld overlays. Flapping provided a smoother surface finish to produce a more reliable ultrasonic examination of the weld overlays.

Summary of Safety Evaluation: This modification was safety-related. The plant was in the cold shutdown condition when this modification was performed. The original design intent for each weld overlay was not changed by this modification. The flapping process was evaluated and controlled by this DCP to ensure that the integrity of each full-structural weld overlay was maintained. Flapping these weld overlays improved their surface finish and increased the reliability of their ultrasonic examinations. No unreviewed safety questions existed.

DCP No. 1390 Remote Shutdown Panel Structural Improvement

Description and Basis for Change: The structural modification of the remote shutdown panel (1C388) consisted of removing sections of 1/8" thick exterior plate in the vicinity of the corner joints, welding gusset plates directly to the angle iron structure, smoothing the surface contour for appearance, and painting all exposed metal surfaces. The addition of gusset plates and welds was necessary due to inadequate butt welds used in joining the structural members at the corner joints.

This deficiency was reported pursuant to 10CFR Part 21 in LER 87-008.

Summary of Safety Evaluation: This modification was not safety-related. This design change ensured that the supporting structure of panel 1C388 met the original seismic design. This modification did not affect any other systems or components. It did not affect the design function of the remote shutdown panel but merely provided adequate seismic support for the panel. No unreviewed safety questions existed.

DCP No. 1391 Replace Discs in Pressure Safety Valves PSV-4403 and PSV-4404

Description and Basis for Change: Disc damage was discovered during testing of main steam safety valves PSV-4403 and PSV-4404. DCP 1391 was initiated to repair the discs in main steam safety valves PSV-4403 and PSV-4404. The disc in PSV-4403 was constructed of ASTM A565, Gr.616, condition "HT" material and was repaired. The disc in PSV-4404 was constructed of ASTM A-479, type 410 material. Rather than repair this disc, it was replaced with a disc constructed of ASTM A565, Gr.616, condition "HT" material, which was evaluated as a suitable replacement.

DCP No. 1391 (continued)

Summary of Safety Evaluation: This modification was safety-related. It was performed when the plant was in the cold shutdown condition. The new disc did not affect the operation or function of the main steam safety valve. The replacement disc either meets or exceeds the requirements for the original disc as stated in the ASME Code, Section XI, 1980 Edition Through Winter 1981 Addenda. The main steam safety valves were retested satisfactorily prior to their installation. No unreviewed safety questions existed.

DCP No. 1392

Replace HGA Relays

Description and Basis for Change: General Electric, in Service Advisory Letter (SAL) 174.1, identified a potential problem with its HGA relays under seismic conditions. Specifically, the contacts, which are closed while the relay is deenergized can chatter at horizontal accelerations above 0.35gs. The DAEC FSAR requires that all instrumentation required for nuclear safety must be qualified to seismic acceleration levels of 1.5gs horizontal and 0.5gs vertical over a frequency range of 0.25 to 33 hz. The results of an indepth evaluation identified fourteen relays that required modification to prevent possible intermittent operation of a safety-related circuit during an accident condition while a seismic event is taking place. These fourteen HGA relays were replaced with GE HFA series relays. The replacement HFA relays were seismically evaluated and were found acceptable.

This design deficiency was reported pursuant to 10CFR Part 21 in LER 87-021.

Summary of Safety Evaluation: This modification was safety-related. The FSAR does not specify a particular type of relay for a given application. Both the HGA and HFA relays are used in safety-related circuitry including the logic channels that were affected by this modification. The HFA relays met Class 1E electrical qualifications and IEEE 344-1975 seismic qualifications. Installation of these new HFA relays in panels 1C30, 1C32, 1C33, 1C43, and 1C44 did not affect the seismic qualification of the panels. The replacement HFA relays meet and exceed the seismic qualification requirements of the DAEC FSAR and perform the same functions electrically as the HGA relays. This modification returns the plant to a condition within its original design basis. No unreviewed safety questions existed.

SECTION B - PROCEDURE CHANGES

During 1987, various procedures as described in the safety analysis report were revised and updated. All changes were reviewed against 10 CFR Part 50.59 by the DAEC Operations Committee. No procedure changes were performed that constitute unreviewed safety questions.

All Special Test Procedures (SpTPs) performed in 1987 were also reviewed by the DAEC Operations Committee. No unreviewed safety questions were found to exist. Summaries of these special tests and their safety evaluations are found below.

TEST NO.	TITLE/DESCRIPTION
SpTP No. 129	Reactor Water Cleanup (RWCU) System Area Differential Temperatures

The purpose of this test was to collect operating data on the temperature differential between air entering the reactor water cleanup area and air exiting this area. This data was used to ensure that the setpoints for the steam leak detection logic were correct as per Technical Specification Table 3.2-A. The test also served to resolve an NRC open item, identified in NRC Inspection Report 50-331/86002 relating to the use of a plastic cover on door 249 that restricted air flow into the reactor water cleanup system heat exchanger room.

This test was performed on March 2, 1987.

Summary of Safety Evaluation: The operation of the plant during the test was within design requirements and Technical Specification limits. Additionally, the plant was protected from an unnecessary isolation of the RWCU system and, in the event of a RWCU system leakage condition, the Group V primary containment isolation function would have occurred per system design. At no time during this test were both sides of the RWCU steam leak detection system inoperable simultaneously. Based on the above, no unreviewed safety questions existed.

SpTP No. 131	Main Control Room Ventilation System Exhaust Isolation Damper Leakage Test

In the event of a Loss of Coolant Accident (LOCA), the Control Room is isolated and required to be maintained at a positive pressure such that habitability can be maintained. The purpose of this special test was to determine at what positive pressure the control room atmosphere can be maintained during Standby Filter Unit (SFU) operation. Specifically, the test would determine whether air leakage past control building isolation damper 1V-AD-31A reduces the maximum attainable differential pressure inside the Control Room.

This test was performed on January 15, 1987.

Summary of Safety Evaluation: Since only one SFU train was activated during the test, one was always available to perform the required protective function during reactor operation. The use of one train during actual accident conditions has previously been evaluated in the FSAR. Based on the above, no unreviewed safety questions existed.

TEST NO.	TITLE/DESCRIPTION

SpTP No. 132 Effect of Battery Room Exhaust Flow on Control Room Pressure

In the event of a Loss of Coolant Accident (LOCA), the Control Room is isolated and required to be maintained at a positive pressure such that habitability can be maintained. The purpose of this test was to determine the effect of reduced battery room exhaust flow on the positive pressure in the Control Room during Standby Filter Unit (SFU) operation. By temporarily reducing the Battery Room Exhaust flow during the test, a substantial increase in control room positive pressure was achieved. A permanent change to the position of the battery room damper was not made and. the damper was returned to its normal position upon completion of the test.

This test was performed on February 11, 1987 and February 22, 1987

Summary of Safety Evaluation: Since only one SFU train was activated during the test, one was always available to perform the required protective function during reactor operation. Additionally, the throttling of battery room air flow did not reduce the plant margin of safety since the 1-1/2 hours required to perform the test was significantly less than the time required to reach the hydrogen concentration by volume (in air) necessary to form an explosive mixture. Based on the above, no unreviewed safety questions existed.

SpTP No. 133 Motor-Operated Valve MO-2202 Opening Operability Testing At Abnormal Differential Pressure

The purpose of this test was to demonstrate valve MO-2202 operability at its maximum-expected accident differential pressure of 1110 psid and to determine whether any adjustments or repairs were required. The valve operability demonstration at 1110 psid was based on an NRC Bulletin 85-03 requirement.

This test was performed on May 23, 1987

Summary of Safety Evaluation: This test was performed while the reactor plant was in the cold shutdown condition. The test pressure was isolated from the reactor, High Pressure Coolant Injection (HPCI) turbine and other connected systems and interfacing piping was vented to atmosphere. Additionally, the test was performed while the HPCI system was shutdown for normal maintenance and normal design backup systems were still available. This test did not effect other ECCS systems. The test pressure of 1110 psig was below the system design pressure of 1365 psig and relief protection was provided on the hydrostatic test rig (pressure source). Based on the above, no unreviewed safety questions existed.

TEST NO.	TITLE/DESCRIPTION

SpTP No. 134 Motor-Operated Valve MO-2312 Opening Operability Testing at Abnormal Differential Pressure

The purpose of this test was to demonstrate valve MO-2312 operability at its maximum-expected accident differential pressure of 1289 psid. The valve operability test at 1289 psid was performed to meet an NRC requirement in Bulletin 85-03 to demonstrate valve operability at its maximum-expected accident differential pressure and to determine whether any adjustments or repairs were required.

This test was performed on May 28, 1987.

Summary of Safety Evaluation: This test was performed while the reactor plant was in the cold shutdown condition. The test pressure was isolated from the reactor, High Pressure Coolant Injection (HPCI) pump and Condensate Storage Tank (CST) and other connected systems and interfacing piping was vented to atmosphere. Additionally, the test was performed while the HPCI system was shutdown for normal maintenance and normal design backup systems were still available. This test did not effect other ECCS systems. The test pressure of 1289 psig was below the maximum service condition pressure of 1590 psig and relief protection was provided on the hydrostatic test rig (pressure source). Based on the above, no unreviewed safety questions existed.

SpTP No. 135 Motor Operated Valve MO-2317 Opening Operability Testing At Abnormal Differential Pressure

The purpose of this test was to demonstrate valve MO-2318 operability at its maximum-expected accident differential pressure at 1416 psid. The valve operability test at 1416 psid was performed to meet an NRC requirement in Bulletin 85-03 to demonstrate valve operability at its maximum-expected accident differential pressure and to determine whether any adjustments or repairs were required.

This test was performed on May 23, 1987.

Summary of Safety Evaluation: This test was performed while the reactor plant was in the cold shutdown condition. The test pressure was isolated from the reactor and the High Pressure Coolant Injection (HPCI) pump and other connected systems and interfacing piping was vented to atmosphere. Additionally, the test was performed while the HPCI system was shutdown for normal maintenance and normal design backup systems were still available. This test did not effect other ECCS systems. The test pressure of 1416 psig was below the maximum service condition pressure of 1590 psig and relief protection was provided on the hydrostatic test rig (pressure source). Based on the above, no unreviewed safety questions existed.

TEST NO.	TITLE/DESCRIPTION

SpTP No. 136 Motor Operated Valve MO-2322 Opening Operability Testing At Abnormal Differential Pressure

The purpose of this test was to demonstrate valve MO-2322 operability of its maximum-expected accident differential pressure of 123 psid. The valve operability test at 123 psid was performed to meet an NRC requirement in Bulletin 85-03 to demonstrate valve operability at its maximum-expected accident differential pressure and to determine whether any adjustments or repairs were required.

This test was performed on March 30, 1987.

Summary of Safety Evaluation: This test was performed while the reactor plant was in the cold shutdown condition. The test pressure was isolated from the reactor and other connected systems and interfacing piping was vented to atmosphere. Additionally, the test was performed while the HPCI system was shutdown for normal maintenance and normal design backup systems were still available. This test did not effect other ECCS systems. The test pressure of 123 psig was below the initial system hydrostatic test pressure of 188 psig and relief protection was provided on the hydrostatic test rig (pressure source). Based on the above, no unreviewed safety questions existed.

SpTP No. 137 Motor Operated Valve MO-2512 Opening Operability Testing At Abnormal Differential Pressure

The purpose of this test was to demonstrate valve MO-2512 operability at its maximum-expected accident differential pressure at 1417 psid. The valve operability test at 1417 psid was performed to meet an NRC requirement in Bulletin 85-03 to demonstrate valve operability at its maximum-expected accident differential pressure and to determine whether any adjustments or repairs were required.

This test was performed on May 28, 1987.

Summary of Safety Evaluation: This test was performed while the reactor plant was in the cold shutdown condition. The test pressure was isolated from the reactor and other connected systems and interfacing piping was vented to atmosphere. Additionally, the test was performed while the RCIC system was shutdown for normal maintenance and normal design backup systems were still available. This test did not effect other ECCS systems. The test pressure of 1417 psig was below the maximum service condition pressure of 1460 psig and relief protection was provided on the hydrostatic test rig (pressure source). Based on the above, no unreviewed safety questions existed.

TEST NO.	TITLE/DESCRIPTION

SpTP No. 138 Motor Operated Valve MO-2517 Opening Operability Testing At Abnormal Differential Pressure

The purpose of this test was to demonstrate valve MO-2517 operability at its maximum-expected accident differential pressure of 128 psid. The valve operability test at 128 psid was performed to meet an NRC requirement in Bulletin 85-03 to demonstrate valve operability at its maximum-expected accident differential pressure and to determine whether any adjustments or repairs were required.

This test was performed on April 10, 1987.

Summary of Safety Evaluation: This test was performed while the reactor plant was in cold shutdown condition. The test pressure was isolated from the reactor and other connected systems and interfacing piping was vented to atmosphere. Additionally, the test was performed while the RCIC system was shutdown for normal maintenance and normal design backup systems were still available. This test did not effect other ECCS systems. The test pressure of 128 psig was below the initial system hydrostatic test pressure of 188 psig and relief protection was provided on the hydrostatic test rig (pressure source). Based on the above, no unreviewed safety questions existed.

SpTP No. 139 Procedure for Low-Amplitude Testing Of Class 1E Control Room Panels

The purpose of this test was to determine the dynamic properties of the control panels in the control room. In order to evaluate the control panel anchorage and the seismic adequacy of the Class 1E equipment mounted in the control panels, the control panels were subjected to low-amplitude vibrational forces by the employment of a mechanical shaker.

This test was performed during the time period of March 31, 1987 through April 8, 1987.

Summary of Safety Evaluation: The maximum input from the mechanical shaker to the panel did not exceed the previously evaluated values for the vertical and horizontal accelerations of the 786'-0" floor elevation in the control building during the design basis earthquake. Dynamic amplification of the input motion did not exist because the input motion of the shaker was applied at (or near) the top of the control panels. Additionally, random vibration input to seismic category I equipment had been previously evaluated and enveloped by the existing analysis discussed in the FSAR. Based on the above, no unreviewed safety questions existed.

TEST NO.	TITLE/DESCRIPTION

SpTP No. 140 ANI 5-Year Diesel Fire Pump Operability Test

The purpose of this test was to demonstrate the operability of, and verify the capacity of, the diesel-driven fire pump. This test is required by the American Nuclear Insurers (ANI).

This test was performed on May 20, 1987.

Summary of Safety Evaluation: The performance of this test was not safety related and did not alter the function or operation of any safe shutdown equipment. This test was performed outdoors so as not to damage any safety-related equipment. There is no cross-connection between the fire protection water and any potentially contaminated systems. The river water supply pumps remained operable throughout this special test procedure such that intake basin level remained above the low limits. Based on the above, no unreviewed safety questions existed.

SpTP No. 141 ANI 5-Year Electric Fire Pump Operability Test

The purpose of this test was to demonstrate the operability of, and verify the capacity of, the electric motor-driven fire pump. This test is required by the American Nuclear Insurers (ANI).

The test was performed on May 20, 1987.

Summary of Safety Evaluation: The performance of this test was not safety-related and did not alter the function or operation of any safe shutdown equipment. This test was performed outdoors so as not to damage any safety-related equipment. There is no cross-connection between the fire protection system and any potentially contaminated systems. The river water supply pumps remained operable throughout this special test procedure such that intake basin level remained above the required limits. Based on the above, no unreviewed safety questions existed.

SpTP No. 143 Motor Operated Valve MO-2316 Operability Testing at Normal Differential Pressure of 300 PSIG and 1200 PSIG.

The purpose of this test was to demonstrate the operability of valve MO-2316 and to determine the valve operator thrust requirements for full pressure operation. This test was performed to meet an NRC requirement in Bulletin 85-03.

This test was performed on July 1, 1987.

Summary of Safety Evaluation: The test was performed while the High Pressure Coolant Injection (HPCI) system was operating for normal surveillance testing and normal design backup systems were available. The maximum test pressure of 1200 psig was below the system maximum service condition pressure of 1590 psig. The test pressure was isolated from the reactor and other connected systems. This test did not effect other ECCS systems. The test was performed within the design capabilities of the HPCI system during normal plant surveillance testing. Based on the above, no unreviewed safety questions existed.

TEST NO.	TITLE/DESCRIPTION

SpTP No. 145 Containment Spray Logic Functional Test of Selected Relay Contacts.

The purpose of this test was to demonstrate operability of all initiation logic components and contacts of the containment spray subsystem of the Residual Heat Removal (RHR) system that were not normally tested under the scope of the normal Surveillance Test Procedure (STP) program. This test was performed in response to NRC Inspection Report 50-331/87004.

This test was performed on June 5, 1987.

Summary of Safety Evaluation: The function and operation of the RHR system was not altered by the special test procedure. The test involved testing certain contacts of the system logic by use of jumpers and/or test switches to simulate closed contacts where open contacts existed due to there being no actuation/initiation signal. This test ensured all contacts in the logic circuits were tested in accordance with the original design intent. Precautions were taken such that no valve cycling occurred, and the logic was returned to its pre-test configuration upon test completion. Based on the above, no unreviewed safety questions existed.

SpTP No. 146 LPCI Trip System Logic Functional Test of Selected Relay and Contacts

The purpose of this test was to demonstrate operability of all Low Pressure Coolant Injection (LPCI) system logic components and contacts that were not tested under the scope of the normal Surveillance Test Procedure (STP) program. Specifically, verification of the opening and closing of all contacts utilized in the LPCI mode of the Residual Heat Removal (RHR) system logic was performed. This test was performed in response to NRC Inspection Report 50-331/87004.

This test was performed on June 5, 1987.

Summary of Safety Evaluation: The function and operation of the RHR system was not altered by the special test procedure. The test involved testing certain contacts of the system logic by use of jumpers and/or test switches to simulate closed contacts where open contacts existed due to there being no actuation/initiation signal. This test ensured all contacts in the logic circuits were tested in accordance with the original design intent. Precautions were taken such that no valve cycling occurred, and the logic was returned to its pre-test configuration upon completion of the test. Based on the above, no unreviewed safety questions existed.

TEST NO.	TITLE/DESCRIPTION

SpTP No. 147 Core Spray Trip System Logic Functional Test of Selected Relays and Contacts

The purpose of this test was to demonstrate operability of Core Spray system initiation logic components and contacts that were not normally tested under the scope of the normal Surveillance Test Procedure (STP) program. Specifically, verification of the opening and closing of contacts within the core spray logic was performed. This test was performed in response to NRC Inspection Report 50-331/87004.

This test was performed on June 4, 1987.

Summary of Safety Evaluation: The function and operation of the Core Spray system was not altered by the special test procedure. The test involved testing certain contacts of the system logic by use of jumpers and/or test switches to simulate closed contacts where open contacts existed due to there being no actuation/ initiation signal. This test ensured all contacts in the logic circuits were tested in accordance with the original design intent. Precautions were taken such that no valve cycling occurred, and the logic was returned to its pre-test configuration upon completion of the test. Based on the above, no unreviewed safety questions existed.

SpTP No. 148 Feedwater Level Set Test

This test provided a mechanism for establishing the optimum liquid level in the low pressure feedwater heaters at the DAEC. Because of recent problems experienced at the DAEC with vibration induced failures of feedwater heater tubes, it was recommended that the liquid level be raised. Raising the liquid level in a feedwater heater will preclude the occurance of steam entrainment in the drain cooler zone of a feedwater heater, and thus eliminate a root cause of the vibration damage mentioned above.

Establishing an optimum feedwater heater liquid level will improve plant heat-rate, since there is an optimum level where extraction steam can provide the most benefit in preheating the feedwater.

This test was performed during the time period of July 17, 1987 through August 6, 1987.

Summary of Safety Evaluation: The consequences of a malfunction of a feedwater heater level control are a loss of a feedwater heater or a turbine trip. Both of these events have been previously evaluated in the Updated FSAR and bound any event that could be caused by the performance of this test. Feedwater heater liquid levels were varied in only one feedwater heater at a time

TEST NO.	TITLE/DESCRIPTION
SpTP No. 148 (continued)	and the incremental level changes made were small relative to level variances experienced during operational transients (the test was performed at constant reactor pressure levels). The original feedwater heater liquid level was restored upon completion of the test. Based on the above, no unreviewed safety questions existed.
SpTP No. 150	HPCI System to CV2315, Test Valve Data

The purpose of this test was to demonstrate and record the valve position of CV2315 and the turbine speed for various High Pressure Coolant Injection (HPCI) flows and pressures. Additionally, this test provided data for the monthly and quarterly surveillance test procedure to meet an NRC commitment to incorporate cold quick starts of the HPCI turbine into the applicable STPs.

This test was performed on November 17, 1987.

Summary of Safety Evaluation: Performing this special test did not adversely affect the HPCI system. This special test required operation of the HPCI pump above its normal flow of 3000 gpm, but below its maximum rated flow at maximum rated turbine speed (3900 rpm). The pump discharge pressure was varied above and below the normal pressure, but below the rated pressure at the maximum rated turbine speed. The auto-initiation function of the HPCI was unaffected by this test. The probability of an inadvertent HPCI injection was unchanged by following the normal Surveillance Test Procedure (STP) for valve manipulation. Based on the above, no unreviewed safety questions existed.

SpTP No. 151 RCIC System to MO2515, Test Valve Data

The purpose of this test was to demonstrate and record the valve position of MO2515 and the turbine speed for various Reactor Core Isolation Cooling (RCIC) pump flows and pressures. Additionally, this test provided data for the monthly and quarterly surveillance test procedure to meet an NRC commitment to incorporate cold quick starts of the RCIC turbine into the applicable STPs.

This test was performed on November 13, 1987.

Summary of Safety Evaluation: Performing this special test did not adversely affect the RCIC system. This special test required operation of the RCIC pump above its normal flow of 400 gpm, but below its maximum rated flow at maximum rated turbine speed (4500 rpm). The pump discharge pressure was varied above and below the normal pressure, but below the rated pressure at the maximum rated turbine speed. The auto-initiation function of the RCIC system was unaffected by this test. The probability of an inadvertent RCIC injection was unchanged by following the normal Surveillance Test Procedure (STP) for valve manipulation. Based on the above, no unreviewed safety questions existed.

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 1987.

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 1987 in accordance with the requirements of Technical Specification 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 1987. There were no relief valve failures during 1987. There was one relief valve challenge during 1987. This event is described below:

Date	Event Description
June 29, 1987	Relief valves PSV-4400, -4401, -4402, -4405, -4406 and -4407 were opened and closed during the satisfactory completion of a normal surveillance test.