Duane Arnold Energy Center

1983

Annual Report of Facility Changes, Tests, Experiments, and Safety and Relief Valve Failures and Challenges

IE24

Section A
PLANT DESIGN CHANGES

This section has been prepared in accordance with the requirements of 10CFR, Part 50.59(b)

A. Plant Design Changes

This section contains brief descriptions of and reason for plant design changes completed during the calendar year 1983, and summaries of the safety evaluations for those changes, pursuant to the requirements of 10CFR, Part 50.59(b).

There are two major milestones for completion of work at DAEC, "site closure" and "engineering closure". The latter generally lags the former by several months. The 1982 annual report excluded 14 engineering packages that had received site closure but not engineering closure in 1982. These are included for completeness. Three engineering packages which received site closure in 1982 and should have been included in the 1982 report are denoted below by an asterisk (*). Future submittals will uniformly report packages that receive site closure in the calendar year of interest.

DCR No. 270 — Off-gas Valve Operator Change

Description of Change: Changed component operator so that CV-4110A fails open instead of failing closed.

Reason for Change: GE P&ID was mistakenly interpreted as showing the failed position of the valve rather than the normal position. The valve was purchased and installed accordingly. Changing the operator to fail open precludes power plant shutdown resulting from that single event failure.

Safety Evaluation: This change has no effect on nuclear safety considerations because it does not interface with any safety related equipment or function.

DCR No. 426 — CAD System Charging Compressor

Description of Change: N_2 compressor was added to CAD system. Compressor takes suction from normal N_2 system and is used to charge the CAD tanks.

Reason for Change: Compressor allows repressurization of the CAD and thereby eliminates the previous requirement for tube trailer equalization.

Safety Evaluation: The change added piping and equipment which interface with a safety related system. However, the addition is connected through seismicly qualified valve and piping which meets the nuclear quality requirements of the safety related systems. Therefore, it presents no additional hazards which were not already considered in the safety analysis report.

DCR No. 428 — Crosstie from 1T-62B to other Phase Separators

Description of Change: The change provided connecting piping and valve to allow the discharge of the Floor Drain Sludge Discharge Mixing Pump, 1P-71B, to be routed to the waste sludge tank, 1T-62A, or to the condensate phase separator tanks, 1T-202 A & B.

Reason for Change: The change provided a crosstie similar to the existing crosstie from 1T-62A to 1T-202A & B. These crossties provide the operator greater processing flexibility and provide a capability to more completely utilize the ion exchange capacity of the resin which was backwashed to those tanks since each sludge tank inlet includes a blowdown connection from the waste and from drain collector tanks. Before processing the drainage from these collector tanks, the waste can be routed to a sludge tank for initial ion exchange. The DCR completes a crosstie network for both sludge tanks.

Safety Evaluation: Since the change does not involve a safety system nor interface with a safety system, it does not present hazard considerations not described or implicit in the safety analysis report.

DCR No. 601* Add Nitrogen Vaporizer Freeze Protection

Description of Change: Freeze protection was provided for nitrogen vaporizer system. Piping drains and heat tracing to instrument tubing was included.

Reason for Change: Piping and instrumentation had frozen up. This change enhances the operation of the system.

Safety Evaluation: The additions do not present significant hazard considerations not described or implicit in the safety analysis report. The additions do not change the intended design function of this system and do not interface with any safety related systems.

DCR No. 685 Turbine Lube Oil Tank Platform

Description of Change: Removed existing handrail and installed new grating and handrails at level 751' on the east side of the turbine lube oil tank.

Reason for Change: A permanent access to the East end of the turbine lube oil tank at elevation 751'-0" was required.

Safety Evaluation: The change does not affect any safety system.

DCR No. 699B DAEC Security Project

Description of Change: The work included erection of a security building which will house security control and support equip, security personnel, and personnel search equipment; modifications to the control room/computer room to install a secondary alarm station (SAS); and modification to a portion of the reactor/recombiner room to furnish new doors.

Reason for Change: The design requirements for the security project were based on federal regulation part 10 section 73.55 and Iowa Electric's "DAEC Modified Amended Security plan (MASP)".

Safety Evaluation: The major portion of this work was performed outside of the power block and did not interface with any safety related system(s).

Intake structure: As this change was outside of the original evaluation done on DCR 699B, the following evaluation was performed. This wall was constructed in a seismic class I structure and was designed with this as a consideration and per the DAEC FSAR. Refer to calculation C-79-2 dated 7-2-79. Other conditions remain as in DCR 699B. FSAR Fig. 12.1.-15 was changed to reflect the as-built condition.

Compartmentilization of MC 1B34 At Ele. 786'-0" Reactor Building: As this change was outside of the original evaluation done on DCR 699B, the following evaluation was performed. This cage is seismic class I and was designed on that basis (Reference Cal. C-79-4 on file in Nuclear Generation). No FSAR change was required.

DCR No. 705 Replaces Bolts on Safety Related MOV

Description of Change: This change replaced bolts on the safety related motor operated valve (MOV) Limitorque operators.

Reason for Change: Originally the MOVs at DAEC had several types of bolt material specifications in use on the Limitorque operators. With the use of different bolting materials there was no assurance of a consistent system of applying a specified torque value on the bolts of the Limitorque operators. By replacing the previous Limitorque operator bolts with a common bolt material specification having an associated torque value based on operator size there is increased assurance that the safety related MOVs will operate as designed.

Safety Evaluation: The change does not present significant hazards or considerations not described or implicit in the Safety Analysis Report. The purpose of the change was to replace the several types of bolt materials used on the MOV Limitorque operators with one type of bolt material having an associated bolt torque based on operator size which increases assurance in the safety related MOV performance.

DCR No. 714 Reactor Water Clean Up System Modification

Description of Change: Piping for the reactor water cleanup system, located in the reactor water cleanup heat exchanger room, was modified. The addition of the piping and valves allows modification of the normal letdown path of the reactor water cleanup system so that reactor water cleanup pumps are downstream of the non-regenerative heat exchangers during all modes of power operation. During shutdown modes when vessel pressure is reduced the letdown path must be valved back to its original configuration with the pumps upstream of the heat exchangers to ensure that sufficient NPSH is available to the pumps.

Reason for Change: The basis for this change is to improve reliability of the reactor water cleanup pumps by placing them in the cooler process water environment downstream of the heat exchangers.

Safety Evaluation: The change does not present significant hazards or considerations not described or implicit in the Safety Analysis Report, since the change does not affect any safety related equipment and the basic design functions are not altered.

DCR No. 758* Exciter Hardware Securing

Description of Change: This change is to increase the strength and stiffness of the exciter cooler enclosure by:

1. Increasing the number of enclosure mounting bolts.
2. Providing metal to metal contact between the cooler enclosure and the alternator frame.
3. Stiffening the enclosure.

Reason for Change: To prevent breakage of the cooler enclosure.

Safety Evaluation: This change poses no unreviewed safety questions since the change does not interface with any safety related components.

DCR No. 771 Control Rod Drive Return Line Modification

Description of Change: The RPV nozzle was examined in the CRD return line and a spectacle flange was installed. The orificed check valve in the exhaust header was replaced with a pair of pressure equalizing valves. Carbon steel pipe in the flow stabilizer loop was replaced with stainless steel.

Reason for Change: The Control Drive Return Line was modified to eliminate the potential for thermal fatigue cracking of the RPV nozzle. These fatigue cracks would be the result of cold (50 to 100°F) CRD water flowing through this line into the hot RPV, which creates high thermal stresses at the nozzle. The situation had been temporarily fixed by closing a valve in this line. However, over a period of time, any leakage past this valve would have fatigued the nozzle. Because of this, a spectacle flange was installed in this line to assure no leakage. Use of a spectacle flange instead of removing the line is preferred so that operations can use the line as an emergency makeup source to maintain vessel level during a limited accident.

With the return line isolated, a higher pressure exists in the remaining section of the return line upstream from the isolation. This would cause a backflow through the exhaust header orificed check valve. This flow is dispersed to the RPV via leakage across the nominally closed directional control solenoid valves of the HCUs. During rod movement, the exhaust flow from the translating drive would be dispersed through the HCU solenoid valves of the latched drives. To avoid subjecting the HCU solenoid valves to a continuous reverse flow, the orificed check valve in the exhaust header was replaced with a pair of pressure equalizing valves set at approximately 80 psid. These valves allow recharging of the exhaust water header to avoid excessive initial withdraw speed of the CRD following a scram or other conditions when a high differential pressure across the solenoid speed control valve exists.

To keep corrosion products from entering the CRD System, all carbon steel pipe in the flow stabilizer loop was replaced with stainless steel. This was accomplished by rerouting the exhaust water header and stabilizing flow lines to bypass the small amount of carbon steel line in the system.

Safety Evaluation:

a) The effect on plant safety with the return line out of service was verified with calculations by GE and by isolation test. The DCR verified that the flow calculated by GE was correct by a flow test.

b) The likelihood of cracks developing in the reactor nozzle has decreased. The quantity of high pressure make-up water that the CRD System can provide to the reactor has decreased. Based on this fact, GE has performed calculations to evaluate the flow required versus the flow available, and determined that the required flow is met with the return line isolated. The NRC accepted this calculation in NUREG 0619. Therefore, this does not increase the probability of occurrence or magnitude of the consequences of an accident or malfunction.

c) Makeup to the reactor has been shown by calculations and was verified by flow tests. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR has not been created.

d) The safety of the plant was not reduced. This has been verified by isolation test.

DCR No. 792 Replace V41-83 with Drain Trap

Description of Change: Replaced a spool piece including V41-83 (drain off) with new line having steam trap with manual isolation valves and bypass.

Reason for Change: This change was required to improve the moisture removal capability of the offgas system. The offgas system was not draining properly and moisture in the system was adversely affecting performance.

Safety Evaluation: The change did not present significant hazards or considerations not described or implicit in the Safety Analysis Report. The drain line itself does not perform an active safety function. However, it is part of the pressure boundary of the offgas system which is an ASME Section III system. Therefore all welding and inspections (including hydrotest) were performed and documented in accordance with the applicable sections of the ASME Boiler and Pressure Vessel Code, Section III, Class 3. This DCR specified the applicable IE procedures which were followed. This piping is seismic class II and therefore a seismic class I analysis was not required.

DCR No. 803A Addition of Chemical Treatment Equipment to CIRC Water System (Electrical Portion)

Description of Change: Provided electrical supply for circulation water treatment system installed by DCR 803.

Reason for Change: Electrical power was required for exhaust fans and heat tracing of chemical tanks.

Safety Evaluation: The changes do not present significant hazards or considerations as set forth in the safety analysis report because they do not interface with any safety related system.

DCR No. 852 Flowmeter for Makeup Demineralizer

Description of Change: The rotameter on the makeup demineralizer drain was replaced by a flow orifice and d/p gauge calibrated in GPM.

Reason for Change: The previous flowmeter on the makeup demineralizer drain was a Brooks rotometer. The rotometer was unstable and inaccurate in indicating flow. Additionally, the indicator glass on the rotometer was broken.

Safety Evaluation: This change poses no unreviewed safety questions since the change does not interface with any safety system or interface with safety functions.

DCR No. 855 Reactor Water Cleanup System Modification

Description of Change: Modification of reactor water cleanup system isolation logic, so that CV2729 closes whenever MO2700 or MO2701 closes.

Reason for Change: Past operating experience has shown that rapid depressurization causes steam flashing while RWCU system isolation is in the blowdown mode. This flashing may cause pump and seal damage.

Safety Evaluation: The design change is not safety-related. It does interface with safety-related system, but the probability of an accident is not increased since the relays K11 & K12 which are Q-components will open CV2729 in case of loss of power or malfunction of the relay itself. This design change will not block an isolation signal calling for closing MO2700 & MO2701.

DCR No. 879 Cooling Tower Fire Hose Sheds

Description of Change: This DCR involves the installation of sheds around the affected hose houses. These houses are framed with 3" diameter carbon steel pipe and 3 x 3 x 1/4 angle iron. The sheds are encased with galvanized, corregated sheet metal roof, painted metal siding and a flexible strip door.

Reason for Change: This DCR is to provide shelter from ice buildup for the hose houses around the cooling towers. Previously, cooling tower drift and condensate collected on the houses during the winter months rendering the fire hydrant system unservicable.

Safety Evaluation: This change does not affect the safety analysis. No hazards or considerations not described or implicit in the safety analysis report are posed. This system does not interface with any safety related systems nor is it required for a safe reactor shutdown.

DCR No. 889 Emergency Lube Oil Heater Temperature Indication Addition

Description of Change: The subject modifications replaced the previous 1"-90° elbow near the heater with a 1 - 1/2" x 1 - 1/2" x 1" tee and added a thermometer, a thermometer well and miscellaneous pipe fittings. The subject modifications to keep warm lube oil systems were carried out for both 3250 kW-Emergency Diesel Generator Sets.

Reason for Change: This DCR provided installation of temperature indicator (thermometer) on the inlet to the keep warm lube oil system heater. The thermometer verifies that the heater maintains the temperature of the keep warm (circulating) lube oil system between 130° to 135°F. The circulating oil temperature is maintained between these limits to permit rapid engine loading without waiting for warm up.

Safety Evaluation: Since the change basically provides an additional liquid filled thermometer on the inlet-piping to the lube oil heater which neither affects the routing of the piping nor the layout of the equipment:

a. The probability of an occurrence or the magnitude of the consequence of an accident or malfunction of the equipment has not increased. Rather, the temperature indication will add to the reliability of operation.

b. The possibility of an accident or malfunction of a different type from any evaluated previously in FSAR has not been created.

c. The margin of safety as defined in the basis of the technical specifications has not been reduced.

DCR No. 894 — Fuel Pool Cooling Pumps Handswitch 3410 A and B

Description of Change: Relocated handswitches and indicating lights of fuel pool cooling pumps from control panel 1C-84 to 1C-136.

Reason for Change: This change is to allow a single person to operate the fuel pool system from control panel 1C-136.

Safety Evaluation: The change does not affect or interface with any safety-related equipment and therefore does not present significant safety consideration.

DCR No. 909 — Containment Radiation Monitoring System

Description of Change: This DCR installed additional plant instrumentation to satisfy NUREG 0578 (Item 2.1.8.b) commitments for providing long-term monitoring capability of the radiation level inside the drywell and inside the torus chamber.

Reason for Change: Redundant channels of each monitored parameter are provided to comply with the requirements for Class 1E instrumentation found in applicable portions of NRC Regulatory Guide 1.97 and the IEEE standards listed in the design requirements section of this DCR. Each of two redundant monitor channels contains monitors, recorders, detectors, and associated cable and hardware for radiation detection with an instrument range of 10 R/h to 10^7R/h.

Safety Evaluation: The instrumentation installed by this DCR supplements existing plant (control room) indication of a specific plant parameter. Because the instruments are for indication only, no plant control functions are added or affected. Therefore, the passive nature of these instruments has no adverse effect on plant safety, and their installation does not constitute an unreviewed safety question.

DCR No. 914 Diesel Oil Storage Tank Cross-Tie

Description of Change: This DCR provides a method of transferrring oil from 1T-34 to 1T-35 and vice versa, and also provides a method of recirculationg oil to and from 1T-34.

Reason for Change: The purpose of this change is to reduce Technical Specification violations due to insufficient diesel fuel oil, and to provide sufficient oil storage capacity to allow oil to be ordered in large quantities.

Safety Evaluation: This change does not add significant safety hazards not described or implicit in the safety analysis report.

DCR No. 916 Baffle Assembly Retrofit for Condensate Demineralizer

Description of Change: The filter demineralizers in the condensate polisher system were modified to incorporate a new design of inlet baffle. New design dispersion baffles were provided as a retrofit kit by the original manufacturer.

Reason for Change: It was determined by Northern States Power Co. at the Prairie Island Station and confirmed by DeLaval that there was a design deficiency on the inlet baffles to the filter/demin. which causes turbulent flow during the precoat cycle. This turbulence causes scouring and subsequent washing away of the resin on the lower 6 to 12 inches of the septums. Scouring was observed through viewing windows installed on the filter/demin. vessel at the Prairie Island Station and further proved by the use of a model constructed by the DeLaval Co. A new baffle design consisting of multiple plates with offset orifices which was manufactured and installed in the vessel at the Prairie Island Station provided a uniform precoat over the total length of the septums.

Safety Evaluation: This change does not affect the safety analysis. No hazards or considerations not described or implicit in the safety analysis report are posed. This system does not interface with any safety related systems not required for a safe reactor shutdown.

DCR No. 928 Fire Protection System/Sprinkler System for Turbine Building Area Outside Emergency Diesel Generator Rooms Including Turbine Building Bay Area

Description of Change: To mitigate the consequences of any fire in the referenced area, an automatic wet pipe sprinkler system was installed.

Reason for Change: The turbine building area outside the emergency diesel generator rooms and including its bay area has oil lines, the hydrogen seal unit and, at times, contains oil drums and resin in plastic containers. These items are a fire hazard to the turbine building structure which can also impair access to the emergency diesel generator rooms.

Safety Evaluation: The sprinkler system is not required for the safe shutdown of the plant and the design change does not alter the original safety analysis. This system assures continued accessibility to the emergency diesel generator rooms.

DCR No. 932 A & B Post-Accident Sampling System (Safety-Related Items and Non-Safety-Related items)

Description of Change: The PASS was designed to enable an operator to obtain representative grab samples of reactor coolant, suppression pool liquid, and containment atmosphere for radiological and chemical analysis in association with a postulated LOCA at the DAEC. The system consists of a sample station, sample control panels, a sample piping station, a sample station exhaust fan, a cyclone separator rack, a refrigeration unit, and demineralized water, nitrogen, and tracer gas supplies.

Reason for Change: The post-accident sampling system (PASS) was installed at the DAEC to meet the requirements of NUREG 0737, Item II.B.3. A new sampling system had to be installed to meet those requirements because the existing reactor coolant and containment atmosphere sample panels were located inside the reactor building and may not have been accessible after an accident. Additionally, the existing reactor sample panel may not have provided a representative sample of post-accident reactor coolant, and may have unduly exposed an operator to high radiation during sampling.

Safety Evaluation: The change does not present any significant hazards or considerations not described or implicit in the Safety Analysis Report. The safety function of the containment atmosphere monitoring system, primary containment pressure boundary, diesel-backed power supply, standby gas treatment system, residual heat removal system, reactor vessel instrumentation, and reactor secondary containment was not degraded by this change. NRC approval of primary containment isolation valves was obtained via Technical Specification amendment. To determine that an unreviewed safety question did not exist, the design was reviewed to verify that it does not create a possibility for an accident or malfunction of a different type than evaluated previously in the FSAR or subsequent submittals. The evaluation found no unreviewed safety question.

DCR No. 932C Post-Accident Sampling Laboratory Facilities

Description of Change: The post-accident sampling laboratory for the DAEC is located in the administration building. There are two laboratories, a post-accident sampling counting laboratory and a post-accident sampling analysis laboratory.

Reason for Change: The post-accident sampling system(PASS) was installed at the DAEC to meet the requirements of NUREG 0737, Item II.B.3. A new sampling laboratory facility was installed to meet those requirements for onsite radiological and chemical analysis capability within 3 hours of initiating reactor coolant and containment atmosphere sampling procedures.

Safety Evaluation: The change does not present any significant hazards or considerations not described or implicit in the Safety Analysis Report. The equipment which was installed for analysis of post-accident samples does not perform a safety-related function.

DCR No. 963 RSCS Rod Group Memory Card

Description of Change: 195B9394AA Rod Group Memory Card as built drawing in 828E435AA Sh. No. 1 was changed to show VR1 zener diode as 1N4746A which is installed instead of 1N4733A.

Reason for Change: The change in documentation occurred because the actual components supplied and documented by GE contain the zener diode 1N4746A for performing the operation VR1, while on the corresponding drawing, the zener diode 1N4733A was shown instead. An update of the document was required.

Safety Evaluation: This change in documentation has no impact upon the safety on the system. Due caution was taken that the component was connected properly and that the performance specifications of the diode matched the mode of operation at VR1.

DCR No. 965 Fuel Support Grapple Modification

Description of Change: The recent fix, to correct RPV in-core sensor and fuel channel impacting, included the installation of core plugs in the bottom core support plate. In certain configurations, these plugs would mechanically inhibit the proper operation of the Fuel Support Grapple. To overcome this problem, a design modification was required to remove the existing slide rod protective legs and replace them with a single, non-interfering guard.

Reason for Change: The fuel support grapple was modified to accommodate the removal of the fuel support plate after the insertion of the bottom core support plate plugs.

Safety Evaluation: This change is not safety related and does not affect the safety system. The fuel support grapple moves only the fuel support plate. It does not move fuel.

DCR No. 969 161 kV Substation Microwave System Modification

Description of Change: The Microwave system from Wellsburg to the IE Tower was upgraded and its path revised. Listed below are the revisions at the DAEC Substation Control House:

1) Install a Microwave Repeater Station including the wave guide.
2) Install a system of Tone Equipment on the Dysart-Vinton Line to interface with the Microwave equipment.
3) Remove the Carrier Equipment for the Dysart-Vinton Line.

Reason for Change: The design change upgraded the communications link from DAEC to the Vinton and Dysart substations. At the DAEC this changed the communications link for OCB7510 and OCB3110 from Carrier to Microwave. This change brought the level of quality up to that of the rest of the 161 kV substation which uses microwave as the communications link to other substations.

Safety Evaluation: The subject changes do not affect the margin of safety.

DCR No. 971 Reactor Water Clean Up System Instruments Modification

Description of Change: The capillary lines between the instruments and their respective root valves were replaced by seamless 316 stainless steel instrument tubing. This allows calibration and maintenance of the instruments in a low radiation area.

Reason for Change: The previous capillary system was a high maintenance item and resulted in high radiation exposure levels for the DAEC maintenance personnel.

Safety Evaluation: The change of the instrument lines had no impact upon the safety of the RWCU system. Instead, this change improves the operation of the instruments and it significantly decreases the radiation level exposure for the DAEC maintenance personnel. The replacement items were nonsafety-related and did not affect the safe shutdown of the plant.

DCR No. 989 Main Steam SRVDL Piping Support Modification

Description of Change: The work involved the design and installation of piping support modifications to the main steam safety relief valve discharge lines which vent to the torus, and change-out of four Main Steam Line snubbers.

Reason for Change: The support modifications are the consequences of an analysis of the SRV discharge piping subjected to new design thrust loads as defined under the Mark I Containment program. In order to maintain the original design margins of safety, the analysis of the SRV discharge piping for these new thrust loads demonstrated the need to add additional dynamic restraints (snubbers) and to replace some existing snubbers with ones of larger capacity.

Safety Evaluation: Adding larger and additional snubbers increases the ability of the SRV discharge and Main Steam Line piping to withstand the discharge thrust loads. The larger capacity and additional snubbers being used are identical to snubbers existing on the piping system. Operation of the reactor pressure relief system is unchanged. The margin of safety for dynamic loads on the SRV discharge piping is not reduced. Operation of the reactor pressure main steam system is unchanged. The margin of safety for dynamic loads on the main steam piping is not reduced.

DCR No. 994 Turbine Building Ventilation Exhaust Modification (non-safety-related)

Description of Change: The modification replaced the existing roof exhausters and combined the exhausts into a common header which allows use of one radiation monitor instead of eight samplers as previously designed. This ductwork runs across the turbine building roof and onto the reactor building roof where three 50%-capacity, two-speed, vane axial fans are located. Each of the fans is provided with a motor-operated exhaust isolation damper and a backdraft discharge damper. The three fans are enclosed in a weatherproof penthouse.

Reason for Change: The turbine building ventilation exhaust modification was made to comply with the requirements of Item II.F.1, Attachment 1 of NUREG 0737 (Noble Gas Effluent Monitor).

Safety Evaluation: The turbine building ventilation exhaust modification installed by this DCR is not safety-related and presents no unreviewed safety question. The modification replaced the existing nonsafety-related roof exhausters and combined the exhausts into a common ductwork to allow radiation monitoring prior to discharge. The ductwork supports and fans are mounted and supported in such a way that the margin of safety inherent in the turbine building and reactor building structures is not affected. All electrical/control changes associated with this DCR are nonsafety-related and any failure which may occur will have no consequence on any safety-related system.

DCR No. 998 Reactor Building Laundry, Laundry Storage Areas, and Turbine Building Electropolish Area.

Description of Change: This DCR relocated the laundry machines in the Reactor Building (ELEV 812'-0") south of the equipment hatch, made provision to store the dirty laundry on the north side of the equipment hatch (Reactor Building ELEV 812'-0"), and changed an area on the Turbine Building operating floor elevation 780'-0" into a permanent decontamination facility (electropolish area). All these areas are now fully enclosed except the laundry machine area.

Reason for Change: Previously, the area on the north side of the equipment hatch (in Reactor Building @ ELEV 812'-0") had two laundry machines with temporary enclosures. This area was congested for the personnel doing the laundry. Contamined clothes were stored near the laundry machine area. There was no permanent facility to decontaminate tools and equipment. This situation created an ALARA concern for the people working with the contaminated clothes, tools and equipment.

Safety Evaluation: This design change does not create an unreviewed safety question. The decontamination facilities created controlled environments for cleaning of laundry and tools. This improves working conditions from an ALARA aspect for personnel assigned to decontamination.

DCR No. 1009 Scram Discharge Volume Isolation Vent and Drain Valves

Description of Change: This design change added redundant vent and drain valves and the associated solenoid valves to the Scram Discharge System. Also, four new pipe supports were added and two existing supports modified. Three of the new supports were located in the vent line and one in the drain line.

Reason for Change: The basis for this change was the partial failure to scram incident at TVA's Brown's Ferry 3 and the subsequent generic Safety Evaluation Report by the NRC, dated December 9, 1980.

Safety Evaluation: The conclusion of the safety evaluation is that this design change did not involve any unreviewed safety questions on the following basis:

A. A single active failure (fail to close) of either the vent or drain valve, with the previous system arrangement, would result in a blowdown of the Reactor Coolant System outside primary containment during a reactor scram. With the addition of the redundant valves, either both vent valves or both drain valves must fail simultaneously to breech primary containment, thus satisfying a single active failure. Therefore, the probability of occurrence of an accident or malfunction of equipment important to safety and previously evaluated in the FSAR will be decreased.

B. The addition of these valves increases the probability of not being able to reopen and drain the SDV following scram reset. Also, with a partial scram originally, it may not be possible to complete the scram. However, this design change does not represent an unreviewed safety question nor does it compromise safety because this postulated accident requires two separate failures. Additional support is found in Safety Criterion 2 of the NRC generic Safety Evaluation Report, dated December 9, 1980 which states "No single active failure shall prevent uncontrolled loss of reactor coolant."

C. The integrity of the SDV System is maintained with these modifications. The integrity of the system with respect to seismic and thermal design has been verified by analysis. The pressure boundary integrity has been verified by the hydro test and the system operational integrity has been verified by the functional test in the EAR. NRC approval was received via Technical Specification amendment prior to implementation.

DCR No. 1013 Scram Discharge Volume Reconnect Instrument Lines

Description of Change: Routing of the scram discharge volume instrument level piping was changed so that the instruments are connected directly to the instrument volume. Other piping was added to eliminate the sharing of instrument sensing lines between instruments.

Reason for Change: NRC concern over scram discharge volume problems at Browns Ferry, Brunswick, and Hatch plants mandated these changes.

Safety Evaluation: The design change does not involve an unreviewed safety question and increases the reliability of the system. The change will eliminate the possibility of a plugged sensing line affecting more than one instrument.

DCR No. 1014 Scram Discharge Volume Computer Tie-In

Description of Change: This design change provided separate computer logging of the status of the SDV (Scram Discharge Volume) and rod block level switches as well as the SDV drain and vent control valves.

Reason for Change: This design change was initiated in order to meet the NRC criterion "E" as stated in letter (IELP to NRC) #LDR-81-28, dated January 14, 1981.

Safety Evaluation: The design change does not constitute any change in the DAEC Technical Specifications or an unreviewed safety question. The purpose of this design change is to provide additional information to control room personnel regarding the sequence of events and status changes of the SDV level switches, rod block level switch, and SDV vent and drain valves during and shortly after a scram. The contacts from the relays are considered to be isolation devices between the RPS and the computer. Any failure mode and/or malfunction of the relays or computer will not jeopardize the safety of the RPS. Moreover, the indicating lights do not participate in the operation and function of the RPS.

DCR No. 1015 Reactor Building Sample Hood Station Shield Wall

Description of Change: A permanent shield wall was built around the sample hood area to reduce the radiation levels. The shield wall is constructed of concrete blocks with all cells filled with grout. Viewing windows are furnished to allow viewing of the sample sink and gauge panel. An access gate is provided to control access to this area.

Reason for Change: The radiation levels at the Reactor Building Sample Hood Station were 100 mr/hr and increasing. This created a significant radiation field (>25 mr/hr) around the normal access door to the radwaste facilities and the surrounding area. Plant personnel were, thus, routinely and repeatedly exposed to significant amounts of radiation.

Safety Evaluation: The shield wall should remain structurally sound during a seismic event. The shield wall is not in the proximity of safety-related equipment, the closest being MO-2000, 'B' Loop Containment Spray Injection Valve, which is located approximately 20 feet east of the wall. Therefore, if the wall would fail and fall down during a seismic event, no safety-related equipment would be damaged. The loading of the floor by the shield wall will have no significant impact on the integrity of this structure during a seismic event.

The Design Change does not constitute an unreviewed safety question. The Design Change does not increase the probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report.

DCR No. 1019 Closed Circuit "TV" (CCTV) Work Platforms

Description of Change: Work platforms, ladders and safety hooks were constructed and installed as detailed in the change package. This change applies to exterior wall mounted CCTVs only.

Reason for Change: This change was made to increase personnel safety.

Safety Evaluation: The addition of the platforms does not affect the safety considerations of any plant system.

DCR No. 1025 In-Plant Safety Relief Valve Discharge Test Instrumentation Installation

Description of Change: The work involved the specification and installation of various pressure, strain, and acceleration measurement devices internal and external to the torus and on a SRV discharge line.

Reason for Change: The test devices and associated instrumentation is utilized to assess the structural response of the torus shell and torus support system during safety relief valve actuation. This analysis is required due to newly defined hydrodynamic submerged structure loads which have been refined since the time of original design.

Safety Evaluation: This installation does not alter the suppression chamber or supports system. The test instrumentation is passive and thus, has no potential for creation of a new accident. The test instrumentation is installed such that the structural capacity of the torus, piping and supports will not be degraded. All installation and testing shall be performed within the limits as specified in the Technical Specifications.

DCR No. 1027 Replacement of Contact Blocks And Test Switches For Temperature Indicating Switches In The Steam Leak Detection System

Description of Change: The existing "silver coated contacts" were replaced with new "gold flashed contacts" for better plant operation and to avoid undesired trips while in "normal" position. Test switches were also replaced.

Reason for Change: Existing test switches had a design problem in that they caused the temperature indicating switches to be unreliable. Their reliability was questionable because they had very poor repeatability in "test" position and had caused spurious trips while in the "normal" position.

Safety Evaluation: The replacement of existing "silver coated contacts" with the "gold flashed contacts" does not change the intended function and configuration of the test switches. This change does not degrade the safety of this system or other systems present in the area.

The replacements are non-safety switches and are justified based on three reasons. The first reason is the unavailability of qualified switches and the intent to have the replacements qualified. The second reason is the mild environment in which the switches are required to function. This is based on the Environmental Qualification Procedure 11186-234-NP-1, Specification 7884-M-411A, and the DAEC semiannual Report on the Environmental Qualification Program for safety related electrical equipment (July 15, 1982). Bechtel Licensing Information News Letter #82-2, dated June 13, 1982, states that qualification testing for mild environment is not required. Only QA requirements need to be met. The third and final reason is that the Failure Mode and Effects Analysis (FMEA) table, verifies that the switches are fail safe.

This change will decrease the probability of a malfunction when compared to the existing equipment. The possibility of equipment failure may be greater than that of safety related equipment. However, there is no safety related equipment available and this design is "fail safe". This satisfies the requirements of FSAR Section 7. There is no increase in the consequences or type of accidents. The margin of safety at the DAEC is not reduced. Therefore, no unreviewed safety questions result from this change.

DCR No. 1029 RHR and RHRSW Relief Valve Flange Installation

Description of Change: Flanges were installed on relief valves PSV-1988, -2068, -1952, and -2043 to facilitate removal of these valves from respective piping for testing per ASME Code XI ISI Requirements.

Reason for Change: Removal for testing may be required as often as once a year. At present, the RHR heat exchangers and connected piping must be hydrostatically tested in order to check the valve's setpoints.

Safety Evaluation: The flanges are installed in safety related piping. However, the addition of the flanges will not affect the operation of any system. The weight added to GBB piping will have negligible effect on the seismic characteristics of the system. Valves PSV-1952 and PSV-2043 weigh about 420 pounds each and the total weight added by the flanges is about 32 pounds to each piping system. The weight of the flanges added to GBC piping containing valves PSV-1988 and PSV-2068 is not negligible. By inspection the configuration of piping containing PSV-1988 has the worst geometry for stresses. But calculation IELP-M81-25 "RHRSW Relief Valve Piping Seismic Calculation" concludes that maximum stresses will not be increased beyond any allowables. This indicates that the other system will have even less of a change in stress values. The seismic integrity of these systems will not be jeopardized.

The installation of this DCR did not present an unreviewed safety question. Specifically: 1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased; 2) the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created, and; 3) the margin of safety as defined in the basis of the Technical Specification is not reduced.

DCR No. 1030 RWCU Flow Control Valve CV-2729 Repair

Description of Change: This DCR provided the guidance and engineering recommendations for repair of CV-2729 in order to comply with Code requirements.

Reason for Change: Reactor water clean-up blowdown flow control valve CV-2729 developed a leak in the valve body and was temporarily repair welded. The cause of the eroded valve was determined to be cavitation during blowdown to the condenser during normal plant operation.

Safety Evaluation: The change is not safety related nor does it impact a safety-related system. In addition, the work area does not contain safety related components. The RWCU system was designed and constructed to meet the requirements for nuclear piping system class 3, ANSI B31.7. The repair to CV-2729 was performed to the original design and construction requirements and therefore does not reduce the overall quality of the system.

DCR No. 1031 Spare Cables to Off-Gas Stack for Future Use

Description of Change: Spare cables were installed between the reactor building and the off-gas stack for undetermined future needs.

Reason for Change: DCR 956B required addition of wires from the reactor building to the off-gas stack. Engineering and Maintenance see future requirements for additional cables to the stack and desired that spares be pulled with these cables in DCR 956B.

Safety Evaluation: The installation is of a non-safety nature. It is safety related only in that the routing uses safety related raceway. This is permissible per the E-512 standard. This DCR does not affect the safety systems in a degrading manner.

DCR No. 1034 Radwaste Solidification Process Fill Head and Shield Plug Lift Points and Modifications

Description of Change: Modifications done as a part of this DCR include installation of lift points for the process fill head and shield plug. erection of a steel enclosure on a foundation for the Hittman cement hopper outside, core drilling for a cement feed line from the cement hopper to the process fill head and laying a set of tracks to guide the process shield from the processing station to the loading area under the hatch.

Reason for Change: Beginning July 1, 1981, the three radwaste burial sites in the country required that radwaste be solidified to be buried. Hittman Corporation did the solidification at the DAEC with a temporary cement system. Various modifications were required on the DAEC to interface with the Hittman equipment.

Safety Evaluation: None of the changes made per this DCR have any affect on the safety of the plant. This system is not safety related. The evaluation of the Hittman solidification system was done separately prior to operation.

DCR No. 1036 Reactor Recirculation System Suction, Discharge and By-Pass Valves' Packing Leakoff Lines

Description of Change: The packing leakoff lines and isolation valves from the recirculation system suction, discharge, and bypass valves were removed. The common header was also removed, the piping upstream of the removed isolation valves was capped. All hangers and their components which were supporting the downstream piping were removed.

Reason for Change: The packing leakoff lines' isolation valves for the recirculation system suction and discharge valves leaked through. The leakoff lines drained into a common header shared by the by-pass valves' leakoff lines.

Since these lines were all seismic, it was decided to remove the leakoff lines and the common header.

Safety Evaluation: A 10CFR50.59 review was done. The proposed change does not involve an unreviewed safety question, nor does it involve a change in Tech. Specs.

DCR No. 1037 Re-Roofing of the Turbine Building

Description of Change: Roofs on the control building, administration building, off-gas recombiner building and the turbine building were replaced with roofing meeting Underwriter's Laboratories, Inc. - Class A, Factory Mutual - Class I requirements.

Reason for Change: The roofs on the subject buildings had deteriorated to a point where economical repair was not feasible.

Safety Evaluation: This change does not affect the safety analysis. No hazard or consideration not described or implicit in the safety analysis report is posed. This system does not interface with any safety related systems nor is it required for a safe shutdown. Reroofing of the turbine building roof improves watertightness capabilities of existing built-up roofing and thereby increases plant reliability.

DCR No. 1043 Ventilation for Electro Polisher Area

Description of Change: 4200 CFM of cooling air was supplied to the electropolisher room by bringing a 36" x 20" branch duct off of the 5' x 10' supply duct in the electropolisher room. This branch duct is located at the approximate center of the room with a bottom diffuser.

Reason for Change: The electropolish decontamination facility is now located in the northwest corner of the turbine building at elevation 780'-0". This area is enclosed and very warm. The working environment during plant operations was approximately 90°F.

Safety Evaluation: This design change does not change the original intended function of the turbine building ventilation system because the same air as was previously used is now being redistributed. This air which is being redistributed is going into the electropolish area. The electropolish area is totally enclosed and has the potential for becoming airborne. For this reason, filters have been provided for the exhausting air. In addition, the supply air can be shut off by manually closing the supply/balancing damper. These precautions have been taken to limit to a minimum any significant release out of the electropolish area. Once out of the electropolish area, the turbine building exhaust system monitors all releases. This design change does not involve a change in the Technical Specification and does not present an unreviewed safety question.

DCR No. 1044 <u>RPS MG Protective Circuit Upgrade</u>

<u>Description of Change:</u> The modifications to the DAEC reactor protection system power supply protective circuitry consisted of the addition of two identical Class 1E "Electrical Protection Assemblies (EPA's)" in series between each motor generator set and its respective reactor protection system bus, and between the alternate power source and the reactor protection system buses, providing redundant protection. Each EPA is composed of two basic sub-systems:

1) A General Electric type TFJ-175A circuit breaker with companion TFK undervoltage release.

2) Electrical Protection Logic Printed circuit card, which disconnects the circuitry from input power whenever voltage or frequency exceeds their normal tolerances to be set per Technical Specification commitment (DCR index item 5.01).

<u>Reason for Change:</u> The NRC by their letter of 8/7/78 (Ippolito to Arnold) expressed a concern about the capability of the Class 1E reactor protection system and other Class 1E systems and components powered by the reactor protection system power supplies to accommodate the effects of possible sustained abnormal voltage or frequency conditions from the non-class 1E reactor protection system power supplies (RPS MG sets). These abnormal conditions could be caused by possible though unlikely combinations of undetectable single failures or by the effects of earthquakes, and could result in damage to the class 1E systems and components powered by the reactor protection system power supplies (RPS MG sets) with the attendant potential loss of capability to perform their intended safety function.

<u>Safety Evaluation:</u> Installation of two identical Class 1E "Electrical Protection Assemblies" (EPA's) in series between each motor generator set and its respective reactor protection system bus, and between the alternate power source and the reactor protection system buses will protect the RPS and associated downstream components from abnormal overvoltage, undervoltage and underfrequency conditions. The proposed modification has been conceptually approved by the

NRC (Reference: NRC's letter dated 2/23/79, Boyd to Sherwood of G. E., DCR index item 10.08).

All realistic failure modes and/or malfunctions were considered and protection provided commensurate with the potential consequences. This modification complies with all applicable regulatory requirements (Reference: DRL, DCR index item 1.01), including Technical Specifications to be proposed, so that the change does not represent an "unreviewed safety question." Also the margin of safety as defined in the bases of the existing and the proposed Technical Specifications, will not be reduced by this change.

Justification that the proposed design modification and EPA's meet the requirements of GDC 2 and GDC 21 of 10CFR Part 50, Appendix "A" is provided in the IELP letter to the NRC (LDR-82-002, dated 1/6/82, DCR Index Item 10.02). Assessment of EPA qualification compliance with NUREG-0588 category 1, IEEE 323-1974 and IEEE 344-1975 is listed in APED-C71-034-NI (DCR index item 6.08).

In conclusion, the proposed modification complies with the safety evaluation requirements of 10CFR50.59 for the reasons stated below:

1) The change to the RPS protective circuit enhances the safety of the RPS and the DAEC.

2) This DCR complete with the written safety evaluation provides records of changes made under the authority of 10CFR50.59, paragraph (a)(1).

3) A Technical Specifications amendment was submitted to the NRC and approved.

DCR No. 1046* Sample Line for the Floor Drain Collector Tank 1T-73 Used in the Liquid Radwaste System

Description of Change: The sample return line was modified to replace a plugged sample line. To prevent plugging of the new sample line, the condensate service water line was modified and added to provide a back-flush to the new sample line.

Reason for Change: The sample line for the floor drain collector tank from the pump discharge to the monitoring rack became plugged. The sample line was designed to provide a representative liquid sample from the floor drain collector tank.

Safety Evaluation: This change does not affect the DAEC Final Safety Analysis Report. No hazards or considerations not described or implicit in the safety analysis report are created by this design. This design change does not interface with any safety related system nor is it required for a safe shutdown. Modifying the previous sample return line of the floor drain collector tank improves the sampling capability of the plant and thereby improves the liquid radwaste processing of the plant.

DCR No. 1048 Reactor Pressure Vessel Water Level Indicator Scales Common Reference Labeling

Description of Change: This change involved minor changes to the indication devices of the fuel zone level instruments (wide-range Yarway instruments and the placement of marker plates adjacent to all the other indicators. This change involved the addition of two "TAF" (top of active fuel) red lines and corresponding marker plates. The upper red line (the current "0") is identified as the TAF when reading the indicator under calibration conditions (0 psig, cold). Another marker plate identifies the lower red line as the TAF when reading the indicator under high pressure (1000 psig) and corresponding temperature conditions in the vessel. This change also involved the addition of marker plates located in the immediate vicinity of the other indicators as an aid in referencing the indicated water level to TAF.

Reason for Change: This change was implemented to comply with NUREG 0737, Item II.K.3.27. Different reference points utilized for the various reactor vessel water level instruments in the control room may cause operator confusion during times of stress (accident conditions).

Safety Evaluation: A Technical Specification amendment was requested and received from the NRC. This design change only involved the addition of labels to existing reactor vessel indicators. This change provides clarification of the information already provided by the indicators. As such, this change enhances the safe shutdown capability of the plant.

DCR No. 1062 RHR Service Water Flow Indicators

Description of Change: Flow indicators FI1944 and FI2050 were updated to get more accurate indication between 2000 gpm and 5000 gpm by replacing new scales (linear) for indication and adding square-root converters.

Reason for Change: The old scales on these indicators were 0 to 8000 gpm on a log scale. When both pumps were running at full load, the flow was 4800 gpm and approximately midrange on the old indicator. The surveillance test procedure requires each pump to provide a flow rate $\geq$ 2400 gpm with $\geq$ 264 psig discharge pressure. Since 2400 gpm was the lower portion of the scale on these FIs, it was very difficult to get an accurate reading.

Safety Evaluation: This design change is not safety related. Implementation of this design change allows easier reading of RHR service water flow by the plant operators. As such, this design change enhances safe plant operation. No unresolved safety questions result from implementation of this DCR.

DCR No. 1063 Fire Protection/Well Water Cross Connection Tie

Description of Change: This design change installed the necessary valves and piping to cross-connect the fire protection and well water systems. The valves and piping are housed in a manhole constructed by this DCR. Electrical outlets (115V) are provided in the installed manhole.

Reason for Change: Periodic draining of the Circulation Water pit is required to perform scheduled maintenance. When the Circulation Pit is drained, the station fire pumps did not have a water source.

Safety Evaluation: This design change is not safety-related. It does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. It does not create the possibility of an accident or malfunction of a different type than previously evaluated in the FSAR. It does not reduce the margin of safety as defined in the basis of Technical Specifications.

DCR No. 1064 Change the Operating Range of PDS-4304 and PDS-4305

Description of Change: The bellows unit assemblies of PDS-4304 and PDS-4305 were replaced to reduce the operating range of these pressure differential switches. These switches control the torus to reactor building vacuum breaker.

Reason for Change: The reduced range of operation increased the setpoint repeatability which decreased the amount of setpoint drift which these switches experienced. The reduced range also decreased the switch reset deadband which reduced the amount of time the vacuum breakers stay open after the design function has been performed.

Safety Evaluation: The bellow unit assemblies installed by this modification are identical to the original bellow unit assemblies except for the range of operation. This design change does not affect the design functions of or the mechanisms of operation for the subject switches. Therefore, this design change does not, with respect to the FSAR or subsequent submittals:

1. Create a possibility of an accident or malfunction of a type different than previously evaluated.
2. Increase the probability of the occurrence of an accident or malfunction of equipment previously analyzed.
3. Increase the consequences of any accident or malfunction of equipment previously analyzed.

DCR No. 1070 Reactor Cavity Drain Line Replacement

Description of Change: Present piping was replaced with new stainless steel piping, sloped continuously downward toward the reactor building equipment.

Reason for Change: The Reactor Cavity Drain Line had, through the life of the plant, become internally contaminated and corroded to the point that access to areas around the piping had become restricted. Temporary shielding and wire cages had been installed to control radiation and access to these areas. The situation had deteriorated to the point that these solutions had become ineffective. Some piping reached as high as 500mr/hr on contact. The installed configuration of the drain line was inadequate as well. Over 75% of the drain line was horizontal. The remainder was sloped in the wrong direction, which added to the crud and corrosion buildup.

Safety Evaluation: The Cavity Drain Line is not safety-related nor seismic Class One. Replacement of the older pipe in no way involved an unreviewed safety question and did not prevent or interfere with the safe and orderly shutdown of the plant. The piping does not traverse any seismic equipment and therefore, need not be seismicly supported. The piping does not interface with any CSCS Systems. Installation of this DCR in no way created a possibility for an accident of a different type than previously analyzed. It did not increase the probability of an accident nor increase the consequences of any accident which had been previously evaluated. Installation of this DCR did not affect or require a Technical Specification change.

DCR No. 1074 Install Vent Openings in Cabinet 1C29

Description of Change: 16" x 10" ventilation openings were added to 1C29 cabinet doors.

Reason for Change. The malfunction rate of recorders in panel 1C29 was very high and resulted in inaccurate indications and excessive maintenance. These recorders, for the most part, are safety related. This failure rate was attributed to recorder temperatures which were running in excess of operating limits specified by the recorder manufacturer. To correct this condition a 16" x 10" ventilation opening was placed in the top and bottom of each cabinet door.

Safety Evaluation: Cabinet 1C29 is designed to meet DAEC seismic criteria. However, it was determined that the cabinet doors were not a consideration in the original assumption for design of cabinet integrity to withstand a seismic event. The cabinet doors, therefore, were chosen for the location of the ventilation openings. This modification did not increase the probability of occurrence or the magnitude of consequences previously evaluated in the FSAR. In addition, this modification did not create the possibility of an accident or malfunction of a different type than previously evaluated or reduce the margin of safety as defined in the basis of any Technical Specification.

DCR No. 1076 Equipment and Floor Drain Sump Level Probes

Description of Change: The existing electrodes were replaced with 316 stainless steel solid rod electrodes for several level switches. The solid rod electrodes are insulated with Dupont Teflon TFE.

Reason for Change: The original PVC electrode holders and PVC insulated suspension wires in the sump pits deteriorated because of the high temperatures and contaminations present in the drain sumps. There was also a problem with sludge deposits between the electrode and its shield which prevented the electrode from generating its intended signals.

Safety Evaluation: The change does not involve a change in Technical Specification or an unreviewed safety question. The purpose of this change is to provide increased assurance concerning the intended function of level sensors.

DCR No. 1079 Replace Two Differential Pressure Switches

Description of Change: This DCR replaced an existing differential pressure switch on each Standby Filter Unit (SFU) with a new differential pressure switch. Each new DP switch has the same range and setpoint as the existing switch. The reset deadband of each new DP switch is narrower (.01 inches WC versus .04 inches WC for each existing DP switch.)

Reason for Change: Existing DP switches tripped on low SFU air flow but did not reset at normal SFU air flow. The narrower reset deadband of the new DP switches enables each DP switch to reset at the normal air flow for each SFU.

Safety Evaluation: This DCR replaced two existing DP switches with improved DP switches. The DCR enhanced Standby Filter Unit performance, thus improving the safety of the SFU System. This DCR had no impact on the FSAR, Technical Specifications or Fire Hazards Analysis.

The safety function of the DP switches is to alarm low air flow through Standby Filter Unit and to inhibit the SFU electric heater operation during low air flow. The heater dries the SFU air before it enters the SFU charcoal filters. Failure of these switches would impair charcoal filter performance and possibly impact control room habitability during SFU operation.

The previous DPSs did not properly reset after a low air flow condition in SFU. No air flow is the normal condition of both SFUs since both are normally out of service when the control room HVAC is functioning properly. Therefore, when the SFU was placed in service, its electric heater would not function because the DPS "stuck" in its alarm position. The new DPSs properly reset during the SFUs operation so that the SFU heater can perform its safety function. The safety of each SFU is thereby enhanced by the new differential pressure switches installed by this DCR.

The new differential pressure switches decrease the probability of an accident, introduce no new possibility of an accident, and preserve existing safety margins of the Standby Filter Unit System.

DCR No. 1082 Snubber Replacement

Description of Change: Five International Nuclear Safeguard Corp. (INC) Snubbers which served the Main Steam Isolation

Valve Leakage Control System in the steam tunnel were replaced by ones of comparable size manufactured by Pacific Scientific.

Reason for Change: During a functional test, four of the five INC snubbers remaining in service at the DAEC were found to be within 3 percent of their design limit for breakaway friction during retraction. Replacing the snubbers greatly reduced the possibility of snubber failure.

Safety Evaluation: This design change replaced five snubbers because the previous snubbers had deteriorated to the point where the breakaway friction limit had nearly been reached. The safety function of the snubbers was increased by this design change since the probability of the snubbers failing the breakaway friction limit was reduced. The new snubbers are smiliar in all respects to the previous snubbers. The change in weight is insignificant compared to pipe weight, the length is identical, and the principal of operation for the old and new snubbers is the same. Also, the maximum acceleration which the snubber allows the pipe to see is within the allowable limits. Although the upper breakaway friction limits differ between the old and new, the difference is insignificant. Therefore, the magnitude of the consequences of an accident or malfunction of equipment important to safety, previously evaluated in the FSAR, did not increase because this design change did not compromise the present ability to mitigate the consequences of an accident or malfunction. The result of the safety evaluation is that this design change did not represent an unreviewed safety question.

DCR No. 1085 Core Spray Discharge Flow Switches

Description of Change: New flow switches, environmentally and seismically qualified to comply with NRC IE Bulletin 79-01B and its supplements, were installed to replace the existing switches.

Reason for Change: Previous core spray discharge flow switches did not comply with NRC IE Bulletin 79-01B and its supplements.

Safety Evaluation: The replacement of these switches upgrades the instrumentation and has no adverse effects on plant safety. The effect of the above modifications on control board seismic analysis was considered. The maximum weight gain per panel due to this switch replacement is judged negligible compared to overall panel weight and will not affect the seismic analysis results for the panels.

DCR No. 1087 Security Lighting Additions

Description of Change: This DCR covers the installation of additional lighting fixtures to bring lighting levels up to or above the 0.2 foot-candle minimum. The affected areas were identified in Lighting Survey Results. Fixtures added in the area of main power block (Turbine Building, Reactor Building, etc.) are high pressure sodium wall pack fixtures. Fixtures added in outlying areas such as the pumphouse, warehouse, etc. are mercury vapor 400w floodlights or high pressure sodium wall pack fixtures.

Reason for Change: DAEC Security conducted light level measurements within the protected area perimeter on October 6, 7 and 8, 1981. The same area was re-surveyed on November 19,1981. Several areas were found where illumination levels fell below the allowed level of 0.2 foot-candles.

Safety Evaluation: The security lighting system is non-safety related. It has no effect on nor relation to plant safety systems.

DCR No. 1089 Control Room Emergency Lights

Description of Change: The control room ceiling lights were modified to use Bodine emergency DC power ballasts. Indicator light is provided for monitoring the charger and battery and a test switch for testing the charge in the battery. This change increased the minimum emergency lighting criteria for the control room to ten foot-candles.

Reason for Change: Several emergency lights in the control room were not in working condition. There was a history of the lights not working when an emergency occurred. Also there had been three different cases of fires breaking out inside the original lights.

Safety Evaluation: The control room emergency lights do not perform a safety related function. Implementation of this design change provided better lighting in the control room in the case of an AC power failure and as such this design change enhanced safe plant operation. No unresolved safety question resulted from implementation of this design change.

DCR No. 1090 Recirculation Pump Motor Oil Level Alarm

Description of Change: Wiring to the oil level alarm contacts was separated. Four relays and four indicating lights to indicate hi and low level on upper and lower bearing for each pump motor were installed in a separate control panel. The alarm is activated on Panel 1C04.

Reason for Change: Each recirculaton pump motor at the DAEC had two oil level switches, one for upper bearing and one for lower bearing. Each oil level switch had two contacts (high/low). All four contacts on each pump motor were connected in parallel and shared one common annunciator window on panel 1C04. When an oil level alarm occurred it was not possible to identify which bearing was involved and whether the indication was high or low oil level.

Safety Evaluation: This modification was non-safety related. Separation of wiring of the oil level alarm contacts in the recirculation pump motor instrument terminal boxes located inside the drywell and the installation of relays and indicating lights in the control panel located in the reactor building assists the operator in identifying which bearing is involved and whether the indication is high or low oil level. The routing of the non-safety related cables through safety related trays was acceptable per the DAEC Standard. This modification did not require a change to the FSAR or to the Technical Specifications and does not affect the safe shutdown of the reactor.

DCR No. 1091 Dryer/Separator Pool Seal Modification

Description of Change: A dryer/separator pool seal has been designed and is installed on the pool side of the shield blocks. The pool seal is designed to prevent leakage of water between the dryer/separator storage pool and the reactor cavity when the storage pool is flooded and the reactor cavity is drained for servicing.

Reason for Change: During a refueling outage, the dryer and separator of the reactor vessel assembly are placed in the dryer/separator pool. It is then necessary to conduct personnel entry to the reactor vessel flange level. An ALARA concern existed due to the radiation and the airborne radioactive contamination given forth from the dryer and separator. With the dryer/separator storage pool filled and the dryer and separator covered with water, the water acts as a shield and reduces the radiation exposure for the people on the refuel floor supervising the work in the reactor cavity. The water also eliminates the airborne contamination from the dryer and separator. The shield blocks of the dryer/separator storage pool were not designed to provide a waterproof seal between the dryer/separator pool and the reactor cavity. In order to allow underwater storage of the dryer and separator with the reactor cavity drained for personnel entry to the reactor vessel flange level, a positive seal was provided between the dryer/separator storage pool and the reactor cavity.

Safety Evaluation: No unreviewed safety question is involved because:

1. The dryer/separator pool shield blocks are structurally secure. No increase in the probability of occurrence or magnitude of the consequences of an FSAR evaluated accident is expected.

2. The dryer/separator pool seal is structurally secure in its storage mode, transit mode and operational mode. No new type of accident is expected because of this design change.

3. The margin of safety of any equipment in the area has not been reduced. The pool seal is not installed near any safety related equipment.

The adequacy of the strength of the dryer/separator pool shield blocks has been checked by calculation (Index Item 5.02) and it is shown that, under the pool water hydraulic load and the seismic load, the stresses on the shield blocks are below allowable limits.

DCR No. 1092 Outage Air Compressor Permanent Power

Description of Change: During the course of DCR review, it was concluded that the permanent power supply for the outage air compressor should not be fed directly from the security transformer, but from the security 480V Motor Control Center. A 4-inch spare conduit was routed from the MCC. A 300 amp circuit breaker was installed in the cubicles. 350 MCM triplex cable was routed through existing 4-inch conduit and additional 4-inch conduit was installed to the location of the air compressor. A 120VAC power source was made available where the air compressor was stationed for control power use. An alarm unit was installed inside the Reactor Building by the drywell entrance for the annunciation of low pressure at the air compressor.

Reason for Change: The outage air compressors previously had a temporary supply. The temporary 480V power was fed from the security transformer on the N.W. side of the plant with a 350 MCM triplex cable that was routed along the ground and over the roofs of the hallway between the Security and Administration Building and the Radwaste Building. A letter from the DAEC Plant Superintendent to the Manager of Design Engineering indicated that this temporary power cable was vulnerable to damage and posed a fire and personnel hazard. The letter requested that Design Engineering review the need for the installation of a permanent power supply for the outage air compressors.

Safety Evaluation: The change is non-safety related. Implementation of this DCR eliminated the vulnerability of the power cable to damage and the possibility of personnel and fire hazards. There were no unreviewed safety questions as aresult of this design change.

DCR No. 1093 Modifications to Auxiliary Boiler Room Fire Protection

Description of Change: Sprinkler System Number 5 was extended beneath the auxiliary boiler's exhaust duct to provide fuel oil pump 1P-53 with adequate water coverage. Also, a fire curb was installed in each of the two doorways to the Auxiliary Boiler Room. Both of these modifications are in accordance with ANI recommendations.

Reason for Change: Sprinkler System Number 5, as previously installed in Auxiliary Boiler Room, did not provide adequate water coverage of the fuel oil pump due to obstruction from the auxiliary boiler exhaust duct. In addition, there were no fire curbs in the doorways of the Auxiliary Boiler Room. This would allow an oil spill in the Auxiliary Boiler Room to spread into the turbine railroad bay and turbine building stairwell.

Safety Evaluation: The modifications are not safety related and do not pose safety hazards which are not directly analyzed for or implicit in the DAEC Final Safety Analysis Report.

DCR No. 1094 Containment Radiation Monitoring Panels 1C-219A/B

Description of Change: Based on the facts that the Grab Sample Assembly on panels 1C-219A/B has never been used and that abnormal operations of the H_2O_2 analyzers may result in Tech. Spec. violations or frequent issuance of LCOs, it was determined that electrically disarming the two solenoid valves of the Grab Sample Assembly system on 1C-219A/B and physically removing the Grab Sample Flasks off panels 1C-219A/B, would avoid inadvertently reconnecting the solenoid valves in the future.

Reason for Change: The Grab Sample logic of the Containment Radiation Monitoring panels 1C-219A/B included two solenoid valves (one being N.O. and the other N.C.) which, upon Hi Radiation conditions in the Process Sample lines, cycle so process air is directed towards and collected by a Grab Sample Flask installed on the front of both panels 1C-219A/B. It was determined that during this process a pressure spike occurs on the process sample lines that are common between the Containment Radiation Monitoring panels 1C-219A/B and the adjacent H_2O_2 analyzers since their replacement during the 1981 refueling outage with newer more sensitive panels.

Safety Evaluation: This design change is not safety related. It has no impact upon safe shutdown of the plant, nor any hazard to the public, nor does it jeopardize any other associated and/or related safety systems. This design change package does not constitute an unreviewed safety question.

However, it shall be noted that the Containment Radiation Monitoring panels 1C-219A/B are safety related. The Grab Sample bottle system of the panels is only an available option (feature) and although it is part of a safety related panel, the Grab Sample system itself is not safety related. This is based on the fact that its one intended function is to collect an air sample in the bottle during Hi Radiation conditions in the process line, and enable the H.P.s to test it. Removal of the sample bottles from the seismic panels 1C-219A and B will not affect the original seismic qualifications of the panels based on the fact that the weight of the bottle is negligible compared to the overall weight of the panel.

DCR No. 1100 Software Modification for Radiation Effluent Monitoring System

Description of Change: This change documents the installation of computer programs on the DAEC VAX. The programs calculate effluent releases from the DAEC.

Reason for Change: These programs replace similar programs on the University of Iowa computer.

Safety Evaluation: This is not a safety related change and implementation of this change will not affect the integrity of other safety related systems or the ability to safely shut down the plant.

This DCR implements software changes only and does not add any equipment or implement any physical changes.

DCR No. 1106 — 1P-70 Fill Pump Impeller Change

Description of Change: This DCR installed a new, larger impeller in the Fill Pump raising its current 45-50 psig discharge pressure up to 65-70 psig. In addition, indicating lights were added to the door of the pump motor MCC cubicle to identify pump operating status.

Reason for Change: The impeller was changed to maintain a higher pressure in the RHR loops. This was to ensure that the RHR piping is at a pressure greater than atmospheric to prevent water hammer.

Safety Evaluation: Placing a larger impeller in the existing pump:

1) Increases fill system pressure. This not only is acceptable but desirable. The system now operates at a pressure of 70 psig rather than the previous 50 psig. This will not adversely affect any system alarms. The system piping has been designed for a pressure of 2300 psig.

2) Increases pump motor load - The existing pump motor is a 3 Hp Westinghouse motor with a service factor of 1.15. The new impeller was chosen so as not to overload the motor at the pump design flow of 10 gpm. Margin is provided by the service factor for temporary, intermittent flow rates in excess of 10 gpm.

3) Affects the seismic/stress analysis of the pump. The seismic/stress analysis for the pump has been reviewed. The impeller change does not affect the conclusions of the analysis. Deflections, stresses and pump natural frequency are still very conservatively acceptable.

The addition of red and green indicator lights to MCC 1B-441G is acceptable because:

1) Normal lamp failure mode is burn out or broken filament, both of which cause an open circuit. An open circuit previously existed across those spare contacts.

2) The addition of a single conductor wire and 2 lights on the MCC door will not adversely affect the seismic integrity of the MCC.

DCR No. 1107 Torus Water Clean up System

Description of Change: Several design changes were incorporated into the Torus Drain System to aid in its operation. A local handswitch for starting the pump was added. To permit pumping the low volume of water for hydrolasing, an air operated diaphragm pump capable of passing a water/silt mixture was added in parallel to the existing 450 gpm pump. To allow drainage of very small amounts of fluid by gravity to the floor drain, a drain with fire hose connection was added. Connections were added at various locations along the piping to allow backflush of clogged pumps and piping and to allow dilution of the silt/water mixture entering the pump(s). A tie-in to the Service Condensate System was installed to provide flush water for the Torus Water Cleanup System.

Reason for Change: The torus must be drained periodically in order to perform work on its internals. The presently installed Torus Water Cleanup System was inadequate with respect to the ability to drain the torus and the excessive manpower required to operate the system.

The present system worked until torus level dropped to a point where vortexing above the drain nozzle inhibited pumping. Throttling the pump discharge aided in reducing the problem but was only a partial solution. As the torus level dropped further, sludge entering the pump suction became a significant problem as the pump was not designed for such service. The sludge also clogged the suction strainer requiring frequent spool piece removal for cleaning. The pump's capacity (450 gpm) was adequate for initial draining of the torus but was too large for use during hydrolasing (one hydrolaser was used with a rate of 18 gpm).

Safety Evaluation: The Torus Water Cleanup System does not interface with any safety systems except the torus. It is isolated from the torus by a removable spool during all modes of operation except cold shutdown.

DCR No. 1108 Feedwater Pumps Seal Line Modifications

Description of Change: 1 1/2" globe valves were added on the downstream side of CV-1203 and CV-1210 in the seal water return lines. Flanges (8 pair) were added near the feed pump main shaft seals in the condensate injection lines and seal water return lines.

Reason for Change: This design change was necessary to make isolation of seal water return lines more accessible. The present isolation valves are located in high rad areas. Troublesome unions in the condensate injection lines and seal water return lines were a maintenance problem.

Safety Evaluation: This Design Change is not safety-related. This change does not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. This change does not create the possibility of an accident different than previously evaluated in the FSAR. This change does not reduce the margin of safety defined in the basis of Technical Specifications.

DCR No. 1109 Replace Solenoid Valves in RHR, CAS, SBGT and NSSS Systems

Description of Change: Replaced the existing 21 solenoid valves with 21 new qualified Automatic Switch Company (ASCO) NP solenoid valves.

Reason for Change: IE Bulletin 79-01B and its supplements.

Safety Evaluation: All of the new ASCO solenoid valves are Class 1E safety-related components, wit ema 4 housing. By evaluating the requirements of the exis and new solenoids, it is concluded that the new so id valves ensure the operability of the system during conditions since the new qualified NP ASCO solenoids mee C requirements. The new solenoid valves are the same size, shape and configuration but are made of new material able to withstand environmental and seismic conditions. The new solenoid valves will not degrade associated control valve operation and will not endanger the health and safety of the public as described in the FSAR. Based on these considerations, it can be concluded that there are no unreviewed safety questions or Technical Specification changes. The criteria of 10CFR50.59 provides the basis for making this modification.

DCR No. 1110 Torus Internal Structural Modifications

Description of Change: The work consisted of the addition of two stiffener brackets to the bottom of each existing torus ring girder (16 locations).

Reason for Change: These changes were part of Mark I program upgrade. This work increased the capacity of the suppression chamber ring girders and catwalk platforms/supports to resist the postulated dynamic loads due to LOCA and safety/relief valve discharge events.

Also included was the removal of existing vacuum breaker platforms, railings, and their associated supports (8 locations), along with the addition of new handrail members required to fill newly created voids. New mid-span hanger supports were added to the catwalk in each non-vent bay (8 loactions) of the suppression chamber.

The catwalk platforms were reinforced by the addition of new cross and diagonal bracing spanning between existing platform stringers in all 16 bays of the suppression chamber. Existing handrail and toeplate members on the catwalk platform were reinforced at selected locations throughout the catwalk.

Safety Evaluation: The catwalk platform/support and ring girder modifications increase the load capacity of these components and do not degrade the existing suppression chamber pressure boundary or ring girders. The suppression chamber catwalk platforms/supports are not addressed in the Technical Specifications and the function of the suppression chamber shell and ring girder is unchanged.

DCR No. 1111 Diesel Generator Bearings Prelube System

Description of Change: The capacity of the lube oil heater was increased from 6KW to 15KW in order to maintain an increased lube oil temperature and reduced viscosity. The piping from the discharge of the new heater to a point where existing prelube was piped in the discharge pipe of the engine driven pump was changed. The upper lube oil header supply line was rerouted so that the header does not readily drain. Increased size electrical circuits were installed to supply the new larger lube oil heater.

Reason for Change: Annual inspection of standby diesel generator 1G-21 revealed a wiped lower crankshaft main bearing (#12) and a wiped thrust bearing (#13) on both the journal and thrust surfaces. The redundant standby diesel generator 1G-31 had similar problems. Wiped bearings was a recurring problem. Since September, 1977, several Licensee Event Reports (LERs) had been written against the diesel generator because of this bearing problem.

Safety Evaluation: The new continuous operating lube oil system provides proper lubrication of the diesel generator bearings when the diesel generator is not running. The new lube oil system for the diesel generator bearings is more reliable and very effective for proper lubrication of the bearings. The design change does not constitute any change in the DAEC Technical Specifications or an unreviewed safety question, previously evaluated in the FSAR. This modification does not jeopardize the safety of any other system or personnel present in the area. This modification adds more reliability to the diesel generator prelube oil system and, hence, to the diesel generator bearings.

DCR No. 1112 Changeout of Sprinkler Heads on Deluge System Number 6

Description of Change: The existing upright sprinklers were replaced with three directional spray nozzles (Viking Spray Nozzles, Model A-2 with 140° inserts) and three pendent sprinklers (Viking pendent sprinkler, Model C, 1/4" orifice). The pendent sprinklers were installed on the north side of the hydrogen seal oil unit as well as above the unit. The directional spray nozzles were installed on the south, east, and west sides of the hydrogen seal oil unit.

Reason for Change: A deficiency was identified in the existing deluge fire protection for the hydrogen seal oil unit. Normally, this deficiency would be corrected by replacing the upright sprinklers with pendent sprinklers by means of a MAR. However, after reviewing the layout of Deluge System Number 6, Design Engineering concluded that the fire protection of the hydrogen seal oil unit could be greatly enhanced by changing some of the upright sprinklers at the tank's side to directional spray nozzles.

Safety Evalaution: The modification described in this DCR is not safety-related and does not pose any safety hazards which are not directly analyzed for or implicit in the DAEC Final Safety Analysis Report.

DCR No. 1113 Annunciators and Computer Logging for Station Batteries

Description of Change: Annunciators and computer logging were added to each of the 125 VDC batteries (1D1 and 1D2) and the 250 VDC batteries (1D4).

Reason for Change: NRC IE Information Notice No. 81-05 described a situation which occurred at Consumer's Power Co. Palisades Plant in which both redundant 125 VDC batteries were disconnected from their associated buses. There were

no annunciators or alarms located in the Control Room to advise the operators that either of the batteries had been disconnected. A review of the DC systems at DAEC concluded that a Palisades type event could occur here, i.e., undetected common-mode station blackout caused by disconnecting both 125 VDC batteries concurrent with the loss of offsite power.

Safety Evaluation: This design change is not safety-related and does not adversely impact any safety-related systems or components. Therefore, no unreviewed safety questions exist as a result of this change. This design change enhances the functionability of the station battery systems and, consequently, improves safe plant operation with annunciation and computer logging in the control room.

DCR No. 1114 Torus Penetration/Internal Piping Project

Description of Change: The work consisted of the procurement, fabrication, installation and inspection of modifications to selected external suppression chamber piping penetrations and to selected small bore and large bore torus attached piping and pipe supports inside the torus.

Reason for Change: These modifications were required to restore the structural design margins in accordance with construction code requirements for newly defined hydrodynamic loads resulting from LOCA or SRV discharge events which were unknown at the time of the original design. This work was part of the Mark I program upgrade.

Safety Evaluation: The modifications do not change the Technical Specifications incorporated in the operating license. The probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR has not been increased. The penetration reinforcement modifications increased the load capacity of the penetrations and did not degrade the existing suppression chamber pressure boundary. The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR has not been created. The margin of safety, as defined in the basis for Technical Specifications, has not been reduced. The function of the suppression chamber shell and penetration nozzles is unchanged. The modifications were performed in accordance with requirements of the ASME Section XI, as documented in the Design Specification No. IOW-44-100.

DCR No. 1115 SRVDL Vent Line Penetration and Wet Well Modification Project

Description of Change: The SRVDL vent line penetration was modified by the replacement of reinforcing steel at the nozzle to insert plate junction and at the insert plate to shell junction. Additionally the SRVDL below the nozzle was reinforced by replacement. The elbow support beam was reinforced by the addition of plates on the top and bottom of the existing beam. The connections to the ring girder were also reinforced by adding plates. A box structure was installed to connect the beam to the SRV discharge line. Reinforcing gussets were added to the T-quencher support brackets. These connect each of the six support bracket bolting plates and the quencher support beam. A reinforcing T-section was added to the T-quencher lateral support beam.

Reason for Change: These modifications were required to restore the structural design margins in accordance with construction code requirements for newly defined hydrodynamic loads resulting from LOCA or SRV discharge events which were unknown at the time of the original design. This work was part of the Mark I program upgrade.

Safety Evaluation: The modifications did not change the Technical Specifications incorporated in the operating license. The probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR is not increased. The modifications to, and addition of, supports on the SRVDL increased the capacity of the piping system to resist the defined loads. The possibility of an accident or malfunction of a different type than previously evaluated in the FSAR is not created. The margin of safety, as defined in the basis for Technical Specifications, is not reduced. It was demonstrated that stress levels in the piping and supports are within ASME Code allowables.

DCR No. 1116 Torus Attached Piping Project - Outage Related

Description of Change: The work consisted of modifications and/or additions to the discharge piping and discharge piping supports outside the torus. The specific piping systems and modifications installed are listed in the detailed design package.

Reason for Change: The purpose of this work was to increase the capacity of the piping and pipe supports to resist the postulated dynamic loads due to LOCA and SRV discharge events and restore the TAP design margins of safety to those originally specified in the DAEC FSAR. This work was part of the Mark I program upgrade.

Safety Evaluation: The pipe and pipe support modifications increased the capacity of these components to resist the defined loads. The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR has not been created. The margin of safety, as defined in the basis for Technical Specifications, has not been reduced.

DCR No. 1117 Torus Attached Piping Project-Nonoutage Related

Description of Change: This change includes removal, disassembly, fabrication, installation, painting, inspection and adjustment of design modification to external small and large bore torus attached piping and piping support assemblies. Additionally the work included removing existing structural and concrete attachments, modifying existing facilities where necessary, installing new pipe supports and snubber assemblies and returning the facilities to a condition acceptable for operation.

Reason for Change: This change was to increase the capacity of the piping and pipe supports to resist the postulated dynamic loads due to LOCA and SRV discharge events and restore the TAP design margins of safety to those originally specified in the DAEC FSAR. This work was part of the Mark I program upgrade.

Safety Evaluation: The probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR has not increased. The pipe and pipe support modifications increased the capacity of these components to resist the defined loads. The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR has not been created. The margin of safety, as defined in the basis for Technical Specifications, has not been reduced.

DCR No. 1118 Torus Support Hold Down Modification

Description of Change: The change consisted of the addition of new stiffeners and base plate brackets to the existing suppression chamber saddles and column supports. Additional anchor bolts were installed to transmit uplift loads from the suppression chamber columns and saddles to the foundation. These modifications to the suppression chamber anchorage provided increased load capacity to the torus saddle/column supports.

Reason for Change: The purpose of this change was to increase the capacity of the suppression chamber anchorage to resist the postulated dynamic loads due to LOCA and safety/relief valve discharge events. This work was part of the Mark I program upgrade.

Safety Evaluation: The suppression chamber anchorage modifications increased the load capacity of these components and did not degrade the existing suppression chamber pressure boundary. The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR has not been created. The margin of safety, as defined in the basis for Technical Specifications, has not been reduced. The function of the suppression chamber anchorage is unchanged.

DCR No. 1119 Electro Hydraulic Control Pump Discharge Filter Maintenance

Description of Change: Manual isolation valves were installed downstream from existing check valves to provide double valve isolation and prevent leakage of fluid during EHC pump discharge filter maintenance.

Reason for Change: The EHC pump discharge filters require periodic maintenance while the plant is on line. The two pumps are full-capacity and are connected in parallel. This allows the continuous operation of the hydraulic power unit while either pump loop is shut down. In the past maintenance was performed using a check valve for isolation. Performing maintenance on a high energy system with single valve isolation was a poor safety practice. Also, the leakage of fluid from the check valve led to cleanup and disposal costs. Should leakage have been significant, it would have prevented reassembly of the discharge filter and required a shutdown of the EHC system and generator.

Safety Evaluation: This change does not present any significant hazards not described or implicit in the FSAR. The possibility exists for starting a pump with the isolation valve in the closed position. However, the relief valve would act to prevent any dangerous build-up of pressure. Safety is increased during maintenance.

DCR No. 1120 Main Steam Root Valve Replacement and Removal

Description of Change: Unneccessary root valves were eliminated, and others were replaced with packless valves. All installed valves are non-nuclear and of the packless diaghragm globe valve variety. The diaphragm prevents escape of the steam, eliminating the need to repack. These valves are in the same location as the previous valves. Previous valves which served no purpose were removed and the pipe capped or plugged.

Reason for Change: The root valves in the vicinity of the main steam lines just upstream of the stop valves required repacking on an average of every two years to prevent leakage. Due to their location, these valves were difficult to repack. The leakage caused an increase in radiation exposure and could have resulted in a shutdown.

Safety Evaluation: This change does not present any significant hazards not described or implicit in the FSAR. Leaking valves or valves prone to leaking were removed or replaced by valves of a higher pressure rating and specially designed not to leak. Without leakage, personnel will be exposed to a lesser amount of radiation.

DCR No. 1121 <u>Standby Gas Treatment System Fenwal Controllers Relocation</u>

<u>Description of Change:</u> Fenwal controllers located in local control station terminal boxes mounted on SGTS units were relocated.

<u>Reason of Change:</u> Due to the radiation effect on electronic components, it was not feasible to replace the Fenwal system with qualified equipment unless it was heavily shielded. Therefore, the local control stations along with the Fenwal controllers were relocated to a less harsh radiation environment. This change was to comply with IE Bulletin 79-01B.

<u>Safety Evaluation:</u> This DCR relocated the existing local control stations with their Fenwal controllers to a less harsn radiation environment in order to minimize the cumulative radiation exposure. The relocation of the existing control stations and their Fenwal controllers did not change the operation of the SGTS. Therefore, the probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety, which was previously addressed in the FSAR, does nct increase. The possibility of an accident or malfunction of a different type than that which was previously discussed in the FSAR is not created. The margin of safety is not reduced.

DCR No. 1123 <u>Main Steam Isolation Valve Lifting Lug Modification</u>

<u>Description of Change:</u> A series of lift and pull lugs sized for a standard clevis, were installed. Supporting assemblies were designed and installed on the existing structure. These lugs are designed to safely carry a 2000 lb. load up to 20° each side of vertical in the steam tunnel and 2000 lb. in any direction in the drywell.

<u>Reason for Change:</u> MSIV maintenance was very difficult due to the lack of anchorage points for hoists to lift MSIV components. For personnel and equipment safety, ALARA, and time savings requirements, an arrangement of anchor points is required above the MSIV's both in the steam tunnel and the drywell. These changes were performed as part of work to comply with NUREG 0612.

<u>Safety Evaluation:</u> The lifting lugs themselves are not safety related items. The affect of installing and loading the lugs on the existing structure was analyzed. Following are the results of this analysis:

1. The stresses induced in the existing steam tunnel structure and all drywell steel 18" or greater in depth by the added MSIV loads are small to the point of being negligible when compared to the existing structure's capacity.

2. The lugs will be loaded only while the plant is down. Therefore operational and extreme environmental loads will not be combined with the lug loading.

3. The duration of loading on the lugs will be very short.

4. The lugs will be positioned so that weld shrinkage effects and weld-induced secondary stresses on the existing steel will be minimal.

The end result of the analysis is that installing and loading the lifting lugs will have a negligible effect on the design margin of safety of the existing structure.

DCR No. 1124 <u>Suppression Pool Temperature Monitor System Thermowells</u>

<u>Description of Change:</u> New thermowells were installed in the torus. These thermowells provide the proper receptacles for temperature sensors and their associated electrical wiring which will be specified in a later DCR. A still later DCR will provide for the termination and control instrumentation for the temperature sensors. Justification of the thermowell locations is given in NSEO-106-1282.

<u>Reason for Change:</u> The Suppression Pool Temperature Monitoring System (SPTMS) was installed at the DAEC to meet the requirements of NUREG 0661 and Mark I program upgrade.

<u>Safety Evaluation:</u> This DCR installed 16 thermowells through half couplings which are mounted on the outside wall of the the suppression pool wall. The thermowell installation required the temporary penetration of primary containment (torus shell). The subsequent installation of the thermowells at each point where the torus was penetrated reestablished the primary containment pressure boundary. The design for containment penetration and thermowell installation has been specified in accordance with FSAR Section 5.2.3.4.2, ASME Section III for Class 2 Pressure Vessels and testing has been specified in accordance with 10CFR50, Appendix J. The thermowell installation is consistent with original design requirements in the torus which establish the primary containment boundary. Performance of containment leak testing following thermowell installation verified that the containment leak rate is within design requirements.

DCR No. 1125 <u>Diesel Generator Room Door Curbs</u>

<u>Description of Change:</u> Installed additional curbing outside each Diesel Generator Room door in addition to leaving present curbing on the inside of the DG rooms.

Reason for Change: As identified in an NRC Inspection Report, the curbs that were installed by a DCR were incorrectly designed to be placed inside the Diesel Generator Rooms. The purpose of these curbs is to keep a fire caused by flammable and combustible liquid from spreading into the rooms. This was in accordance with a Fire Protection Safety Evaluation Report. With the curbs inside the rooms, a fire would spread to both sides of the three hour fire doors, theoretically downgrading them to 1-1/2 hour doors based on NFPA definitions of fire doors.

Safety Evaluation: The proposed change does not present any significant hazards or considerations not described or implicit in the Final Safety Analysis Report. The addition of the curbs in fact provides increased assurance that the Diesel Generators operation will not be violated by a fire outside the rooms. In addition, the integrity of the turbine building is not affected by this change.

DCR No. 1128 Installation Of An Additional Flammable Liquids Storage Locker

Description of Change: An additional 45 gallon flammable liquids storage locker was located adjacent to the two existing flammable liquids storage lockers. This DCR also included additional foundation, protective framing, and welding for the new locker.

Reason for Change: Two 45 gallon Flammable Liquids Storage Lockers were located within the Reactor Building Railroad Air Lock Bay. However, additional capacity was required for storage of paint, oil, and solvents which are needed for use within the plant. In addition, because of the limited storage space for flammable liquids, it was sometimes difficult to fix responsibility for proper locker usage/maintenance to individual plant user groups. An additional storage locker allows the plant Fire Marshall the option of assigning specific lockers to users, thereby clearly defining responsibility for locker upkeep.

Safety Evaluation: This modification was not safety related. The addition of a third flammable liquids storage locker in the Reactor Building RR Airlock does not present additional changes which had not been evaluated in the Final Safety Analysis Report and does not reduce the margin of safety defined in the basis of any Technical Specification. There is no safety related equipment in the Reactor Building R.R. Airlock.

DCR No. 1129 1P-1B Seal Water Return Line Modification

Description of Change: Stress cracking of the half coupling welds was relieved by installation of thermal expansion loops in the seal water return line. Return pressure was also raised in both feed pump seal water return lines to minimize flashing and pipe erosion by restoring the pump seal water controls to the values recommended by the pump manufacturer.

Reason for Change: The reactor feedwater pump 1P-1B seal water 1-1/2" return line, had developed cracking of the welds at the half coupling joining a 10" line. On inspection of the coupling and pipe, it was noted that the 1-1/2" line was severely eroded on the downstream side of the last valve. Reactor feedwater pump 1P-1A seal water return line had also exhibited erosion at the point of discharge into the heater spill line.

Safety Evaluation: The piping changes do not present any significant safety hazards or considerations not described or implicit in the Final Safety Analysis Report. Repair and recalibration of the seal water discharge controls restored the system to the design described in the DAEC FSAR.

DCR No. 1132 Standby Gas Treatment Deluge Piping

Description of Change: This design change consisted of the rerouting of two small sections of pipe and the enlarging of the drain line orifices from 1/8" to 1/4".

Reason for Change: Problems with the Standby Gas Treatment System charcoal bed deluge system resulted in the wetting of the charcoal. It was determined that a contributing cause of the wet charcoal bed was the design of the deluge system piping, including the small 1/8" diameter of the drain line orifice.

Safety Evaluation: This design change has no impact on any safety considerations or safety evaluations in the FSAR. The deluge piping is routed over safety-related equipment and was seismically supported in accordance with seismic Class "II over I". The piping changes were supported per seismic Class I requirements.

DCR No. 1133 Reactor Water Clean-up Piping Replacement

Description of Change: A section of 4" RWCU piping was replaced with low carbon stainless steel piping and fittings.

Reason for Change: Two crack indications were found and identified along 2 welds during NDE performed on the RWCU system. The replacement material, low carbon austenitic stainless material, has good weldability and higher resistance to intergranular stress corrosion cracking.

Safety Evaluation: This change was non-safety related. This DCR provided replacement of a section of a 4" RWCU piping where two crack indications were identified. The replacement piping was SS304L/SS316L material which has improved weldability and higher resistance to intergranular stress as compared to the original SS304 material. SS304L/SS316L material has marginally reduced tensile strength as compared to SS304 material. However, per calculation on the evaluation of replacement piping, the stresses are within allowable stress values of the replacement piping material.

DCR No. 1135A Breathing Air Piping Network - Part A

Description of Change: This DCR authorized the deletion of a portion of the existing breathing air piping and the installation of a replacement network of higher capacity and better quality. The network is a stainless steel piping scheme to deliver Grade D breathing air from an outside air supply system to specific high use areas throughout the plant. Only the piping above elevation 773'-0" in the Reactor Building was included in the scope of DCR 1135A. This portion of the network serves the upper floors of the Reactor Building with five individual stations on the Refueling Floor and one Station in the RWCU Heat Exchanger Room.

Reason for Change: The previous service-air-supplied carbon steel Breathing Air Supply System was inadequate in terms of capacity, air quality and efficiency.

Safety Evaluation: This portion of the system is located such that it could not affect the operation of safety-related equipment either before, during or after accident. Since no "2 over 1" condition exists on this portion of the system, the piping is designed to be equivalent to all other non-safety-related piping in the plant. The effects of installing this portion of the system on plant structural integrity have been evaluated and found to be negligible.

DCR No. 1137 Suppression Pool Temperature Monitoring System

Description of Change: This DCR specified the system design for a Class 1E Suppression Pool Temperature Monitoring System. However, only one (1) division consisting of eight (8) RTDs was connected under this DCR to meet the requirements of NUREG 0661 and Mark I program upgrade. The second division of eight RTDs will be connected in the future. The cable was installed to the cable spreading room and coiled for later use. The required Class 1E thermowells were purchased and installed in DCR package 1124.

Reason for Change: The suppression pool temperature monitoring system (SPTMS) was installed at the DAEC to meet the requirements of NUREG 0661. The justification for the number and location of these temperature sensors is contained in GE study NSEO-106-1282, DCR Index Item #6.14, and IE verification of that study.

Safety Evaluation: The equipment installed under this DCR performs no safety-related function per FSAR 5.2.3.11 and presents no significant hazard or consideration not described or implicit in the Safety Analysis Report. The safety-related status of this DCR is due to the physical interface of the cable installation with Division I and Division II cable tray systems. This DCR was part of an overall plan to install in the Control Room a means of visual indication for the Control Room operators of the torus water temperatures. Even though the work done in this DCR performs no safety-related function, all design and installation work was done to Class 1E requirements.

DCR No. 1140 Reactor Feedwater Pumps Suction Valves Bypass Lines Supports

Description of Change: One additional pipe support was installed on each Reactor Feedwater Pump suction valve bypass line.

Reason For Change: The Reactor Feedwater Pumps suction valves bypass lines amplify the vibration in the 18-inch suction lines. To prevent failure of the nozzle welds at the connections between the suction line and the bypass line caused by fatigue, an additional support to reduce the bypass line vibration amplitude was added.

Safety Evaluation: This change does not present any significant safety hazards or considerations not described or implicit in the Final Safety Analysis Report. Addition of this support does not significantly increase the loading on the existing suction valves support. This load increase adds less than 0.6% to the existing pipe support load and is judged not significant. Therefore, no unreviewed safety question exists for this modification.

DCR No. 1143 Diesel Fire Pump Engine Cooler Piping

Description of Change: Engine cooling discharge piping was increased from 1" to 1 1/2" and a separate drain tie provided for the pump gearbox cooling discharge line.

Reason for Change: The diesel fire pump diesel engine overheated due to inadequately sized engine cooler discharge piping and being tied with cooling water discharge from the pump gearbox.

Safety Evaluation: This change is non-safety related. This design change does not interface with any safety related system and does not affect any safety support system. This change does not affect the operation of the Fire Protection System so no operating procedure revisions are required. No Technical Specification changes are required.

DCR No. 1145 Chlorine Leak Detection System

Description of Change: Two new high chlorine gas concentration detectors were installed in the pump house. This simplified the scheme by eliminating the "stepping switch" and allowing individual detection and indication for each room. The sample tube that was connected between the detector and the evaporator exhaust line was removed. The sample tubing was replaced with plastic sample tubing to prevent chlorine corrosion.

Reason for Change: The Chlorine Leak Detection System indication did not make it clear which room had a high chlorine gas concentration. This presented a hazard to personnel safety. A "stepping switch" for sampling the Chlorine Storage, Evaporation, and Dispenser and Pump and Educator rooms caused the problem. It stopped sequencing through the rooms when high chlorine was detected in any one of them preventing indication of the other rooms. Another problem was that the "chloralert" detector was drawing chlorine gas directly from the evaporator exhaust line into the detector thru a sample tube. An erroneous alarm occurred when the evaporator discharged chlorine gas to the atmosphere outside the pumphouse. Also, the sample tubes were corroded.

Safety Evaluation: These modifications are within the Chlorination and Acid Feed System. It is not a safety related system. There are no interfaces between the modifications and any safety related equipment or systems. Physical separation is maintained between the modifications and the safety equipment south of the pumphouse door. Therefore, these changes have no impact on Licensing, the FSAR or Technical Specifications. Therefore, there will be no increase in probability, consequences or type of accidents, and there will be no reduction in the margin of safety at the DAEC. There are no unreviewed safety questions.

DCR No. 1146 Sprinkler Drain Modifications in Railroad Bays

Description of Change: This change extended the 2" main drain to a point capable of receiving full flow discharge. Smooth bore corrosion resistant orifice was added giving flow equivalent to one sprinkler to the test pipe for inspector's tests. This terminates at a point capable of receiving that flow. The non-listed control gate valve on the Reactor Building railcar airlock was replaced with an UL listed indicating gate valve. The test connection of system #8 was extended through the Reactor Building RR airlock wall and a sprinkler head was installed with the deflector and yoke removed. The test connection for system #9 was extended through the wall in the railroad bay and a sprinkler head was installed with the deflector and yoke removed.

Reason for Change: These modifications were performed to meet the requirements of NFPA-13, Sprinkler Systems, and recommendation 80-2 of American Nuclear Insurers (ANI) for the Reactor Building railcar airlock sprinkler system #8 and the Turbine Building railroad bay sprinkler system #9.

Safety Evaluation: This change is not safety related and does not present additional hazards which have not been evaluated in the Final Safety Analysis Report or Technical Specifications.

DCR No. 1155 Feedwater Pumps Monorails

Description of Change: This design change installed permanent monorails with trolleys and hoists above and parallel to the feedwater pump shafts. FW Pump 1P-1B required a curved monorail to avoid relocating the instrument rack at the north end of FW Pump 1P-1B. To reduce the frequency of manual cleaning of the lube oil coolers, backwash valving was provided to reverse the cooling water flow direction, periodically back flushing the exchanger tubes.

Reason for Change: Normal annual feedwater pump maintenance includes removal of the upper pump casing half and the pump rotor. Previous monorails used for removal of these loads were I-beams suspended from the ceiling by hanger rods. Also, the feedwater pumps lube oil coolers tend to silt up with time, reducing the coolers' heat transfer capacity.

Safety Evaluation: No impact on plant safety will occur as a result of this design change. Movement of the feedwater pump upper casing and rotor by the monorail will not increase the consequences of any accident or increase the likelihood of an accident. Inadvertent dropping of the FW pump components while being moved would not cause any decrease in safety margins. This design change decreases the probability of dropping components. Movement of these components complies with NUREG 0612, Section 5.1, Criterion IV.

DCR No. 1165 HPCI Turbine Casing Drain Line

Description of Change: A union fitting on the HPCI turbine casing drain line was replaced with straight pipe of the same size and type as existing pipe.

Reason for Change: The union fitting was a source of leakage which was inconsistent with the DAEC leak reduction program.

Safety Evaluation: This design change does not impact any safety considerations or safety evaluations previously considered in the FSAR. Neither the function nor structural integrity of the subject piping is impaired. The only effect of this design change is to remove a leak location (union) from the subject piping.

DCR No. 1166 Drywell Purge Debris Screens

Description of Change: This modification provides debris screens for the drywell purge supply and exhaust lines.

Reason for Change: The debris screens provide protection for the isolation valves against debris that may become entrained in the escaping air/steam mass due to the occurrence of a LOCA while purging with the reactor at power. This change was made to comply with NUREG 0737, Item II.E.4.2.

Safety Evaluation: The design modification does not present any significant hazards or considerations not described or implicit in the safety analysis report. The sole safety functions of the debris screen is to protect the drywell purge exhaust isolation valves from debris in the event of an LOCA while purging. Subsequent to closure of the isolation valves, no safety function is associated with the debris screen.

DCR No. 1167 Purge Isolation Valve Leak Test Connection

Description of Change: The change added a 3/4-inch test line downstream of CV-4310 on line 2"-HLE-23. This line is routed outside the H&V valve room where a 3/4-inch isolation test valve (V-43-226) and test connection were installed. The modification provides leak testing capability from outside the H&V valve room during normal power operation.

Reason for Change: This change was made to comply with NUREG 0737, Item II.E.4.2. Technical Specification change RTS-133, requires leak testing for control valves CV-4302 and CV-4303 during power operations. The existing leak test connection was located within the heating and ventilation (H&V) valve room. The H&V valve room serves as a shield to protect against neutron streaming through penetration X-25 and cannot be accessed during power operations.

Safety Evaluation: The modification does not present any significant hazards or considerations not described or implicit in the safety analysis report. The change added a test connection similar to other test connections described in the FSAR to allow leak testing to be performed in an ALARA manner during power operations. The safety functions of the purge system, purge isolation valves, and the standby gas treatment system are not affected by this design change. The design change is the result of NRC generic concerns regarding containment purging and venting which required compliance with Branch Technical Position CSB 6-4. CSB 6-4 requires the capability for leakage testing of purge/vent system valves equipped with resilient seats during power operation.

The new leak rate test line piping and valve do not perform an active safety function. However, they are part of the pressure boundary of the containment atmosphere control system and an extension of the containment. Therefore, all welding and nondestructive testing (including pneumatic testing) was performed and documented in accordance with the applicable sections of the ANSI standards or ASME nuclear vessel codes. A Seismic Category I analysis was performed.

DCR No. 1170 Security Camera Door #225

Description of Change: A camera was added at the access control door. An intercom was added to provide quick communications.

Reason for Change: The addition of the camera helps identify the cause of an alarm.

Safety Evaluation: This DCR is not safety related and does not have any relationship to any safety systems. This DCR will require a core drill between the administrative building and secondary containment above door #225 and will also require a core drill into the airlock at access control from secondary containment. These activities have been carefully coordinated with plant operations in order to meet Tech. Spec. requirements of section 3.7.c, Secondary Containment.

DCR No. 1174 MSIV Leakage Control System Relay Replacement

Description of Change: This modification replaced the original General Electri. relays 145C3238 used in this system with Amerace/Agastat Model EGPI002 relays which are acceptable replacements.

Reason for Change: The original GE relays were failing and GE no longer supplies these relays.

Safety Evaluation: Review of the environmental and seismic qualification of the Amerace/Agastat relays determined that these relays are acceptable replacements for the original relays. The Amerace/Agastat relays have the identical contact configurations, contact ratings and coil ratings. Therefore, there is no unreviewed safety question associated with this modification.

DCR No. 1178 Add LLS System to Control Non-ADS SRVs

Description of Change: A Low-Low Set (LLS) relief logic system was installed to automatically control the two non-ADS safety relief valves (SRVs). The LLS logic causes the non-ADS SRVs to blow down for a longer period of time to allow the water leg in the discharge line to return to its normal level before another SRV actuation. The LLS system is designed as a Class 1E system because it supports a component (torus) which is important to safety.

Reason for Change: This system was installed to mitigate postulated thrust load concerns of subsequent actuations of SRVs during an abnormal transient or a small break loss-of coolant accident (LOCA) and to reduce stresses on the torus. This change is required to ensure that the Mark I containment is in conformance with the requirements of NUREG-0661.

Safety Evaluation: The addition of the LL Set Function to PSV-4407 does not involve an unreviewed safety question. The safety function of thse non-ADS SRVs presented in the FSAR (Section 4.4) is the self-actuated overpressure relief at a fixed setpoint. This safety function, as presented in the FSAR, will not be affected by the LL Set modification. The probability of occurence or the magnitude of the consequences of an accident or malfunction of PSV-4401 and PSV-4407 as previously evaluated in the FSAR will not be increased, and the possibility of an accident or malfunction of a different type than those previously evaluated for PSV-4401 and PSV-4407 in the FSAR has not been created. The margin of safety as defined in the bases of any Technical Specification will not be decreased because the LL Set modification has no effect on the safety function of the valves. Considerations in this evaluation are detailed in this DCR on file.

DCR No. 1181 Nitrogen Supply to Non-ADS Safety/Relief Valves (SRVs)

Description of Change: As part of Mark I SRVs load cases evaluation, a new feature of Low-Low Set function was added to the non-ADS SRVs to mitigate certain load cases. To assure reliability of SRVs operation, nitrogen supply tubing was rerouted by tapping into lines from accumulators to ADS Safety/Relief Valves.

Reason for Change: This modification enables the non-ADS SRVs to operate in a manner which will reduce the loading on the torus in certain cases.

Safety Evaluation: The accumulators that were tapped into by this modification were oversized for the original intended use. Calculation M83-02 determined that these accumulators will support both the original ADS usage and the new Low-Low Set function with a significant safety margin still present. Therefore, no unreviewed safety questions exist.

DCR No. 1182 RPV Level and Main Steamline Pressure Setpoint Change for Group 1 Isolation

Description of Change: The reactor pressure vessel (RPV) water level setpoint for the closure of the main steam line isolation valves (MSIV) was reduced from Level 2 to Level 1. The RPV pressure setpoint for MSIV closure was reduced from 880 psig to 850 psig.

Reason for Change: The RPV pressure setpoint was too close to the operating pressure and thus needless isolations could occur due to spurious pressure transients.

Safety Evaluation: The effect of the lowered RPV level setpoint was determined in the DAEC Mark I Plant Unique Analysis performed by General Electric and published as report NEPC 22204. This GE report established that lowering the RPV level setpoint for MSIV closure does not involve an unreviewed safety question.

The DAEC Plant Unique Licensing Supplement determined that the lower limit of the RPV pressure setpoint for MSIV closure is 825 psig. Therefore the new 850 psig setpoint is justified and no unreviewed safety question exists. NRC approval was obtained through Technical Specification amendment prior to installing this change.

DCR No. 1183 Main Steam Isolation and Automatic Depressurization System

Description of Change: The two logic channels of ADS were separated into two physically independent wiring channels. Scheme 1R1210 was renamed ADS Logic B and the scheme number was changed to reflect a Division II scheme. Routing was done in Division II raceway. When the wiring for Logic B was routed, it was also physically separated from the HPCI system routing. To meet separation criteria for the interfacing of Division I, II, III, and IV of the manual control of ADS, two handswitches for the Division II and IV operators on panel 1C03 were divisionalized to panel 1C45 and all cabling installed in conduit.

Reason for Change: As stated in the UFSAR the automatic depressurization system (ADS) is currently designed as Division I engineered safeguard system which does provide backup to the Division II engineered safeguard system HPCI. All tests of ADS indicated that it had performed functionally as it was designed. The design changes brought ADS to a higher degree of reliability and follow applicable separation criterias.

Safety Evaluation: The modifications placed the logic 'B' channel wiring with Division II wiring separate from HPCI wiring. Also, the project extended the separation criteria into the connecting manual control subsystem of both the Auto Depress Valves and the manual relief valves. The modifications have increased the ability of the Auto Depress system to withstand wireway and control cabinet failures without loss of the ADS function. As ADS reliability was improved, the consequences of an accident previously evaluated in the FSAR has not increased. Similarly, a different type of accident than previously analyzed has not been created. This modification does not reduce the margin of safety for ADS or the DAEC as discussed in DAEC Technical Specifications Section. This modification does not involve an unreviewed safety question.

DCR No. 1196 Service Platform Lifting Lugs Modification

Description of Change: The three existing lifting lugs of the Reactor Vessel Service Platform were replaced by three new lifting lugs based on Bechtel's design calculation.

Reason for Change: The lifting lugs of the Reactor Vessel Service Platform did not meet the safety factor of 10 required by NRC's NUREG-0612.

Safety Evaluation: A 10CFR50.59 review was performed for this work. No change was required to the plant Technical Specification and no unreviewed safety questions were created by this change. The margin of safety as defined in the bases for the FSAR and Technical Specifications is not affected.

DCR No. 1199 Reactor Well Shield Plug Lifting Lugs Modification

Description of Change: Each of the 24 Reactor Well Shield Plug lifting lugs was modified by welding a new plate to one side of the existing plate.

Reason for Change: The lifting lugs of the Reactor Well Shield Plugs did not meet the safety factor of 10 required by NRC's NUREG-0612.

Safety Evaluation: A 10CFR50.59 review was performed for this work. No change to the plant Technical Specification was required and no unreviewed safety questions were created by this change. The margin of safety as defined in the basis for the Final Safety Analysis Report and Technical Specification was not affected.

DCR No. 1200 Drywell Head Lifting Lugs Modification

Description of Change: The four existing lifiting lugs of the Drywell Head were modified by welding two washers on both sides of each lifting lug based on Bechtel's design calculation No. 273-31. These lifting lugs were modified during the 1983 outage by an emergency DCR (letter NG-83-847, dated March 1, 1983.)

Reason for Change: The lifting lugs of the Drywell Head did not meet the safety factor of 10 required by NRC's NUREG-0612.

Safety Evaluation: A 10CFR50.59 review has been performed for this work. No change to the plant Technical Specification was required and no unreviewed safety questions were created by this change. The margin of safety as defined in the bases for the FSAR and Technical Specifications was not affected.

DCR No. 1208 Diesel Generator Intake Ductwork Repair

Description of Change: In order to gain access to the diesel generator intake ductwork, a 22" x 22" opening was established near the damaged area of the ductwork. After completing the necessary inspection and repairs, a 24" x 24" Ruskin Access Door, hinged and with double latches, was installed to close the access opening.

Reason for Change: The seismically supported, safety related, air intake ductwork to diesel generator 1G-21 was damaged by the trolley of the Turbine Building crane. As a result of this damage, the ductwork required inspection of its internal insulation and repairs to the affected areas. Previous to the implementation of this DCR, access to the interior of this ductwork was unavailable.

Safety Evaluation: The new access door in the duct was installed such that the door is held closed by the air pressure of the duct as well as by the latches on the door. The change was consistent with the existing duct system and did not degrade the system. The duct system was not changed in geometry, location or support from that previously installed. This design change does not affect any Technical Specification.

DCR No. 1210 Pipe Hangers Downstream of Turbine Bypass

Description of Change: This change replaced the rigid hangers with variable load supports. These supports allow thermal expansion and contraction of the piping while providing the support required.

Reason for Change: The subject hangers, are located on the 10-inch extraction line running from the high pressure turbine to the first stage reheaters. Hanger anchorage is made in the concrete of the turbine pedestal. The previous hangers were rigid vertical supports and had insufficient flexibility to withstand loads induced by thermal expansion of the piping.

Safety Evalulation: The pipe involved is by definition not safety related and the modification is therefore non-safety related. No safety related equipment is located under the pipe and failure of the hangers would not endanger safe shutdown of the plant.

DCR No. 1211 Termination for Scram Pilot Solenoids

Description of Change: This modification mounted a small terminal strip inside the condulet of each CRD unit scram pilot solenoid. The solenoid pigtail wires and the control cable wires were then connected at the terminal strip using qualified ring tongue screw lugs.

Reason for Change: Crimp connections on the cable between the solenoid and the control cable required the wire to be shortened each time the solenoid was rebuilt. The control cable was becoming so short that a new cable would have to be pulled. Use of screw lugs and a terminal strip eliminated future cable shortening.

Safety Evaluation: The terminal blocks are not qualified to IEEE 323-1974. This qualification, however, is not now required in the CRD's mild environment. Each set of blocks has very low mass (about 3 oz.) that will have no impact on the CRD seismic response. Each set of terminal blocks is anchored by two screws to the condulet. In the unlikely event that the terminal blocks break loose, the condulet has a cover plate to contain the blocks. In the unlikely event that the control and solenoid wires separate, the scram solenoids will fail to the scram (de-energized) state. This change does not increase the probability of an accident or increase the magnitude of consequences of an accident or malfunction of equipment important to safety. No new accident possibility is introduced. The margin of safety, as defined in Technical Specifications, is not reduced.

DCR No. 1216 Reactor Head Spray Pipe Hangers

Description of Change: The change involved adding a 1/2 inch threaded rod through the support arms of the pipe hangers and securing it with double nuts. Both of the pipe hangers in question were drilled and bolted. This method of repair permanently fastened the support arms to prevent them from spreading apart and captured the existing forging rod pin securely while still permitting rotation.

Reason for Change: While replacing the reactor insulation head to which pipe hangers are anchored, it is believed movement of the head during outages caused the pipe hanger support arms to spread apart. The existing forging rod pin did function with the support arms spread apart. However, had the spreading condition continued much longer, the pin would have failed.

Safety Evaluation: Addition of this spacer bolt does not cause any decrease in safety margin or create the possibility of an accident or malfunction of a different type than any previously evaluated because the constant support pipe hanger operation is not changed from the original design. Because the operation of the hanger is not changed, the possibility of an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created. The functional operation of the hanger is not affected by this change. Therefore, the margin of safety, as defined in the basis for any Technical Specification, is not reduced.

DCR No. 1220 Cable Replacement for Motor Operator - 2401

Description of Change: Cable ends were replaced, which involved replacement of a portion of cable approximately 35 feet long. In order to facilitate any future replacements, a new terminal box was added. A total of 15 terminal points were required using two eight point terminal blocks.

Reason for Change: The insulation on the wires to the valve were "found to be bad" when the sealtite was removed.

Safety Evaluation: The terminal blocks are fully qualified to IEEE 323-1974 and IEEE 344-1975. Also, the terminal boxes in which the terminal blocks were installed are seismically supported. This change enhances the functionability of MO-2401.

DCR No. 1224 LLS Test Light Modification

Description of Change: This DCR lowered the circuit resistance in the LLS actuation circuit in order to increase current flow to the neon lights. All circuit changes were made internal to control room cabinet 1C45.

Reason for Change: This work was part of the Mark I program upgrade. This design change increased the current in the LLS actuation circuit to a level that is sufficient to operate the single failure (neon) lights.

Safety Evaluation: The lowering of the resistance in the actuation circuits of the LLS System does not involve an unreviewed safety question. The safety evaluation for the LLS System as previously evaluated (Ref: DCR 1178) is still valid. Changing the circuit resistance does not change the design intent of the LLS System.

DCR No. 1228 Add Vertical Support to 3/4" - HLE - 13

Description of Change: A standard small pipe hanger was added to the vertical run of 3/4 inch vent top line from an RHR test line.

Reason for Change: This 3/4" line is a class 2 vent tap from an RHR test line. In the as-built condition, the line did not meet the code allowable stresses and was sagging. This condition was outside the FSAR and the Technical Specifications.

Safety Evaluation: A 10CFR50.59 review has been performed for this DCR. No change to the Plant Technical Specifications was required, and no unreviewed safety questions were created by this DCR. The probability of an accident or malfunction is reduced because the line was upgraded to meet the code requirements. The margin of safety as defined in the basis for the Technical Specifications is unchanged.

DCR No. 1229 Dryer Separator Canal Plug Lifting Lugs Modification

Description of Change: A half-inch washer plate was welded to each face of the existing lifting lugs to increase the bearing surface. The two washers and lug have a minimum combined thickness of 2.857 inches to provide for a minimum safety factor of 10, and up to a maximum combined thickness of 3.0625 inches to provide an acceptable minimum clearance between the shackle pin and the lug hole.

Reason for Change: Iowa Electric's commitment to NUREG-0612 required that several heavy loads on the refueling floor be upgraded to meet the safety factor of 10. The Dryer Separator Canal Plug lifting lugs of the upper plug failed to meet the required safety factor due to bearing stress. Bechtel recommended a modification which would allow the lug's safety factor to meet the NUREG-0612 requirement.

Safety Evaluation: A 10CFR50.59 review has been performed for this work. No change to the plant Technical Specification was required and no unreviewed safety questions were created by this change. The margin of safety as defined in the bases for the Technical Specification is not affected.

DCR No. 1230 Refueling Slot Gate (outer) Lifting Lugs Modification

Description of Change: The spent fuel slot gate lifting lugs were modified to allow the lugs' safety factor to meet the NUREG-612 requirements. The change consisted of removing the original lugs (2) and replacing them with 1-1/4-inch plate lugs (2) without chamfer at the hole.

Reason for Change: The spent fuel slot gate (outer) lifting lugs did not meet the safety factor of 10 required by Iowa

Electric's commitment to NUREG-0612, secton 5.1.6(3)(b). The lifting lugs were analyzed by Bechtel and it was found that the lugs failed to meet the required safety factor due to bearing stress.

Safety Evaluation: A 10CFR50.59 review has been performed for this work. No change to plant Technical Specifications was required and no unreviewed safety questions were created by this change. The margin of safety as defined in the basis for the Technical Specifications is not affected.

DCR No. 1232 Reactor Building Hatch Lug Modification

Description of Change: In order to increase the Reactor Building Hatch's safety factor, a 1/4" thick x 7" wide x2'8-1/2" long steel plate was welded along the bottom leg of the angle iron of each hatch section.

Reason for Change: The lifting lugs of the reactor building hatch did not meet the safety standards of NUREG 612, which required a design safety factor of 10 for a non-redundant or non-dual lift point system. The safety factor for the rigging hatch was 8.3, and therefore needed to be increased. Mode of failure of the RB hatch during lifting was flexure of the 3" x 2" x 1/4" angle iron.

Safety Evaluation: This DCR provided for an increase in the safety factor of the reactor building hatch in order to comply with the heavy loads requirements in NUREG 612. No change to the plant Technical Specification was required and no unreviewed safety questions were created by this change. The margin of safety as defined in the basis for the FSAR and the Technical Specifications is not affected.

DCR No. 1233 Refueling Shield Lug Modification

Description of Change: A 1/2" x 3-1/4" x 0'3-3/4" plate with a 2" diameter hole drilled through it was welded to one face of each lug. The 1/2" plate is ASTM A36 (certified).

Reason for Change: The existing safety factor for the refueling shield lifting lugs was 8.4 (as analyzed by Bechtel) and did not meet the safety factor of 10 required by NUREG-0612.

Safety Evaluation: A 10CFR50.59 review was performed for this DCR. No change to the Plant Technical Specifications was required and no unreviewed safety questions were created by this DCR. The probability of an accident or malfunction is reduced because the safety factor is improved. The margin of safety as defined in the bases for the Technical Specifications is unchanged.

DCR No. 1234 Reactor Vessel Head (RPV) Strongback Modification

Description of Change: Additional ASTM A36 plate was welded to the RPV strongback and four plates were welded on the top.

Reason for Change: The existing strongback had a safety factor of approximately 3 which did not meet the minimum safety factor of 10 required by NUREG-0612.

Safety Evaluation: A 10CFR50.59 review was performed for this DCR. No change to the Plant Tech. Specs. was required, and no unreviewed safety questions were created by this DCR. The probability of an accident or malfunction is reduced because the safety factor is improved. The possibility of an accident or malfunction of a different type is not created because there was no change in the system. The margin of safety as defined in the basis for the Tech. Specs. is unchanged.

Section B
PROCEDURE CHANGES

This section has been prepared in accordance with the requirements of 10CFR, Part 50.59(b)

B. Procedure Changes.

During 1983, various procedures as described in the safety analysis report were revised and updated. All changes were reviewed against 10CFR50.59 by the DAEC Operations Committee. No procedure changes were performed that constituted unreviewed safety questions.

All special test procedures performed in 1983 were also reviewed by DAEC Operations Committee. No unreviewed safety questions were found. Summaries of these special tests are listed below.

SpTP No. 103 RWSS Screen Wash Pump 1P-1123 Discharge Pressure Test

The purpose of this test was to provide a method of testing the shut off head of the RWSS travelling screen wash pump 1P-112B for determination of pump wear.

This special test was performed January 4, 1983.

SpTP No. 104 Installation and Testing of Limit Switches and Relays for Radwaste Conveyor.

This procedure provided a method of testing the radwaste drum handling conveyor system for determination of defective components.

This special test was performed August 10, 1983.

SpTP No. 105 Containment Atmosphere Monitoring System Helium Leak Test

The purpose of these tests was to identify sources of leakage from the CAM system as part of the Annual Leakage Measurement Program outlined in the Tech. Spec., Sect. 6.8.5 and specific commitments resulting from NUREG 0578, Sect. 2.1.6.a.

This special test was performed May 4, 1983.

SpTP No. 106 VOID

Section C
EXPERIMENTS

This section has been prepared in accordance with the requirements of 10CFR, Part 50.59(b)

C. Experiments

There were no experiments conducted during the calendar year 1983.

Section D

SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES

This section has been prepared in accordance with the requirements of NUREG 0737, Item II.K.3.3 and in accordance with Iowa Electric commitments provided by a letter from L. D. Root to H. R. Denton dated December 31, 1980 (LDR-80-393).

D. Safety and Relief Valve Failures and Challenges

This section contains information concerining relief valve and safety valve failures and challenges for calendar year 1983. Note that all instances in which the main steam relief valves were manually cycled open, for surveillance testing or other reasons, are also included for your information. There were no (0) safety valve failures or challenges during 1983. There were no (0) relief valve failures during 1983. There were two (2) relief valve challenges during 1983. These events are described below.

Date	Event Description
05/05/83	Relief valves PSV-4400, -4401, -4402, -4405, -4406, and -4407 were opened and closed during the satisfactory completion of a normal surveillance test.
10/28/83	Relief valves PSV-4401 and PSV-4407 were opened and closed to relieve vessel pressure and lower vessel level during shutdown following a scram and Group I isolation.

DMB

Iowa Electric Light and Power Company

March 1, 1984
DAEC-84-129

PRINCIPAL STAFF			
RA		DPRP	✓
D/RA		DE	
A/RA		DRMSP	
RC		DRMA	
PAO		SCS	
SGA		ML	
ENF		File	

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Mr. James G. Keppler
Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Duane Arnold Energy Center
OP License DPR-49
Docket No. 50-331
1983 Annual Report of Facility Changes, Tests, Experiments, and Safety and Relief Valve Failures and Challenges

Dear Mr. Keppler:

In accordance with the requirements of Appendix A to Operating License DPR-49, 10CFR50.59(b), Regulatory Guide 10.1, and NUREG 0737, Item II.K.3.3 please find enclosed the original and 39 copies of the subject report for the period of January 1, 1983 thru December 31, 1983.

Very truly yours,

Rick Hannen for

Daniel L. Mineck
Plant Superintendent - Nuclear
Duane Arnold Energy Center

DLM/WRK/pv*

Enclosure

cc: Director of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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