

IOWA ELECTRIC LIGHT AND POWER COMPANY
Duane Arnold Energy Center

1981

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

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 bases for plant design changes completed during the calendar year 1981, and summaries of the safety evaluations for those changes, pursuant to the requirements of 10CFR, Part 50.59(b).

DCR No. 279 Change Pressure Switch and Trip Setting

Description of Change: Radwaste solids handling system pump pressure switches were replaced with ASCO Tri-Point units and present switch pressure setting of 25 inches HG absolute was reset to 10 inches HG absolute.

Reason for Change: The original switches in this application would not work since the process (approx. 20% solids) tends to plug bourdon tubes. The original settings caused the pumps to trip each time they are started due to a starting transient pressure drop to 15 inches HG absolute.

Safety Evaluation: This is not a safety system and does not interface with a safety system. This change will replace the existing pressure switches with more reliable ones. Resetting the pressure switches to 10 inches HG absolute will provide more conservative pump protection than 5 inches HG absolute.

DCR No. 496 Provide Permanent Radiation Shielding

Description of Change: Neutron shielding material was added around the containment spray line penetration (X-39A). An annular ring approximately 2 1/4 inches thick and 2 feet long of biosilicone foam, consisting of 4-6 inch rings, was inserted between the process pipe and containment penetration sleeve.

Reason for Change: With the reactor operating at approximately 85% power there is a 5 to 20 MR/HOUR neutron dose at contact around the penetration. The additional shielding material should reduce the dose rate to below 0.5 MR/HR.

Safety Evaluation: This change did not present significant hazards considerations not already addressed in the Safety Analysis Report. The modification does not change or affect any safety related equipment.

DCR No. 540 Cooling Tower Water Level Control

Description of Change: The capacitance probes and associated equipment were replaced with a bubbler system.

Reason for Change: The capacitance level system gives false level readings, due to its tendency to accumulate algae.

Safety Evaluation: This work did not change any of the previously approved design functions of this non-safety related equipment. This change will improve the accuracy and reliability of the water level alarm/control system.

DCR No. 586

Diesel Generator Lube Oil Makeup Tank Level Switch

Description of Change: Level switches on diesel tanks 1T-1/4A&B were relocated and level indicators were installed. Additional restraints and guides were added on piping.

Reason for Change: Two level switches were located too high on the tank and the alarm was normally in, while two other switches were located too low and the tank emptied before alarm annunciation. Level indication did not previously exist. The instrument piping had never been seismically restrained, so to prevent pipes breaking due to a seismic event restraints were added.

Safety Evaluation: This change did not present significant hazards or considerations not already addressed in the Safety Analysis Report. The integrity of the instrument piping has been upgraded and equipment facilitating maintenance has been installed.

DCR No. 599

Addition of Discharge Temperature Recorder

Description of Change: A canal discharge temperature recorder was added in panel 1C102 to continuously monitor the canal discharge temperature.

Reason for Change: Technical Specifications require the canal discharge temperature to be continuously monitored. This system will provide a backup for the computer monitoring.

Safety Evaluation: This change did not present significant hazards or considerations not already addressed in the Safety Analysis Report. The new recorder is a non-safety related component and was mounted in a non-safety related panel. This addition does not change the reliability of any of the interfaces with safety systems previously approved.

DCR No. 690

Relocate Vent Valves, Core Spray System

Description of Change: The high-point vent line (3/4 in-EBB-17) has been relocated on the discharge of core spray pump 1P-211A (8"-EBB-17) outside the Reactor Water Cleanup Heat Exchanger Room immediately downstream of valve MO-2115.

Reason for Change: The vent line was originally located in the Reactor Water Cleanup Heat Exchanger Room, which is a very high radiation area. The vent line was moved to a readily accessible location in order to facilitate the periodic operational check of the Core Spray System.

Safety Evaluation: This change does not present significant hazards or considerations not described or implicit in the Safety Analysis Report. The vent line does not perform an active safety function but it is part of the pressure boundary of the Core Spray System which is a safety system. All welding and non-destructive testing was performed and documented. A Seismic Class I analysis was performed.

DCR No. 702

Auxiliary Boiler Recirculation Pump

Description of Change: The auxiliary boiler system has been modified to allow periodic sampling and chemical addition of the boiler water during recirculation and while the boiler is shutdown.

Reason for Change: During shutdown of this system no method had previously been available for mixing water treatment chemicals with the boiler water, although a tank for this had originally been provided.

Safety Evaluation: The auxiliary heating boiler recirculation system is not safety related nor does it interface with any safety related systems. Operation of the recirculation system is not required for safe shutdown of the reactor.

DCR No. 721

Blank Flange MSIV Leakoff Valve Line

Description of Change: Handle blanks were placed between the two flanges of each MSIV Leakoff Line since replacing the Leakoff Line valves would be more expensive than shutting the lines off.

Reason for Change: The "Handle Blanks" were used to shut off the MSIV packing leak-off lines since leakage was occurring through the leak-off line valves.

Safety Evaluation: The proposed change does not present significant hazards or considerations not described or implicit in the Safety Analysis Report. Since the valves on the MSIV leak-off lines are operated in the closed position during operation, the blanking off of these

lines will give increased assurance that any valve leakage during operation or abnormal conditions will not violate the primary system isolation boundary.

DCR No. 700D

Installation of Switchgear

Description of Change: Installation of switchgear for the Recirculation Pump Trip (RPT) system.

Reason for Change: The RPT system will reduce the severity of thermal transients in the fuel caused by turbine trip (turbine stop valve closure) or generator load rejection (fast control valve closure) with assumed bypass failure. RPT will be accomplished by installing and tripping redundant breakers between the recirculation pump drive motor/generator sets and the recirculation pump motors.

Safety Evaluation: The RPT System was designed in compliance with IEEE-279-1971 except for section 4.17 which covers manual trip features. The two systems are electrically, mechanically, and physically independent. Further the severity of events considered in the safety analysis report are reduced by the installation of the RPT System. There is no potential for a new type of event being created which is more severe than those already analyzed. Therefore, it can be concluded that there is no unreviewed safety question.

DCR No. 724

Torus Support Reinforcement

Description of Change: The work involved the design and installation of the torus support reinforcement.

Reason for Change: This change was made to restore structural design margins in conformance with construction code requirements for newly defined loads unknown at the time of the original design.

Safety Evaluation: The probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety was not increased. The addition of the reinforcement increased the capacity of the torus supports for containment loadings.

The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR was not created. The torus supports are passive structural components and thus have no potential for creation of a new accident.

The margin of safety, as defined in the basis for technical specifications, was not reduced. Suppression chamber supports are not the subject of technical specifications.

The reinforcement of the suppression chamber supports was performed in accordance with the requirements of ASME Section III and XI. Attachment of the reinforcement was to a "butter" layer previously deposited on the torus shell rather than attaching directly to the shell.

As part of the installation of the reinforcing web assemblies the kicker struts connecting the base of the torus support column and the bottom of the torus shell was removed. The absence of this member existed only until the reinforcing web assembly was installed. The removal of this member was evaluated with regard to the loads given in the original design report and Mark I Containment Short Term Program loads and it was determined that the kicker struts can be removed without violating Mark I Containment Short Term Program Criteria.

A stress analysis was performed to determine the increased capacity of the torus supports brought about by their reinforcement. This analysis and the modification are documented in NUTECH Report NO. IOW-05-036, "Design Report for Torus Support Reinforcement". The existing torus supports have a ASME Code allowable capacity of 284 kips with the critical element being the column connection to shell weld. The reinforced assembly has a ASME Code Service Level A allowable capacity of 760 kips, an increase of 168% over the existing design. Thus, the capacity of the containment to resist loads as defined in the original design specification and those presently being evaluated in the Mark I Containment Program has been increased by the reinforcement of the torus supports.

DCR No. 737

Emergency Diesel Generator Fuel Oil Daytank

Description of Change: This change removed existing float type level switches and replaced them with Barton Model 288A switches.

Reason for Change: The existing level switches were not qualified for seismic category 1 service. This change provided the reliability required for seismic category 1 components.

Safety Evaluation: The change did not present significant hazards or considerations not described or implied in the Safety Analysis Report. The change increased the systems operability to be consistent with the requirements for seismic category 1 service.

DCR No. 749

Fire Protection System

Description of Change: This change increased the diesel fire pump engine speed to improve pump performance.

Reason for Change: This change was made to ensure the requirements of Technical Specification 4.13.B.1.e are met.

Safety Evaluation: This system is not required for safe shutdown of the plant. The change did not alter the original safety analysis.

DCR No. 814

Fire Protection Hose Stations for Cable Spreading Room And Control Room

Description of Change: This change provided hose stations at the southwest entrance to the cable spreading room, the northwest entrance to the control room, and in the southwest corner of the control room.

Reason for Change: This change was made as a result of Fire Protection SER items 3.1.4.1 and 3.1.4.2.

Safety Evaluation: This system is not required for safe shutdown of the plant. The change did not alter the original safety analysis. This change provides greater availability for safety related systems.

DCR No. 823

Relocate Hydrogen Analyzers

Description of Change: This change was to move hydrogen analyzers AN 4162A & B from the offgas retention building at elevation 739'6" to the truck bay in the offgas retention building at elevation 757'6".

Reason for Change: The previous location of the analyzers was below the water level in the hot well of the main condenser. At the new location the analyzers can drain to the main condenser.

Safety Evaluation: The change did not present significant hazards or considerations not described or implicit in the Safety Analysis Report. The hydrogen analyzers are not safety related and the connecting piping is neither safety related nor seismic Class 1. A seismic Class 1 analysis is not required.

DCR No. 825

Water Separator Loop Seal

Description of Change: This change increased the diameter of the vertical pipe rise downstream of the loop seal so that water surface tension cannot support the water column necessary to siphon the loop dry.

Reason for Change: To help prevent the water separator loop seal from blowing out into the equipment drain sump which then vents gas to the reactor building.

Safety Evaluation: This change is neither a part of nor does it interface with any safety related system. The change does not effect the original safety analysis.

DCR No. 838

RWCU Sample (2738)

Description of Change: This design change modified the piping for the reactor water cleanup system sample line.

Reason for Change: This change allowed proper sampling of the reactor water cleanup system in both the hot and cold system line up operating modes.

Safety Evaluation: This change did not alter the basic design function and did not affect any safety related equipment.

DCR No. 840

Modifications to the Service Air System

Description of Change: This change installed an air receiver and a check valve into the air line to the radwaste filters and an air receiver and a check valve into the air line to the condensate filter demineralizers. The receivers have a drain trap with a manual bypass to drain any water which might backflow from the filters.

Reason for Change: During operation the service air system is used to clean plant filter demineralizers. However, when these filters are on-line, operating pressure exceeds that of the service air system. During this condition valve leakage could allow contamination to enter the service air system. The addition of an air receiver and a check valve provided an additional barrier to prevent or at least slow the backflow of water into the service air system.

Safety Evaluation: This change did not pose any significant hazards not described or implicit in the Safety Analysis Report. The service air system does not itself perform an active safety function nor would its failure prevent a safe reactor shutdown and these changes did not interface with safety related systems.

161/345 KV Substation

Description of Change: Each transmission line circuit breaker at the DAEC was equipped with reclosing relays which automatically reclosed the breaker once, approximately 1/3 second after opening the breaker to clear a line fault. This is termed "High Speed Reclosing". Its purpose is to quickly restore transmission circuits after temporary faults due to lightning surges. The modification replaced the present reclosing relay with a reclosing relay that will check for synchronism before allowing a reclosure. It blocks reclosure for severe faults, but allows selective reclosing on less severe faults. For a successful reclosure, the operating time is approximately 1.3 seconds.

Reason for Change: Turbine Generator manufacturers now recommend elimination of high speed reclosing at generating stations because of significant probability of resulting damage to turbine generator shafts. Consultation with G.E. engineers and computer studies of fault and switching transients for the DAEC unit, confirm that these recommendations are applicable to the DAEC unit.

Safety Evaluation: No new failure modes were created. The change reduced the severity of Turbine-Generator transients initiated by a fault since the line breakers will not reclose into a permanent fault. This change does not change the consequences of any previously analyzed accident or malfunction. The margin of safety is not affected.

Separation of Diesel Generator Air Intakes

Description of Change: This change provided separation between the air intakes for standby diesel generators 1G-21 and 1G-31.

Reason for Change: This change reduced the probability of a fire in one diesel generator room impacting the operability of the redundant diesel generator.

Safety Evaluation: This change did not present any significant hazards not described or implicit in the safety analysis report. This change increased the reliability of one standby diesel generator in the event of a fire in the other diesel generator room. The functions of the diesel generators were not altered or reduced by this change. The seismic integrity of the new ductwork and penthouses was assured by proper design, analysis and installation.

DCR No. 862

Corrections to DCR 524

Description of Change: The design philosophy for this DCR was the same as that of the closed DCR 524 as originally defined. However, additional changes were found to be necessary so this new DCR was prepared. DCR 524 installed selector switch HS-6102A in Panel 1C-179.

Reason for Change: This change makes corrections on drawings and provides some field connection changes.

Safety Evaluation: The safety evaluation provided in DCR 524 is also applicable to this change.

DCR No. 865

Torus Drain Lines

Description of Change: The change involved the design, installation and testing of the torus drain lines.

Reason for Change: This change was necessary to facilitate easier and faster torus drainage.

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 Final Safety Analysis Report (FSAR) has not been increased. The torus drain line has been designed to meet the same requirement as the other containment penetrations. Additionally the drain line incorporates a flexible expansion bellows to accommodate the dynamic displacements of the torus shell when subjected to hydrodynamic loads.

The possibility for an accident or malfunction of a different type other than previously evaluated in the FSAR has not been created. The torus drain line terminates in a bolted blind flange with the valve opened only during refueling outages. Thus during plant operation the drain line serves no active role for shutdown and has no potential for creation of a new accident.

The margin of safety, as defined in the basis for the Technical Specifications, has not been reduced. Reactor operation with the drain line will remain unchanged. The drain line has the ability to accommodate greater torus displacements and accelerations than existing penetrations and therefore does not decrease the margin of safety.

The torus drain lines were installed in accordance with the requirements of ASME Section XI and ANSI B31.7 for Class II piping.

An evaluation was performed to calculate stress levels in the torus penetration and attached shell, drain piping and supports resulting from torus loadings. The evaluation utilized the methodology contained in GE Report NEDO-21888, "Mark I Containment Program, Load Definition Report". The results of this evaluation are contained in NUTECH Report No. IOW-05-114, "Design Report for Torus Drain Pipe Installation" and show that for the loads considered all member stresses are within appropriate allowables.

DCR NO. 868

Replace SV7600A/B to Correct Operational Problem

Description of Change: This change provided new solenoid valves which don't require minimum operating pressure for dependable operation or for seating the valve properly.

Reason for Change: The previous solenoid valves had a 10 PSI minimum design operating pressure. However, a 10 PSI pressure was not easy to maintain with a working system pressure range of 3-15 PSI. The previous solenoid valves did not always seat tight in the open or energized position.

Safety Evaluation: The change did not present significant hazard considerations to the safety of the plant. It is a more conservative approach than the original valve design. The systems safety function was not changed from that of existing, since the logic remains the same.

DCR No. 869

Replacement of Chemical Feed Flow Switches

Description of Change: This DCR replaced flow switches FS-8000, FS-8004A&B and FS-8012A&B with pressure switches. This change was a direct replacement of the switches with no functional change involved.

Reason for Change: The previous flow switches had proven to be unreliable. Replacement of these switches reduced the maintenance required by the chlorination and acid feed systems.

Safety Evaluation: This change did not present any hazards not described or implicit in the Safety Analysis Report. These switches are not safety related nor do they affect any safety related systems.

DCR No. 874

Drain Line on Line to CV-5837A&B

Description of Change: This change provided a drain line just upstream of CV 5837A&B with a one inch gate valve and associated piping.

Reason for Change: This change was needed to drain the piping before CV 5837A&B for STP 413C005. The drain valve was needed to prevent water from entering the carbon bed filter on the standby gas treatment system while performing this fire protection STP.

Safety Evaluation: The change did not present significant hazards or considerations not described or implicit in the Safety Analysis Report and the original design criteria was not altered. The piping is classified as seismic Category II, requiring no detailed seismic stress analysis.

DCR No. 876

Reroute Condenser Drains to 1P-7

Description of Change: This DCR routed the drain for the condenser water boxes into the circulating water drain pump suction.

Reason for Change: This change allowed the water boxes to be drained without dumping high conductivity water into the radwaste system.

Safety Evaluation: This change did not add any significant safety hazards not described or implicit in the safety analysis report. This change did not interface with any safety related systems nor can it affect the safe shutdown of the Reactor.

DCR No. 883

Valve V-19-48 RHR Cross-Tie Indication

Description of Change: This change added limit switches to RHR manual cross-tie valve V-19-48 to indicate valve position on control room panel 1C-03.

Reason for Change: The change was made to provide remote indication of valve position which augments system safety.

Safety Evaluation: The valve is safety related. However, the additional weight of two limit switches and associated brackets is extremely negligible compared to the weight of the valve. In addition, the limit switches in no way affect the operation of the valve.

DCR No. 886

Process Computer

Description of Change: This change provided reactor high water level signals from LS 4559, 4560 and 4561 to the process computer.

Reason for Change: This change provides a record of these points printed out which will be helpful in diagnosing plant trips due to reactor high water level.

Safety Evaluation: The change did not affect any safety related system and is not related to safety itself; therefore, the change does not pose an unreviewed safety question.

DCR No. 890

Installation of a Light in the Turbine Building

Description of Change: This DCR installed light F1-2-40 between condensate pumps 1P8A, 1P8B and the south wall of the heater bay in the turbine building.

Reason for Change: This change was to make the area safer for working personnel.

Safety Evaluation: Installation of this light fixture had no affect upon any safety systems that are located in the area. Loss of power to the light or failure of the fixture will in no way cause any mitigating consequences that would affect the health and safety to the public.

DCR No. 893

Adjust Setpoint of PCV 7333

Description of Change: The setpoint of PCV 7333 was changed from 80 psig to 95 psig.

Reason for Change: In the previous configuration with PCV 7333 set at 80 psig and compressors 1K-3 and 4 set to start at 90 psig decreasing pressure (PS 7333A and B) the compressors ran frequently to maintain the system pressure at 90 - 100 psig and no contribution was made from the main plant air system. Resetting PCV 7333 from 80 to 95 psig insured that the main plant air system will be the primary supply to the H&V instrument air system and compressors 1K-3 and 4 will serve only as a backup. On loss of the plant air system, 1K-3 and 4, designed to seismic category I requirements, will maintain system pressure.

Safety Evaluation: The change resulted in less frequent operation of compressors 1K-3 and 1K-4, thus increasing their reliability. The change did not present any additional safety hazards or considerations not presently described or implicit in the safety analysis report.

DCR No. 896

Jockey Fire Pump

Description of Change: This change installed a 2-stage turbine pump for high head low flow service as the jockey fire pump. Also installed was an open tank with float controlled level supplied by the well water system on the suction side of the pump. The change also furnished a 400 gallon surge tank as an accumulator to help maintain the pressure in the fire protection distribution piping system.

Reason for Change: The previous jockey fire pump design had proven to be unreliable which had resulted in much maintenance activity and frequent operation of the main fire pumps.

Safety Evaluation: This change posed no unreviewed safety questions, since the changes did not interfere with any safety system or interface with safety functions.

DCR No. 897

Replacing Traversing Incore Probe (TIP)

Description of Change: This change replaced the existing thermal TIP with General Electric's Gamma TIP System.

Reason for Change: This change provided DAEC with a more accurate indication of reactor power distribution.

Safety Evaluation: The gamma TIP hardware, software and services are classified as "not safety related" and therefore do not present any significant safety consideration.

DCR No. 899

Paint and Oil Storage - Railroad Airlock

Description of Change: This DCR replaced two standard storage lockers with safety cabinets designed for flammable liquids. The lockers were installed in the railroad airlock to provide accessibility to the rest of the plant.

Reason for Change: Oil and paint used for routine plant operation are stored in the railroad airlock area. The safety cabinets meet the requirements of NFPA #30 for the storage of flammable and combustible liquids.

Safety Evaluation: This change did not present any significant hazards not described or implicit in the safety analysis report. This change reduced the possibility of a fire which would damage the plant.

DCR No. 912C

TSC Instrument Installation (Radiation Monitoring)

Description of Change: An Eberline Monitoring System consisting of three detector assemblies, three readout channels, and three remote indicators providing the capability to monitor radiation at both the TSC air intake and within the TSC itself was installed.

Reason for Change: This system was installed in the TSC to meet the requirements of NUREG 0578, Section 2.2.2.B.

Safety Evaluation: The radiation monitoring system for the TSC is not safety related. It has no interface with any other radiation monitoring system associated with the plant.

DCR No. 915

Suppression Chamber RHR Discharge Line Elbow and Support Modification

Description of Change: The change involved the design and installation of reducing elbows and structural supports to the RHR full flow test return lines within the torus.

Reason for Change: The supports restored structural design margins in conformance with construction code requirements for newly defined loads not considered at the time of the original design. The reducing elbows promote mixing of the suppression pool water during a safety relief valve discharge event.

Safety Evaluation: An analysis of the RHR modification yielded stresses in the suppression chamber penetration, the added supports, the return line, and the torus shell to which attachments are welded which were less than the applicable code allowable stresses.

DCR No. 917

Condensate Demineralizers/Torus Connection

Description of Change: This change provided the capability to pump water from the torus during torus cleanup operations directly to the condensate demineralizer system.

Reason for Change: This change eliminated the need to use prefilters during torus cleaning and draining operations.

Safety Evaluation: The change did not present significant safety hazards or considerations not previously described or implicit in the Safety Analysis Report. The condensate system is not safety related. The connections provided will be used only during plant shut down during torus clean up. Normally, the two valves of these connections are closed.

DCR No. 920

Move Valve V-34-20

Description of Change: Valve V-34-20 was replaced and was repositioned within its system.

Reason for Change: The valve was relocated so that it could be safely operated.

Safety Evaluation: This DCR did not functionally change any system. The movement of this valve reduced the hazards to operations personnel. The system itself is not safety-related nor will this change present any hazards or considerations not described or implicit in the safety analysis.

DCR No. 931

Independent Identification of MO 4841A&B

Description of Change: This DCR provided independent identification for each of the two containment isolation valves MO 4841A&B.

Reason for Change: This change was made so the control room operator could determine the position of each valve independently.

Safety Evaluation: This change did not affect the safety function as already described in the FSAR, and therefore it did not present any new hazard consideration to the system.

DCR No. 935

Installation of ESW System's Additional Flow Indicators and Pump Motor Ammeters

Description of Change: Additional flow indicators and ammeters for the pump motors were installed on the ESW system.

Reason for Change: This change was made in order to allow for more proper and accurate evaluations of ESW System performance.

Safety Evaluation: This change was to provide reliable information in addition to existing indication for measuring system performance. It did not in any way degrade the safety function nor present any significant hazard consideration to the ESW System. As such, the components used for this change, namely ammeters, current transformers indicators, and flow transmitters are purchased as non-safety components. Where a connection was to be made so that a pressure boundary could be impaired (such as a connection of the flow transmitter to the flow element) a shut-off valve was installed, so that in case of a rupture or break within the non-safety related components, it can be easily isolated.

The seismic response of the panel due to the addition of ammeters and flow indicators was not significantly changed since the mass of the meters and indicators is small compared to the mass of the panel. The meters and indicators were mounted the same way as the seismically qualified equipments so that it was assured that they would not fall off during an earthquake and cause failure of seismic category I instruments.

DCR No. 937

Electric Fire Pump Controller

Description of Change: This change replaced the electric fire pump motor controller.

Reason for Change: The old controller was obsolete and spare parts were not available.

Safety Evaluation: This change is not safety related nor does it affect any safety systems.

DCR No. 939

Electric Fire Pump Test Valve

Description of Change: A throttling valve was installed in the electric fire pump test pipe to be used in throttling flow during pump testing.

Reason for Change: The previous test valve was a gate valve and was not appropriate for throttling service.

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

DCR No. 942

Isolation Valve Downstream of Nitrogen Accumulators
IR-002A, B, C, D

Description of Change: This change provided isolation valves downstream of the nitrogen accumulators for MSIVs CV 4413, 4416, 4419 and 4421.

Reason for Change: This change allows the accumulators to be isolated for maintenance.

Safety Evaluation: The isolation valves were added to seismic category I piping. The weight of each of the valves added was about two lbs. It was Engineering's judgement that the addition of the valves did not affect the seismic analysis. Also, the proposed change did not present significant hazards or considerations not described or implicit in the Safety Analysis Report.

DCR No. 947

Spent Fuel Pool Storage Expansion - Phase II

Description of Change: Phase II of the spent fuel pool storage expansion program increased the storage capacity to 1918 spaces including 20 spaces in multi-purpose storage racks. This phase removed all remaining GE-type spent fuel racks.

Reason for Change: This change was made to increase the spent fuel storage capacity in the DAEC spent fuel pool. Additional information is provided in the SER for Amendment 45 to Operator License DPR-49.

Safety Evaluation: A safety evaluation for this change was provided by the NRC Office of Nuclear Reactor Regulation in the Safety Evaluation Report for Amendment 45 to Operating License DPR-49, dated July 7, 1978.

DCR No. 948

Replace N₂ Drywell Penetration Inboard Isolation Valve

Description of Change: This change replaced CV-4371B with a manual stop check valve, V-43-214. In addition a drywell nitrogen header pressure transmitter/indication instrument loop was added.

Reason for Change: CV-4371B closed on an isolation signal and if not reopened before drywell N₂ dropped, could not be opened without a drywell entry to manually open it.

Safety Evaluation: The only change in this DCR relating to safety is the affect of this change on primary containment (drywell) integrity. Two penetrations are affected: X-22 and X-32C.

Penetration X-22 previously had three automatic isolation valves, one inside and two outside (in parallel) which meet the requirements of 10 CFR 50, Appendix A, Criterion 56(4). This DCR replaced the inboard valve with a self-acting (stop) check valve (globe type lift check design) which allows flow only in the inward direction. The check valve is permitted per Criterion 56(4). Periodic leak rate testing (LLRT) will be performed on the check valve to assure the leak tightness of this penetration. Additionally, periodic operability tests will be made to assure the check valve plug goes to the closed position to satisfy ASME B & PV Code Section XI, Article IWB-2000. Finally, the nature of the service requirements for this check valve are such that the valve should not operate more than just partially open (say 20% open) which will preclude seat damage from occurring if and when the valve is called upon to slam closed. On the basis of this discussion, it was concluded that the integrity of the primary containment was not decreased and the probability or consequences of a previously evaluated condition have not increased at penetration X-22.

Penetration X-32C was previously a 1", capped, spare drywell instrument penetration. This DCR utilizes this penetration for monitoring the drywell N₂ header pressure from outside the drywell. Normal N₂ system pressure is 90-110 psig. The new configuration² installed at this penetration is consistent with the design defined in FSAR Appendix G (page G.11-7) and Section 5.2.3.5.4 (especially the top paragraph on page 5.2-24). All the tubing, valves, fittings and instrumentation connected to the outboard side of the penetrations meet seismic category I requirements to assure integrity of the penetration. Additionally, the inboard side of the penetration has a 1/4" diameter sharp-edged orifice installed to provide a further restriction to flow in the unlikely event of failure of the outboard drywell pressure boundary components connected to the penetration. This penetration must maintain pressure boundary integrity only (the pressure instrumentation loop performs no safety-related, active function) and with the normal system pressure well above the 54 psig maximum Design Basis LOCA conditions, the pressure boundary integrity of this outboard tubing system is continuously assured. Further, the periodic ILRT will provide additional assurance of the integrity of this penetration.

The modifications made to penetrations X-22 and X-32C resulted in final configurations very similar or identical to other drywell penetrations and isolation valve configurations. Therefore, the potential for a "different type" of accident or malfunction has not been created beyond those previously evaluated.

The applicable Technical Specification Bases relate to primary containment integrity (see Tech. Spec. pages 3.7-35 through - 38). The margin of safety based on 2% of drywell volume/day of DBA leak rate has not been reduced. This is assured by 1) the pneumatic leak test performed on the system following installation, 2) the periodic LLRT performed on penetration X-22, and 3) the periodic ILRT performed on the entire primary containment.

It is, therefore, concluded (based on 10 CFR 50.59) that the changes in the DAEC facility described in DCR 948 do not create an unreviewed safety question. A Technical Specification change has, however, been identified and appropriate action has been taken to request a change to the Technical Specifications.

DCR No. 950

Wire Glass for Control Room

Description of Change: The DCR replaced the existing plate glass panes with wire glass.

Reason for Change: This change was made to comply with the Fire Hazards Analysis Report for DAEC, Section III.

Safety Evaluation: This change did not affect the safety analysis. No hazards or considerations not described or implicit in the safety analysis report were posed. This system does not interface with any safety related systems nor is it required for a safe shutdown. Replacing the existing plate glass by wire glass enhanced control room fire protection capabilities and thereby increased its reliability.

DCR No. 951

Flow Detector for Cooling Tower Performance Test

Description of Change: This DCR installed flow detectors in the cooling tower riser pipes and also installed the associated instrumentation.

Reason for Change: This change was to allow performance testing on the cooling towers.

Safety Evaluation: This change posed no unreviewed safety question since the change did not interfere with any safety system or interface with any safety function.

DCR No. 952

Modification of Recirculation Pump Trip (RPT) System

Description of Change: This change deleted four indicating lights on Panel 1C15 and four indicating lights on Panel 1C17, and installed two selector switches on panel 1C15 and two selector switches on Panel 1C17.

Reason for Change: This change was made to assist in the monthly surveillance testing of RPT systems A and B.

Safety Evaluation:

1. The main purpose of the indicating lights was to check the operation of the RPT sensors (Relays C71A-K8A, K8B, K10E, K10B, K8D, K8C, K10H and K10C) during monthly surveillance testing of the RPT logic. The deletion of indicating lights (C71A-DS8A, DS9B, DS8C, DS9D, DS8B, DS9A, DS8D and DS9C) did not in any way affect the capability to check the RPT sensors, since sensor operation can be checked by observing the relay dropout upon closure of control valve or stop valve.

2. Control Valve Logic

To avoid scram, the operator can test only one control valve at one time. The installation of key lock selector switch (key removable in the left position only) in parallel with each control valve sensor will, in fact, assist in the monthly surveillance testing of the RPT systems A and B. This can be accomplished by tripping one control valve logic output of one RPT system and testing the other control valve of the same RPT system.

3. Stop Valve Logic

To avoid scram, the operator can test two stop valves simultaneously. As such, the installation of key lock selector switch in parallel with each stop valve sensor was not required.

4. The replacement of four indicating lights with two new class 1E switches on each RPS panel 1C15 and 1C17 (Class 1E) did not affect the seismic response of the panel, because the difference in mass of two switches and four indicating lights was negligible as compared to the mass of the panel.

The switches were mounted in the same way as seismically qualified switches so that it can be assured that they will not fall off during an earthquake and cause failure of class 1E instruments.

5. The selector switches, General Electric Co. Type CR2940 were procured as safety related 'R' components, and were installed on RPS panels located in the control room.

Since the control room is a non-harsh environment, compliance to IEEE 323-1974 was not required. However, compliance to IEEE 344-1975 (or equivalent as the document has been issued after the DAEC was licensed) and the DAEC seismic requirement was essential.

The G.E. Co. CR 2940 switch is seismically qualified (FSAR Table M.3-1 and Table 5-M3.7-1) and has been used on various class 1E panels at the DAEC.

6. Leaving the key lock selector switch in the close (test) position will trip one control valve logic output of one RPT system. The closure of the second control valve of the same RPT system will cause RPT and reactor scram which will bring the plant to the safe shut down condition. As such, this deliberate bypass need not be continuously indicated in the control room.
(Ref: IEEE 279-1971 Section 4.13)

Conclusion:

Based on considerations discussed above, it was concluded that there are no unreviewed safety questions or technical specification changes and the criteria of 10CFR 50.59 provided the basis for making this modification.

DCR No. 953

Engineered Safety Features (ESF) Reset Control Modifications

Description of Change: The modification involved equipment that changes position from its safety or emergency mode to its normal mode following removal and/or manual resetting of the various isolation or actuation signals that triggered the equipment to be in its emergency mode. The result of such change in position could compromise the protective actions of the affected systems once the associated actuation signal is reset. Seal-in circuits were provided to the control circuits of the ESF systems affected to prevent change from emergency to normal mode upon removal and/or manual resetting of the actuation signal. The control circuits are activated by manual operation of a handswitch.

Reason for Change: This change was made to meet the intent of the NRC Bulletin 80-06 as committed to in Iowa Electric Letter LDR-80-159 dated June 17, 1980.

Safety Evaluation: The modification involved in this DCR affected plant control functions, but was limited to the ESF systems. The sole purpose was to prevent safety-related equipment from moving out of its emergency mode upon reset of an ESF actuation signal, thus enhancing the margin of safety and decreasing the probability of an accident. The design change, therefore did not introduce an unreviewed safety question.

DCR No. 954

Convert MO 1939 and MO 2029 to Throttle Valves

Description of Change: This change converted RHR heat exchanger inlet block valves MO 1939 and MO 2029 so they could be throttled in the closing direction.

Reason for Change: This change was made to allow a slow rate of reactor Temp. change during a cold shutdown mode of operation. Gradual closing of the block valves also prevents undue cycling in the RHR SW System.

Safety Evaluation: The block valves are safety related and so are the functions controlling the rate of change of the Rx Temp in conjunction with the operation of the bypass valves. The change did not in any way impair this significant function, but rather improved its controllability to maintain a constant slower cooldown rate. Overall, this design change did not present any potential hazard consideration not already described or implicit in the safety analysis report.

DCR No. 957

DAEC Security Project Isolation Zone Lighting

Description of Change: The change upgraded the security lighting at various locations.

Reason for Change: This change was made to ensure the security lighting requirements were met.

Safety Evaluation: This change was not a safety related change, nor did it affect the safety of other systems.

DCR No. 958

Cable Repair for MO 1027

Description of Change: This change placed a thermal strip in a field option box near the MO and labeled the box. All conductors to the MO were spliced in this box.

Reason for Change: The original conductors were scorched due to a steam leak.

Safety Evaluation: This change was not safety related nor did it affect the safety of other systems.

DCR No. 959

Scram Discharge Volume Drain Valve

Description of Change: This change removed CV-1867 from the scram discharge volume drain line, rotated the valve 180°, and rewelded the valve in place.

Reason for Change: This change was made when it was found the valve had been installed backwards during plant construction.

Safety Evaluation: This DCR resolved the non compliance of the backwards valve by removing and reinstalling the same valve in accordance with the original design. A 10CFR50.59 review was performed for this work. No change to the plant technical specifications was required and no unreviewed safety questions were created by this change.

DCR No. 960

Off-Gas Stack Flow Recorder FR-4133

Description of Change: The DCR replaced the existing recorder chart (scale 0-65 CFM) with a new chart of a higher scale (0-10,000 CFM).

Reason for Change: The new chart eliminated estimating off-gas stack flow that went beyond the 6500 CFM scale and ensured onscale readings at all times.

Safety Evaluation: The change improved the means of obtaining accurate off-gas stack flow for calculating stack release rate as per technical specification requirements. The replacement of the recorder chart and re-calibration of DPT-4133 did not affect nor interface with any safety related item.

DCR No. 961

Installation of a Manual Gate Valve in the RHR/Fuel Pool Cooling Cross Tie Line (GGB-23)

Description of Change: This DCR installed an 8 inch manual gate valve in the RHR/fuel pool cooling cross tie line (GGB-23).

Reason for Change: The purpose of this change was to prevent the repetition of water hammer events in this line which had occurred earlier.

Safety Evaluation: The change did not present significant safety hazards or considerations not described or implicit in the Final Safety Analysis Report. The addition of the valve reduced pressure surges and enhanced the reliability of the RHR/Fuel Pool Cooling System during the period when RHR pumps are required to be started.

It should be noted that the added, manual gate valve V-19-144 is in series with valve V-34-48. So in addition to safeguarding against waterhammer, V-19-144 also performs the same safety related functions that V-34-48 now performs; namely, FPC pressure boundary, RHR pressure boundary, and isolation. Thus, the safety of the RHR/Fuel Pool Cooling cross tie line is increased by system redundancy.

Finally, a seismic analysis was performed by Bechtel at Ann Arbor, Michigan. The results demonstrated that after the valve was installed, the cross tie line (BGG-23) met the design criteria for the system.

DCR No. 962

CRD Vent Line Down Stream of CV1859

Description of Change: The scram discharge volume vent line was cut and left open. After further evaluation the line was restored to its original configuration.

Reason for Change: NRC Bulletin 80-17 initially required the line be open to the reactor building. After further evaluation it was determined this was unnecessary.

Safety Evaluation: The change was not safety related and the system was restored to its original configuration.

DCR No. 966

Feedwater Pump Recirc. Lines' Pipe Supports on 6"-DBD-2-1 and 6"-DBD-2-2 at the Condenser Bay South Wall

Description of Change: The pipe support anchor bolts were changed from 5/8" hilti drop in bolts to 3/4" x 8 1/2" long hilti-kwik bolts.

Reason for Change: Pipe vibration had loosened the original anchor bolts.

Safety Evaluation: This design change did not affect or interface with any safety-related equipment. No unreviewed safety questions result from this design change.

DCR No. 967

Second Stage MSR High Load Valves/MO 9147, MO 9148 - Convert to Manual Open

Description of Change: This DCR converted second stage MSR high load valves MO 9147 and MO 9148 so they could be opened manually only.

Reason for Change: This change allows for proper warming of tube bundles when taking second stage reheat out of service or returning it to service at high power levels.

Safety Evaluation: The load valves are not safety related. The functions controlling the load valves are not safety related. The change in no way impaired any functions, but actually improved the operation of warming MSR tube bundles.

DCR No. 968

Fuel Pool System Reject Valve/MOV 3435- Remove Seal-In

Description of Change: This change removed the seal-in feature of MOV 3435 to make it a throttle valve.

Reason for Change: This change allows for controlled operation of the fuel pool system when draining.

Safety Evaluation: The valve is not safety related. The functions controlling the valve are not safety related. The change in no way impaired any function, but actually allowed for a more controlled operation of the fuel pool system when draining.

DCR No. 973

Relocate HS-1379 to 1C07

Description of Change: This change relocated control switch HS-1379 for off gas loop seal isolation valve to 1C07 in the control room.

Reason for Change: The previous location was in an area which becomes highly contaminated. Relocation of switch to the control room prevents unnecessary contamination of personnel and results in efficient and stable operation.

Safety Evaluation: The design change did not present hazard considerations not already called out in the safety analysis report, nor did it interface with safety related functions.

DCR No. 974

Control Rod Drive Handling Equipment Improvements

Description of Change: Several improvements to the CRD handling system were made per General Electric FDI #97-21771.

Reason for Change: These changes were made to improve system reliability and to reduce drive replacement time.

Safety Evaluation: This change did not affect the safety analysis. No hazards or considerations not described or implicit in the safety analysis report result from this change. This system is not required for a safe reactor shutdown.

DCR No. 975

Condensate Service System

Description of Change: This change removes the operation of solenoid operated control valves (CV-5228A&B and CV-5229A&B) of each pump from selector switch HSS-5222 and installs individual switches for each control valve within instrument rack 1C211.

Reason for Change: This change added flexibility in the selection of condensate service pumps and their associated solenoid operated control valves.

Safety Evaluation: The change did not affect the safety function of the system, therefore, did not constitute an unreviewed safety question. This change improved the flexibility of condensate service pump back-up selection. More reliable operation and reduced frequency of pump repair should result.

DCR No. 977

Feedwater Control Valve Disc Stack

Description of Change: The valve disc stacks in feedwater control valves CV-1579 and CV-1621 were changed from material made from inconel 600 to material made from 410 stainless steel.

Reason for Change: High velocity water flowing through the valves caused excessive wear in the lower section of the valve disc stacks. The new material reduces wear, is less expensive, and is easier to obtain.

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

DCR No. 984

Temperature Switches 3270A&B

Description of Change: Temperature switches 3270A&B, originally manufactured by Consolidated Controls Co., were replaced with new switches manufactured by Colt Industries.

Reason for Change: TS 3270A had broken and needed replacement. Because the original manufacturer was out of business, both switches were replaced by Colt Industries switches to maintain component homogeneity in the system.

Safety Evaluation: The replacement components are class 1E safety related. The operability of the temperature switches is essential to normal operation of the diesel generator because the heaters on which they are installed must function properly at all times and maintain the design temperature to keep the lube-oil warm and ready to circulate. The temperature switches are able to withstand any seismic event as well as environmental conditions at all times.

DCR No. 986

ASCO Solenoid Valves in the HPCI, RCIC, and Torus Rooms

Description of Change: The ASCO solenoid valves inside the HPCI, RCIC, and Torus rooms were replaced with ASCO NP solenoid valves qualified to the same environmental and seismic conditions except for a higher 40-year integrated radiation dose.

Reason for Change: In response to NRC Bulletin 79-01B, it was determined that these valves should be replaced with ones qualified to the expected radiation dose. The plastic disc holder in the exhaust pressurizer junction of the existing valves might have deteriorated at the expected radiation dose.

Safety Evaluation: All of the new ASCO solenoid valves are class 1E safety-related components and are qualified to expected post-LOCA environmental and seismic conditions. It is concluded that the new solenoids will improve the reliability of the system at LOCA conditions because they are qualified to a 40-year integrated radiation dose of 2.0×10^8 Rads, instead of 4.0×10^5 Rads. The new solenoids will not degrade associated valve operation and will not, if failed, endanger the health and safety of the public as described in the FSAR.

DCR No. 988

Suppression Chamber Spray Header Internal Supports Modification

Description of Change: The support system for the 4-inch spray header piping within the suppression chamber was modified.

Reason for Change: The support system modification was required to restore structural design margins in conformance with construction code requirements for a newly-defined load (i.e., pool swell/froth impact) which was unknown at the time of the original design.

Safety Evaluation:

1. The change does not change the technical specifications incorporated in the operating license.
2. The probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report has not been increased.
3. The possibility for an accident or malfunction of a different type than previously evaluated in the Final Safety Analysis Report has not been created. The reinforcements of the brackets are passive structural components and thus have no potential for creation of a new accident.
4. The margin of safety, as defined in the basis for any technical specification has not been reduced. The spray header support brackets are not addressed specifically in the technical specifications. The function of the suppression chamber ring girder is unchanged.

Suppression Chamber Catwalk Internal Support Modification

Description of Change: The internal support system for the suppression chamber catwalk was modified.

Reason for Change: The support system modification was required to restore structural design margins in conformance with construction code requirements for newly-defined loads that were unknown at the time of original design. These newly-defined loads include pool swell impact and drag loads, and SRV and LOCA submerged structure hydrodynamic loads.

Safety Evaluation:

1. The change does not change the technical specifications incorporated in the operating license.
2. The probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report has not been increased.
3. The possibility for an accident or malfunction of a different type than previously evaluated in the Final Safety Analysis Report has not been created. The new catwalk supports are passive structural components and thus have no potential for creation of a new accident.
4. The margin of safety, as defined in the basis for any technical specification, has not been reduced. The catwalk supports are not addressed specifically in the technical specifications. The function of the suppression chamber ring girder is unchanged.

Suppression Chamber RHR Relief Line Internal Support Modification

Description of Change: The fitup and welding for a modification to the suppression chamber RHR relief line internal support was completed. However, the actual support struts have been removed and are being stored for future installation after other suppression chamber modifications have been completed. The existing RHR relief line supports have been left in place until future suppression chamber modifications are completed.

Reason for Change: The modification of the internal supports was required to restore structural design margins in conformance with construction code requirements for the newly-defined hydrodynamic submerged structure loads unknown at the time of the original design.

Safety Evaluation:

1. The probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report (FSAR) has not been increased.
2. The possibility for an accident or malfunction of a different type than previously evaluated in the Final Safety Analysis Report has not been created.
3. The margin of safety, as defined in the bases for the technical specifications, has not been reduced. Operation of the residual heat removal system will be unchanged. The margin of safety for dynamic loads on the RHR pressure relief lines will not be reduced.
4. The added pipe attachment and suppression chamber attachment plate that were left installed as a result of this change do not effect the piping responses due to the loading conditions defined in the Final Safety Analysis Report.

DCR No. 992

Suppression Chamber Vent Header Deflector Installation

Description of Change: Structural steel "V"-shaped deflectors were installed to the underside of the suppression chamber vent header between the downcomers. The deflectors are welded to the vent header and the downcomer.

Reason for Change: The installation of the vent header deflectors was required to increase the capacity of the vent header to resist postulated dynamic impact loads due to pool swell.

Safety Evaluation:

1. The change does not change the technical specifications incorporated in the operating license. Downcomer submergence has not been changed.
2. The probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report has not been increased.

3. The possibility for an accident or malfunction of a different type than previously evaluated in the Final Safety Analysis Report has not been created. The vent header deflectors are passive structural components and thus have no potential for creation of a new accident.
4. The margin of safety, as defined in the basis for any technical specification has not been reduced. The function of the compression chamber vent header is unchanged.

DCR No. 993

Installation of Keylock Switches for the Performance of STP 43B003

Description of Change: 2-position keylock test switches were installed at panel 1C28 for the rod worth minimizer (RWM) and rod sequence control system (RSCS). One position of the switch indicates test (jumper installed or lifted) and the other position indicates normal, with the key removable at normal position only.

Reason for Change: The surveillance test procedure STP 43B003 for RWM and RSCS operability requires the installation of jumpers to prove the RWM operable and the lifting of two leads to test the RSCS prior to shutdown. Plant procedures require that only maintenance personnel install/remove jumpers and lift/restore leads. The keylock switches were installed to eliminate the problem of maintenance personnel availability onsite when needed to perform the RWM-RSCS operability tests (prior to pulling rods or after attaining 35% power on a shutdown).

Safety Evaluation: The addition of switches improves the method of installing & lifting jumpers performed by maintenance personnel during the testing of RWM & RSCS operability. This change does not affect the safety function of the system and therefore does not constitute an unreviewed safety question.

DCR No. 995

Reactor Water Cleanup Sample System Oxygen and Ph Analyzer

Description of Change: The inlet tubing for the reactor water cleanup (RWCU) sample system oxygen and Ph analyzers (AE-2751 and AE-2752) was relocated from the downstream side of PCV-2738 to the downstream side of valve V-27-138. This change also required that the newly routed inlet tubing be passed through sample coolers SC-4639 and SC-4640.

Reason for Change: The existing inlet tubing connection did not pass sufficient sample flow to the analyzer, thereby making oxygen and Ph monitoring difficult. This change relocates the analyzer inlet tubing connection to obtain sufficient pressure and flow to monitor oxygen and Ph in the sample flow.

Safety Evaluation: The change does not involve an unreviewed safety question. Based on engineering judgement, it was determined that this change does not increase the probability of occurrence of an accident or equipment malfunction important to safety. The change also does not worsen the consequences of a potential accident and does not reduce the margin of safety as defined in the basis for any technical specification. The tubing change is outside the seismic boundary of the safety system.

DCR No. 996

Torus Water Cleanup

Description of Change: Piping modifications were performed to allow the torus water to be cleaned by the condensate demineralizer and stored in the condensate storage tanks and/or the condenser hotwell during a refueling outage.

Reason for Change: Prior to this modification, there was no practical way to drain, clean, and store the torus water at DAEC if it was necessary to perform maintenance on the torus during a refueling outage. This modification eliminates the need to subcontract the torus water cleaning and transferring effort, which was expensive and time consuming.

Safety Evaluation: The installation of torus water cleanup system does not present any significant safety hazards or considerations not described or implicit in the safety analysis report. The torus water cleanup system will operate only when the plant is down for a refueling outage and when the fuel is removed from the reactor.

A removable spool piece is used as an intertie between the torus and torus water cleanup system. During the plant operation, this removable spool piece will be removed and a blind flange will be installed at the torus drain outlet. Since there will be no physical connection between the torus and the torus cleanup system during plant operation, this installation has no effect on the safety of the reactor operation.

A portion of the piping system of this installation runs in the proximity of existing Seismic Class I piping and equipment. Therefore, a seismic analysis for the piping (Seismic II over Seismic I) was performed by Bechtel, Ann Arbor. The piping installation was per the recommendations of that analysis.

DCR No. 997

Control Room Accident Monitoring Panel 1C-09

Description of Change: Panel 1C-09 was installed in the control room to be used for post-accident monitoring of plant safety parameters.

Reason for Change: This panel was required to be installed as a result of the post-TMI review detailed in NUREG 0737.

Safety Evaluation: The panel is a safety-related, Class 1E, Seismic Category I enclosure. Installation was in accordance with DAEC seismic requirements. The addition of this panel in the control room enhances the monitoring of the plant safety systems. Based on the above information:

1. The probability of an accident considered in the FSAR is not increased.
2. The margin of safety as defined in the technical specifications is not reduced.
3. The addition of this panel does not cause any interaction with associated class 1E panels.

It is concluded that this design change does not introduce any unreviewed safety question.

DCR No. 999

Emergency Diesel Generators, Mechanical Seals on the Lube Oil Circulating Pumps

Description of Change: The shaft seal on the emergency diesel generator lube oil circulating pumps was changed from a lip seal type to a mechanical seal.

Reason for Change: The existing lip seal type shaft seal had been a high maintenance item requiring periodic replacement to control leakage of lube oil through the seals. The oil leakage created a fire hazard.

Safety Evaluation: The lube oil circulating pump shaft seals are safety-related because they are part of the pump pressure boundary. However, the shaft seals are not required for emergency diesel generator operability and thus have no direct impact on the emergency diesel generators. Providing mechanical seals for the shaft seal provides better reliability, minimizes fire hazard due to oil leakage, and will result in less maintenance costs.

DCR No. 1000

Replacement of Reactor Water Cleanup Check Valve V-27-11

Description of Change: Check valve V-27-11 in the reactor water cleanup system discharge line to the main feedwater heater was replaced.

Reason for Change: The hinge pin seat of the check valve had eroded resulting in leakage from the hinge pin.

Safety Evaluation: This change does not affect the plant safety analysis. No hazards or considerations that are not described or implicit in the safety analysis report are created. Because the existing leaky check valve is irreparable, replacing it by a new valve improves system reliability.

DCR No. 1001

Emergency Diesel Generator Room Curbing

Description of Change: Curbing was installed in the emergency diesel generator rooms to contain possible oil spills from the diesel generators.

Reason for Change: Prior to this change, any oil leak or spill in the emergency diesel generator rooms would be unchecked and could flow throughout the room. Because sprinkler protection is only provided over the area of each diesel engine, the curbs were installed to contain possible oil spillage within the area covered by sprinklers.

Safety Evaluation:

1. Safety Hazards - The safety hazards presented by this change are limited to the possibility that personnel may trip or stumble upon encountering the installed curbs. This design change considers this potential and minimizes the hazard by placing the curbs such that the possibility of a fall is not increased beyond that which previously existed before the change.
2. Seismic Considerations - The welding of four #4 steel concrete reinforcement bars to each engine support frame, Seismic Class 1 structures, does not affect the ability to withstand a seismic event. The engine support frame is considered a passive system as it is bolted and grouted in the concrete floor.

DCR No. 1004

Standby Filter Unit Flow Transmitter Drift

Description of Change: The change involves taking parallel contacts off of each flow-controlled relay in the start circuits of both standby filter units such that the availability of auto-start will effectively be doubled.

Reason for Change: Prior to this change, flow transmitters FT 7320A&B have tended to drift upscale and prevent the automatic start of the standby filter units. With this change, both FT 7320A&B will have to malfunction together before the auto-start of either standby filter unit is impaired.

Safety Evaluation: The routing of wires from Div. 1 to Div. 2 safety systems is covered by Reg. Guide 1.75. Otherwise, the probability of occurrence of an accident or malfunction to equipment previously evaluated in the FSAR is not increased but actually decreased, and the margin of safety as defined by Tech. Spec. is not only maintained, but improved.

Reg. Guide 1.75 describes the separation criteria for handling Class 1E circuitry of which the use of an isolating device such as a relay is acceptable. The existing relays 94-038A in division 1 and 95-038B in division 2 are identified as the isolating devices by which we are meeting the requirements of the Reg. Guide. The relays are separated by a fire barrier and the wires used to interlock the contacts run through the barrier via a penetration that was sealed after installation. The integrity of the Class 1E system has been maintained and any postulated failure in one system will not have a detrimental affect on the operation of the other system.

The entire failure mode can be summed up as either the relay contacts close inadvertently or open inadvertently. In either case, the start/stop logic is not affected in a non-conservative way.

DCR No. 1005

Reactor Vessel Bottom Head Temperature Indication

Description of Change: A cable was run between reactor vessel bottom head temperature element TE 4576F and panel 1C04 to provide bottom head temperature indication on temperature indicator TI 2713 located on panel 1C04 in the control room.

Reason for Change: Due to a recent Technical Specification change, the reactor vessel bottom head temperature must be recorded every 15 minutes during reactor heatups and cooldowns.

Safety Evaluation: This change is not a safety-related change and does not affect the safety of the other systems.

DCR No. 1010

Modification/Replacement of Electrical Terminal/Junction Boxes and Conduit Systems in Harsh Environments

Description of Change: SEMCO PR-855 silicone RTV foam was applied in existing Class 1E, NEMA Type 1 junction boxes, terminal boxes, and field option boxes inside the drywell and steam tunnel or in the conduits at the box end. Also, certain boxes and terminal blocks for which environmental qualification could not be sufficiently established were replaced.

Reason for Change: Because these boxes in the drywell and steam tunnel were not leaktight, the silicone foam was applied to prevent the transportation of liquid from the subject boxes to the essential equipment. The foam application and the replacement of boxes and terminal blocks resulted from NRC Bulletin 79-01B.

Safety Evaluation: The change does not present any significant hazards or considerations not described or implicit in the Safety Analysis Report. The implementation of a liquidtight conduit system and the substitution of qualified terminal blocks improves the capability of the systems involved to perform their intended safety functions.

DCR No. 1018

Diesel Generator Governor Drain Petcock

Description of Change: The existing diesel generator governor oil drain petcock was replaced with a 1/4-inch NPT, carbon steel plug.

Reason for Change: The petcock was replaced by the plug because it is located in a position that is easily bumped and could cause the inadvertent draining of the emergency diesel governor oil.

Safety Evaluation: This design change does not create an unreviewed safety question. This design change increases the reliability of the emergency diesels by replacing the emergency diesel governor oil drain petcock with a plug, further assuring the governor oil will not be inadvertently drained.

DCR No. 1020

Main Generator Stator Cooling Alarm Modification

Description of Change: A low-low flow alarm for the main generator stator cooling water flow was added to the data logger. In addition, the upper low flow alarm setpoint was increased.

Reason for Change: Modifications to monitor the stator cooling water flow were recommended by the main generator vendor.

Safety Evaluation: This change is not safety-related and its implementation does not interfere with the operation of any other safety system.

DCR No. 1021

Replace Pressure Switches on Safety Relief Valves

Description of Change: The existing pressure switches on the safety relief valves were replaced with seismically and environmentally qualified, Class 1E, pressure switches.

Reason for Change: Although the indication provided by these switches in the control room is considered to be a non-safety function, qualified switches were installed to fulfill the requirement for reliable indication.

Safety Evaluation: This change modifies a non-safety system. However, this change affects the Safety Related Classification List due to requirements that the equipment be Class 1E seismic to insure positive indication in the control room.

The failure of the pressure switches will not jeopardize the manual or automatic operation of the valves. The existing temperature sensors in the discharge of each safety/relief valve will provide information to enable the operator to close a stuck open valve from the control room.

Failure of the switches or tubing would not jeopardize the safe shutdown of the reactor. These switches are not located on the primary pressure boundary and any leakage (which would only occur when a safety/relief valve was open) would be contained.

Based on considerations discussed above, it can be concluded that there are no unreviewed safety questions, and the criteria of 10 CFR 50.59 provides the basis for making this modification.

DCR No. 1023

Condensate Storage Tank Level Elements, LE 5218 and LE 5219

Description of Change: Suspension wires from the electrode holder for LE 5218 and LE 5219 were moved to the adjacent existing pull box where the connections were made. The conduit between the pull box and the electrode holder were sealed with RTV to prevent moisture entering the pull box.

Reason for Change: Cable connections were changed to eliminate the problem of moisture in the electrode holder causing the wire splice to short to ground or to each other. In the past, this short circuiting problem had caused LE 5218 and LE 5219 to fail to operate.

Safety Evaluation: Changing the cable connections from electrode holder to P. Box will eliminate the problem of short circuiting and improve the reliability of the system. This way, it will not jeopardize the safety of any other system or personnel present in the area.

DCR No. 1026

Turbine Building Roof Scuppers

Description of Change: The two existing 3-inch diameter scuppers on the west side of the north and south walls of the turbine building were replaced with 4-inch by 12-inch scuppers. Six 4-inch by 12-inch scuppers were installed on the east wall of the turbine building parapet.

Reason for Change: These scuppers were installed to meet flood protection requirements for the emergency diesel generator air intakes.

Safety Evaluation: This design change does not create an unreviewed safety question or a change to the Technical Specification. The addition of these scuppers assures that water will not accumulate on the roof and, therefore, assures the integrity of safety-related equipment in the turbine building.

DCR No. 1028

Off Gas Isolation Time Delay

Description of Change: This DCR added a 20 second time delay to the off gas system isolation signal provided by PDIS 4182.

Reason for Change: Pressure spikes caused by the starting of process pumps 1P-105A&B were causing nuisance trips of the off gas isolation logic circuitry. The change eliminated the problem.

Safety Evaluation: This system is not safety related and the implementation of this DCR did not affect the integrity of any other safety systems in a non-conservative manner.

DCR No. 1032

Radwaste Solidification Air Supply

Description of Change: This DCR added air lines to provide a filtered air supply from the plant service air system for the radwaste solidification system.

Reason for Change: The air supply was necessary as part of the solidification system installation.

Safety Evaluation: This is not a safety related system and will not affect the safe shutdown of the plant. No potential radiation release points have been created in the radwaste building.

DCR No. 1038

Replacement of Deficient INC
(International Nuclearsafeguards Corporation)
Mechanical Snubber S/N 08211

Description of Change: This DCR replaced snubber S/N 08211 made by INC with a spare pacific scientific PSA-1 mechanical snubber.

Reason for Change: The previous INC snubber S/N 08211 had been tested per NRC Bulletin 81-01 and had been found to have excessive breakaway friction.

Safety Evaluation: The changes provided by this DCR did not present significant hazards or considerations not described or implicit in the Safety Analysis Report. Specifically, the changes did not:

1. increase the probability or the magnitude of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report (FSAR),
2. create the possibility of an accident or malfunction of a different type than any condition previously evaluated in the FSAR,
3. reduce the margin of safety as defined in any Technical Specification, or
4. involve an unreviewed safety question.

DCR No. 1039

Replacement of Deficient PSA Mechanical Snubbers 17,
DBA-5-SS-31, DBA-5-SS-38, and DCA-6-SS-50

Description of Change: Snubbers 17, DBA-5-SS-31, DBA-5-SS-38, and DCA-6-SS-50 were replaced with similar mechanical snubbers which were tested and found acceptable.

Reason for Change: The previous snubbers had been tested per NRC Bulletin 81-01 and had been found to have impaired freedom of movement.

Safety Evaluation: The changes provided by this DCR did not present significant hazards or considerations not described or implicit in the Safety Analysis Report. Specifically, the changes did not:

1. increase the probability or the magnitude of an accident or malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report (FSAR),
2. create the possibility of an accident or malfunction of a different type than any condition previously evaluated in the FSAR,
3. reduce the margin of safety as defined in any Technical Specification, or
4. involve an unreviewed safety question.

DCR No. 1040

RCIC Auto-Suction Switchover from the CST to the Suppression Pool

Description of Change: This change provided automatic RCIC pump switchover from condensate storage tank (CST) suction to suppression pool suction on CST low water level.

Reason for Change: This change was made to comply with NUREG 0737, Item II.K.3.22.

Safety Evaluation: The change did not present any possible hazards or considerations not described or implicit in the Safety Analysis Report. The operator is no longer required to manually switch over on low CST water level. The operator will no longer be distracted by the manual switchover and the possibility of inadvertent failure to switchover is eliminated. Credit is not taken for RCIC as a safety system. This change did not leave any unreviewed safety question.

DCR No. 1051

Replacement of RCIC Globe Valve MOV 2515

Description of Change: This DCR replaced the 4"-600# globe valve with a 4"-1500# ASME Section III Class 1 globe valve.

Reason for Change: The valve was replaced after a pin hole leak in the valve body of the previously installed valve was found.

Safety Evaluation: The replacement of the existing 4"-600# carbon steel valve with a 4"-1500# stainless steel valve did not present any safety hazards or considerations not described or implicit in the Final Safety Analysis Report. Since the existing valve had a pin hole leak, replacing it by a new valve with higher pressure rating code class and better material improved system reliability.

Section B
SPECIAL TESTS

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

B. Special Tests

This section contains summaries of those special tests conducted at the plant during the calendar year 1981. The Special Test Procedures, which governed the performance of these tests, were reviewed by the DAEC Operations Committee and were found not to present any unreviewed safety questions.

SpTP No. 75 Demonstrate and Document DG Loading Sequence and Magnitude

The purpose of this procedure was to verify the capability of the standby diesel generators to meet the requirements of revised FSAR Table 8.4-1, which included the increased size of the emergency service water pump motors.

This special test was performed on May 14, 1981.

SpTP No. 86 Exterior Instrumentation for SRV In Plant Test

The purpose of this procedure was to direct the installation of exterior instrumentation on the torus shell, support columns, support reinforcements, HPCI and RCIC piping. The instrumentation was used to collect data during the SRV in-plant test conducted after the 1981 refueling and maintenance outage.

The instrumentation for this special test was installed during the 1981 refueling outage. This SpTP will be completed when this instrumentation is removed.

SpTP No. 87 Checking for Obstructions to Steam Flow Through PSV-4400 and Associated Piping

The purpose of this procedure was to ensure valve PSV-4400 and the associated piping were not obstructed.

This special test was completed on March 21, 1981.

SpTP No. 88 Locating Leaking Jet Pump Plugs

The purpose of this procedure was to locate and reposition leaking jet pump plugs.

This special test was performed on April 19, 1981.

- SpTP No. 89 Draining "B" Recirculation Loop
- The purpose of this procedure was to drain the "B" recirculation loop between the recirculation pump discharge valves and jet pump plugs.
- This special test was performed on April 17, 1981.
- SpTP No. 90 Leak Rate Determination for N2D Safe-End
- The purpose of this procedure was to determine the approximate leak rate through the N2D safe-end.
- This special test was completed on April 17, 1981.
- SpTP No. 91 In-Plant Safety Relief Valve Discharge Test
- The purpose of this test was to measure the structured response of the DAEC suppression chamber (torus) and related structures to hydrodynamics loads resulting from actuation of safety relief valves. The data will be used to confirm the analytical model of torus response.
- This special test was completed on June 14, 1981.
- SpTP No. 92 Fire Water Supply During Circ. Pit Draining
- The purpose of this procedure was to define a method of ensuring fire water pumping capability during periods when the circulating water pit is drained.
- This special test was completed on May 9, 1981.
- SpTP No. 93 Lost Parts Analysis Core Flush
- The purpose of this procedure was to establish sufficient core flow to dislodge the missing strongback cotter key parts prior to performing CRD friction testing. This provided assurance that the parts would not interfere with control rod motion during operation.
- This special test was performed on May 15, 1981.
- SpTP No. 94 Condenser Performance Test
- The purpose of this procedure was to measure condenser performance as a function of plant power level and circ water temperature.
- This special test was completed on July 9, 1981.

SpTP No. 95

MOV 2515 Integrity Verification

The purpose of this procedure was to verify that MOV 2515 would not allow flow to the CST while in the shut position.

This special test was completed on June 18, 1981.

SpTP No. 96

Recirculation Pump Suction and Discharge Valve Leakage Isolation Test

The purpose of this procedure was to backseat the recirculation pump suction and discharge valves to test for packing leakage.

This special test was performed on July 24, 1981.

SpTP No. 97

Recirculation Pump Vibration Test

The purpose of this procedure was to determine the magnitude of recirculation pump "B" vibration as a function of pump speed.

This special test was completed on August 21, 1981.

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 1981.

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 concerning relief valve and safety valve failures and challenges for calendar year 1981. 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 safety valve failures or challenges during 1981. There was one (1) relief valve challenge on July 15, 1981 which resulted in three relief valves opening and closing following a turbine trip. This event is described below.

<u>Date</u>	<u>Event Description</u>
3-21-81	PSV 4400 was opened one (1) time while performing Special Test Procedure #87.
5-27-81	At 0253 hours, relief valves PSV 4400, 4401, 4402, 4405, 4406 and 4407 were successfully opened and closed during surveillance testing.
5-29-81	At 0053 hours, relief valve PSV 4400 was successfully opened and closed during the followup surveillance test for the valve pilot replacement.
6-1-81	At 2010 hours, relief valve PSV 4406 was successfully opened and closed during the followup surveillance test for the valve pilot replacement.
6-6-81	At 0200 hours, relief valve PSV 4406 was successfully opened and closed during the followup surveillance test for the valve pilot replacement.
6-14-81	PSV 4400 was opened fourteen (14) times while performing Special Test Procedure #91.
6-18-81	At 0448 hours, relief valve PSV 4405 was successfully opened and closed after valve pilot replacement.
7-15-81	A reactor scram occurred at 2017 hours due to a turbine trip while performing a turbine overspeed trip mechanism test. Following the scram PSV 4401, PSV 4405 and PSV 4407 lifted to control reactor pressure. The valves operated properly and seated on closure.
10-29-81	At 0245 hours, relief valve PSV-4407 was successfully opened and closed during the followup surveillance test for the valve pilot replacement.