Duane Arnold Energy Center

1984

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

IE24

SECTION A - PLANT DESIGN CHANGES

This section contains brief descriptions of and reasons for plant design changes completed during the calendar year 1984, and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR, Part 50.59(b).

The basis for inclusion of a Design Change Package (DCP) in this report is site closure of the package in the calendar year of interest. It is noted that in certain cases, portions of these DCPs have received partial closure in previous years.

DCP No. 804 — Radwaste Centrifuge Torque Recorders

Description and Basis for Change: The Radwaste Centrifuge Torque Recorders loop consisted of a force transducer, transducer exciter demodulator, signal converter and signal converter recorder. The signal converter was a Transmation type 230T with an input and output signal that mismatched the rest of the loop. Using a plant modified version of RIS model SC-1300 signal transmitter the correct signals are now fed to the recorders, thus making full use of the recorder scale.

Summary of Safety Evaluation: This change was not safety-related nor did it affect a safety-related system.

DCP No. 806 — Radwaste Conveyer Drain Sump

Description and Basis for Change: The Radwaste Conveyor floor drain sump pump discharge line was continually becoming plugged. The velocity of the fluid in the pump discharge pipe was too low to adequately transport spent resin to the sludge tank.

The 7-1/4" pump impeller was replaced with a 7-13/16" impeller. This change resulted in an increased fluid velocity. Also the spring was removed from each check valve (V-37-34 and V-37-36). This change allowed the valves to partially open a greater amount.

Summary of Safety Evaluation: This change was not safety-related nor did this change affect a safety-related system.

DCP No. 882 — APRM/LPRM Power Supply Modules

Description and Basis for Change: SIL 295 recommended changing resistor R3 on all integrated circuit power supplies (ICPS) to a 2 watt 150 Kohm resistor instead of maintaining the previous 1 watt rating due to power dissipation failures.

SIL 302 described failures of the rectifying diodes in these ICPSs. Replacement of all IN4004 diodes made by Transitron alleviates failure possiblities due to these components. The faulty resistors and diodes on the 80 power supply modules for the LPRM/APRM circuits were replaced. All R3 resistors were replaced and a random check (20-25% of the modules) was made on the R4 resistors. If R4 was not within +/- 20% of 10 Kohm, it was replaced. All modules checked for R4 were identified and all failures were noted. All IN4004 diodes having a capital "T" stamped on the diode body were replaced and then checked for excessive ripple per GE's recommendation Neither IN4006 diodes nor IN4004 diodes without the "T" stamp were required to be replaced.

Summary of Safety Evaluation: These changes were not safety-related. These changes did not degrade any safety-related system functions. The replacement of the resistors and diodes improved the availability of the APRM/LPRM system. Any failures of the ICPSs will cause indicated flux levels to be higher than actual flux levels and a plant scram may occur. Power failure does not pose a safety or equipment hazard.

DCP No. 933 Containment Pressure and Level Indication

Description and Basis for Change: Control Room indication of containment pressure and torus water level was installed to satisfy NUREG 0578, Enclosure 3 to Reference 1, commitments made in Iowa Electric Letter LDR-80-03 dated January 3, 1980. Indication of containment hydrogen concentration was installed under a separate DCP.

Summary of Safety Evaluation: This change was safety-related. This instrumentation supplements previously existing plant (Control Room) indication of specific plant parameters, and utilizes existing instrument process connections. Because the instruments are for indication only, no plant control functions were added or affected. Therefore, the passive nature of these instruments had no adverse effect on plant safety. Their installation did not constitute an unreviewed safety question.

DCP No. 949 — Change Window #43 Engraving at 1C08 Panel

Description and Basis for Change: Annunciator engraving of window #43 on Control Room panel 1C08 was changed from "4KV Ess Bus Paralleled or Auto/Man SW Not in Auto" to "4KV Bus Auto Transfer Inop." as requested by NRC. (Reference discussion between K. Meyer, Iowa Electric and T. Kavern, NRC via telephone memo dated May 21, 1980.)

Summary of Safety Evaluation: Although this change was safety-related, it had no bearing on safety since no equipment or system was functionally affected.

DCP No. 956B — Airborne Radiation Monitor System Replacement

Description and Basis for Change: The existing airborne effluent radiation monitoring system was supplemented with an extended range system to satisfy NUREG 0737, Item II.F.1, Attachments 1 and 2, and Regulatory Guide 1.21. This change also provided for remote indication of airborne effluent radiation activity in the control room and the chemical and radiological laboratory.

Summary of Safety Evaluation: This change was not safety-related. This change supplemented the existing airborne effluent radiation monitors with an extended range system and transfered effluent radiation monitoring and indication from one panel to another. The existing radiation monitoring system continues to be operational. The addition of this system has upgraded the radiation monitoring of the plant and had no adverse effects on plant safety. The probability of an accident considered in the FSAR was not increased. The margin of safety as defined in the technical specification was not reduced. In conclusion, the proposed change did not present any significant hazards or considerations not described or implicit in the Safety Analysis Report.

DCP No. 983 — Installation and Testing of Concrete Expansion Bolts on CRD System

Description and Basis for Change: This change covered the testing, inspection, repair, and replacement of pipe support concrete expansion bolts and their associated baseplate(s) on Seismic category I piping. The changeout from shell to wedge type was based on the resolution of Item 4 of NRC Bulletin IE 79-02 and was in response to NRC Bulletin IE 80-17.

Summary of Safety Evaluation: This change was safety-related. This action did not represent significant departure from the hazards or conditions described in the DAEC FSAR. The confidence level of the safety of the system was increased beyond that stated or implied in the DAEC FSAR.

DCP No. 1016A Computer Room HVAC System Modification, Short-Term

Description and Basis for Change: The computer room HVAC system modification (short-term) was completed to maintain an operable environmental condition for the new computer system until the 1982 refueling outage. The modification consisted of adding to the control building an air handling unit (with distribution plenum) and support stand, humidifier, condenser, and all associated piping, tubing, valves, fittings, drains, and controls. The air handling unit (with distribution plenum) and support stand, humidifier, and associated controls are located in the computer room. The condenser-receiver-compressor package is located on the control building roof with associated controls.

Summary of Safety Evaluation: The system is mechanically arranged and located in such a way that the portion of the system which is located inside the control building is also within the boundary of the computer room. In the event of failure of the system or supporting structure, no equipment or structure which is necessary for the safe shutdown of the plant would be affected.

All electrical/control changes associated with this change were nonsafety-related and any potential failure would have no consequence on any safety-related system.

The combustible material added to the plant as a result of this DCR had a negligible effect on the overall weight of combustible materials listed in the fire hazards analysis. In conclusion, this modification was not safety-related and presented no unreviewed safety questions.

DCP No. 1016B Computer Room HVAC Modification, Long-Term

Description and Basis for Change: The computer room HVAC system modification (long-term) was provided to maintain an operable environmental condition for the new computer system and the Secondary Alarm Station (SAS). It consisted of decommissioning portions of the original HVAC system, demolition of existing ceiling, installation of an air handling unit with a supply and return air duct system, electric heating coil, humidifier, condensing unit, ceiling, as well as all associated piping, tubing, valves, fittings drains, controls, lighting, 480V power panel, support platform and ladder, and relocation of conduits and associated cables which interfered with the support platform. The air handling unit and humidifier are located above the computer room ceiling. The air cooled condensing unit is located on the control building roof with the associated controls. This long-term modification for the computer room was designed to carry the heat generation load of the new VAX computer and a future computer.

Summary of Safety Evaluation: The computer room modification (long-term) was not safety-related and presented no unreviewed safety questions. The system is mechanically arranged and located in such a way that the portion of the system which is located inside the control building does not extend beyond the boundary of the computer room into the control room. The supporting structure for the air handling unit is designed as Seismic Category 1. This will protect the equipment located in the computer and control rooms from impact during a seismic event. No equipment or structure required for the safe shutdown of the plant was affected. The combustible material added to the plant as a result of this change had a negligible effect on the overall weight of combustible materials listed in the fire hazards analysis. This change did not create a possibility for an accident or malfunction of a different type than evaluated previously in the FSAR or subsequent submittals. This change did not increase the probability of occurrence of an accident or malfunction of equipment previously analyzed in the FSAR or subsequent submittals. This change did not increase the consequences of any accident or malfunction of equipment previously analyzed in the FSAR or subsequent submittals. This change did not affect the technical specification requirements.

DCP No. 1047 — Scram Discharge Volume Diverse Level Instrumentation

Description and Basis for Change: This change resulted from Iowa Electric's commitment to the NRC (reference letter LDR-81-176) to install diverse level instrumentation for the Scram Discharge Volume (SDV) Instrument Volume (IV) in order to meet the requirements of Safety Criterion 3 (reference NRC Generic SER dated 12-1-80) as clarified by NRC Generic Letter 81-18.

The NRC Generic SER further clarified acceptable means of complying with Safety Criterion 3 as follows:

> "(1) With respect to single failures (random) provide sufficient redundancy in the automatic scram level instrumentation to meet the single failure criterion on each instrumented portion of the SDV."

With respect to common-cause failures, Duane Arnold selected "Alternative 1" which is restated below for convenience:

> (a) provide additional (or substitute) level-sensing instrumentation for the automatic scram function to include diversity as well as redundancy. The diversity should, as a minimum, be achieved by level sensors that employ different operating principles for measuring the water level;
>
> (b) for the instrumentation selected, demonstrate how common-cause failures; such as those identified in the forward to IEEE 379-1977 will be considered.

Four thermally actuated liquid level switches were installed on the SDV instrument volumes in addition to the existing Magnetrol float switches LS-1861A, B, C, and D. These switches are redundant to the existing float switches, perform the same scram functions, and provide the required redundancy and diversity. These switches operate on an entirely different principle than the existing float switches. They detect the difference in the heat transfer properties of various liquids and gases as a function of the temperature difference between a heated and reference sensor.

Another two (2) of these new level switches were installed on the north SDV instrument volume and are redundant to the existing float switches on the south SDV instrument volume, LS-1861E and LS-1861F. The new switches perform the same "block rod withdrawal" and "alarm" functions, respectively. The addition of these two (2) switches was not specifically required by the NRC but was deemed necessary in order to detect adequate hydraulic coupling between the two SDV instrument volumes which are coupled by a relatively small (2") and long (approximately 120') drain line. This responds to the possibility of the north SDV instrument volume accumulating a significant amount of water before the alarm or rod block switch on the south side would be actuated. Therefore, this addition improves overall system safety for the SDV system.

Each new level switch except LS-1862F operates a dedicated interposing relay in a control room panel. The relays LS-1862A through D supply inputs to RPS trip logic, "SDV Hi Water Level CRD Trip" annunciation and the plant computer. The interposing relay for LS-1862E provides inputs to the CRD rod block function and the plant computer. LS-1862F provides direct input to "SDV Not Drained" annunciator in control room panel 1C05 and as such did not require any interposing relay.

Summary of Safety Evaluation: The additional level instrumentation meets the requirement of "acceptable compliance" with "Safety Criterion 3" of the NRC "Generic Safety Evaluation Report" with regard to single random failures as well as common-cause failures.

The level-sensing instruments are safety-related, Class 1E and meet the requirements of IEEE 323-1974 and IEEE 344-1975. The instruments mounted directly on the instrument volume did not require any additional instrument piping or valves, and thereby, avoid the possibility of certain human errors such as instrument valves misalignment after testing (reference LER 81-039). The volume available for scram discharge water was not changed by addition of these instruments. The new scram level sensors were located approximately twelve (12) to eighteen (18) inches below the existing scram level sensors to allow for a possible one second delay on the response of the new sensors.

A complete seismic analysis was performed for the Duane Arnold SDV piping including the six new level sensors. The analyses show that the modified instrument volume meets seismic Category I requirements.

The level sensors were not required to be designed to conform with ASME Boiler and Pressure Vessel Code, Section III. Justification for exemption from these requirements was provided in Iowa Electric memo NG-82-1256.

FCI Kapton Polyimide cable used for connecting level sensing elements to level switch electronics did not conform to all the requirements of IEEE 383-1974. From the test reports and thermal aging analysis made available to Iowa Electric Engineering by FCI, it was concluded that the FCI cable was suitable for the intended application.

The installations of relays, fuses and associated cables did not have any impact on fire hazard analysis. Cable tray fill was kept within the allowable limit of NFC and FSAR.

Evaluation showed that the integrity of level switch boxes, LS-1862A through F, mounted on local racks would be maintained after a design basis earthquake.

This design change did not involve any unreviewed safety questions. This determination was based on the following:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR was not increased. The additional level sensors that employ different operating principles provide diversity. The new level sensors are not adversely affected by hydrodynamic forces or flow characteristics and provide reliable instrumentation needed to detect water buildup and to scram the reactor.

2. The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR was not created. Rather the new SDV level instrumentation provides sufficient redundancy and reliability to the automatic scram level instrumentation and meets the single failure criterion on each instrumented portion of the SDV.

3. The margin of safety, as defined in the basis for technical specifications, was not reduced since the new level sensing instrumentation provides diversity as well as redundancy for the automatic scram function.

DCP No. 1050 Turbine Building Crane Stops

Description and Basis for Change: The permanent mechanical crane stops on the south end of the turbine building for overhead crane are located south of the emergency diesel generator air-intake duct which is safety-related ductwork. It was requested that the crane stops be relocated to prevent the crane from hitting the ductwork. An electrical interlock was selected over relocating the permanent crane stops as this provides the same protection and allows flexibility in crane operation.

Two temporary mechanical stops were installed north of the ductwork. Proper measurement was taken to assure that the temporary mechanical stops were located to maintain at least 6" clearance between the crane and the ductwork after the crane is stopped at its furthest extension to the south. The temporary mechanical stops serve as backups to the electrical interlocks. Electrical interlocks were achieved by installing two limit switches, one on either side of the south end of the crane, and a keylock switch to bypass the limit switches whenever needed. These switches were connected to the circuit which controls the southbound movement of the bridge. As the northbound movement of the crane is controlled by a different circuit, the crane operator is still able to move the crane in that direction. If the crane is ever needed south of the temporary mechanical stops, the temporary mechanical stops will have to be moved, then the crane operator can bypass the limit with the aid of the keylock switch.

Summary of Safety Evaluation: This change is not safety-related. The previous installation of new emergency diesel generator air intake ductwork which is safety-related, created the possibility of the turbine building overhead crane being able to damage the ductwork. Implementation of this change allows for safe operation of the crane by the crane operator. No unresolved safety question resulted from implementation of this change.

DCP No. 1058B Hotwell Transfer System

Description and Basis for Change: During a plant outage the condenser hotwell is used to store condensate from the torus, reactor vessel, etc. Previously, to transfer condensate from the condenser hotwell to the condensate storage tanks, a condensate pump was required. The increased start/stop cycles and operation at low flows in this service was detrimental to the life of the condensate pumps.

The hotwell transfer system provides a means of transferring condensate from the condenser hotwell, through the condensate demineralizers (for cleanup) and to the condensate storage tanks without the use of a condensate pump.

The hotwell transfer pump, 1P-258, which is located in the turbine building, on the 734'-0" floor between the condensate service jockey pump, 1P-11, and the condensate storage tank heat exchanger, 1E-15, takes suction from the hotwell through the suction line to condensate pump 1P-8A. The pump suction line, 6"-HBD-202, has a removable spool piece which is replaced with a blind flange when the plant is in operation. This isolates the transfer system and ensures that there is no air inleakage into the condensate pump suction. There is a recirculation line around the pump to assure minimum flow through the pump and a paddle-type flow switch in the pump suction to energize a low flow warning light on the local panel. The pump discharge is routed to the condensate pumps discharge header, GBD-1. A gate valve and blanking plate provide isolation. A ring spacer is inserted when operation of the system is required.

Summary of Safety Evaluation: This change was not safety-related. The hotwell transfer system did not present any significant safety hazards or considerations not described or implicit in the final safety analysis report. The hotwell transfer system is only used when the plant is in cold shutdown. A removable spool piece and blanking plate are used to isolate the transfer system from the condensate system when the plant is in operation.

None of the piping added is in the proximity of any existing seismic Class I piping or equipment. Therefore, a seismic analysis of the piping was not required. The piping was supported in accordance with article 121 of ANSI B31.1, "Power Piping Code".

DCP No. 1075 TIP Area Shielding and Access Control

Description and Basis for Change: The area around the TIP drive machines is subject to high contamination and high airborne radioactivity if a TIP detector is pulled into the machine. Fire hazard requirements precluded the previous wood and plastic access control in the area.

A permanent shield wall and roof slab was built in the TIP area to provide shielding and access control. The shield wall was constructed of high density grout-filled masonry and the slab of reinforced concrete on deck and structural steel. Access control to the new room is provided by a lockable door, and access control to the nearby CRD filter room provided by a lockable gate. Ventilation to the new room is supplied by a louver in the door and the existing 500 CFM negative pressure in the TIP room. ARM monitor RE9176 remains inside the enclosure for airborne radiation detection. ARM hardware, steam tunnel cooling unit boxes, and associated conduit and instrument lines were repositioned above the slab to ease access to them. No other equipment was moved. Penetrations were made in the wall and slab as required for other existing piping and conduit. Access to the repositioned equipment is over the enclosure roof and a permanent ladder is provided at the south end. Additional lighting and a page unit are inside the enclosure to support TIP drive maintenance.

The enclosure roof is suitable for temporary storage of equipment. The roof is designed for a 250 psf uniform live load and a maximum permissible singlewheel load of 1500 pounds at any location.

Summary of Safety Evaluation: As a minor structural addition required only for shielding and access control, the enclosure is not in itself a safety-related item. Because the completed structure will contain no safety-related items, the structure is not classified as a seismic Class I structure. The wall and slab were designed as seismic Class I. The door, screen gate, ladder, and handrail attached to the structure are not in a position to affect any safety-related equipment and are therefore not designed on a seismic Class I basis.

The affect of the addition of the enclosure upon existing structural components was checked. The enclosure had a negligible affect on the structural integrity of the existing building and did not have a significant or unacceptable impact on the existing margin of safety.

Since the added structure does not house safety-related items and is not important to plant operation, no unreviewed safety questions exist. The structure presented no different type of accident and did not increase the probability or consequences of an accident, or reduce the margin of safety.

DCP No. 1083 PCIS Valve Access Platforms

Description and Basis for Change: Access to the four PCIS solenoid and valve lines at El. 733' 3-1/4" in the torus room was previously made by temporary scaffolding. Personnel safety, fire hazard, and seismic considerations required the replacement of the temporary scaffolding with permanent steel platforms. Four steel platforms were, therefore, designed and detailed. In addition, to facilitate valve operation, four 1/2" lines were shortened and the valves in these lines repositioned.

Summary of Safety Evaluation: This change was safety-related. The platforms were designed as seismic Class I. They were designed to withstand a seismic event without failure or excessive deflection. The platforms were designed as trusses (axial deformation only) and calculations determined that elastic deflection under design load would be nil. Clearance around the platform ensures that any elastic deformation of the platform would not affect other equipment. The piping being shortened was reattached to existing seismic Class I supports.

1. The platforms and revised piping are structurally secure. There was no increase in the probability of occurrence or magnitude of the consequences of an FSAR evaluated accident.

2. The platforms are equivalent to any other seismic Class "2-over-1" structural component. No new type of accident can be caused because of this design change.

3. The margin of safety of any equipment in the area was not reduced. The added components have negligible effect on the existing structure and did not affect the design margin of safety. This change did not present any unresolved safety questions.

DCP No. 1102 Switchgear Addition to Bus 1A2

Description and Basis for Change: This change consisted of the following items:

1. Installation of new switchgear cubicle and breaker on 4160 volt bus 1A2.

2. Installation of the breaker control switch and indicating lights on control panel 1C08.

3. Interconnection of the breaker and control switch.

The installation of new switchgear on existing 4160 volt bus 1A2 provided a reliable and sufficiently sized power source for production well no. 4.

Summary of Safety Evaluation: The equipment installed by this change performs no safety-related function (reference FSAR Section 10.10 and DAEC System Description No. 8). Physical interfaces exist which impact Class 1E equipment. This required those portions of the change to be safety-related.

1. Control cable is qualified for Class 1E installations due to the physical interface wwith Class 1E raceway. The control cable was installed in accordance with cable and wire installation procedures. The cable was routed in Nondivisional and Divisional 2 raceway such that cable separation criteria was maintained.

2. The installation of the control switch and lights on control panel 1C08 did not affect the seismic response of the panel because the additional mass of the switch and lights is negligible as compared to the mass of the panel. The control switch was mounted in the same manner as seismically qualified switches to ensure that it would not fall off during an earthquake and cause failure of Class 1E instruments.

3. The addition of the new switchgear to the switchgear room did not add combustibles which require additional fire protection requirements. The existing fire hazard analysis for the switchgear room was not changed.

DCP No. 1105 Radwaste Evaporator Bottoms Tank to Floor Drain Sludge Tank

Description and Basis for Change: This change provides a cross-tie from bottoms tank recirc pump 1P-137 to pipe HBC-106 which empties into the floor drain sludge tank 1T-62B. Two remote operated plug valves and associated controls were added as part of the change.

This change allows the contents of the radwaste evaporator bottoms tank 1T-60 to be transferred to 1T-62B. This is beneficial for several reasons: 1) should tank 1T-60 be filled with resins that are too "hot" for solidification, there was previously no satisfactory way to dilute the resins before solidification. This installation eliminates the problem; 2) this installation also allows a more thorough analysis of the resins before solidification; and 3) this installation reduces processing time for filling up 1T-60.

Summary of Safety Evaluation: This system is not safety-related and does not affect the safe shutdown of the reactor. New piping for this system was done to the latest criteria concerning radwaste piping, Regulatory Guide 1.143. This Reg. Guide deviates from an FSAR commitment to install nuclear Class 3 piping in the radwaste system. However, based upon the safety significance of the system, the NRC concluded in Reg. Guide 1.143 that the installation of non-nuclear piping in the radwaste system does not reduce the margin of safety. Because this system ties into ANSI B31.7, Class 3 piping, the interface points were modified to reflect the original construction code.

This change did not increase the probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. This change did not create the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR. This change did not compromise the margin of safety as defined in the basis of any Technical Specification.

DCP No. 1130 Radwaste Centrifuge Bypass Valves Replacement

Description and Basis for Change: In the past, the centrifuge bypass valves, CV-4064 and CV-4065, plugged frequently. This required manual rodding out of the valves and resulted in personnel radiation exposure. Plugging of these valves also resulted in an inability to transfer resins for blending prior to shipment offsite. The existing centrifuge bypass valves were 3/4" diaphram valves. They were replaced with 1-1/2" plug valves, with new pneumatic operators and limit switches. The work included electrical disconnection and removal of existing valves, installation of new valves and associated field fabricated piping, and reconnection of electrical wiring. Increasing the valve size should eliminate the plugging at these valves.

Summary of Safety Evaluation: This change was not safety-related. The piping and valve changes did not present any significant safety hazards or considerations not described or implicit in the Final Safety Analysis Report. Replacement of the centrifuge bypass valves to alleviate plugging should result in lower personnel radiation exposure.

DCP No. 1144 Replace Damaged Safety-Related Cables

Description and Basis for Change: Damaged portions of safety-related cables were replaced. The various cables included drywell radiation monitor output to control room recorder, inboard RHR shutdown cooling isolation valve (MO-1908), RHR reactor head spray isolation valve (MO-1900), solenoid valve "B" on FW excess flow check valve (V-14-1), RCIC steam drain pot level switch, and core spray inject valve position valves and testable check valve solenoid valves. Associated flexible conduit and terminal boxes were replaced or added.

Summary of Safety Evaluation: The "like-for-like" replacement of cables for safety-related circuits did not compromise the initial safety evaluation of the circuits in question. The installation of junction boxes to facilitate possible future replacements did not affect the continuity of the circuits. Installation in accordance with applicable conduit, terminal box, cable, and wire installation procedures ensured integrity equal to that of the original circuit design.

Since the cables were routed in conduits, or trays less than 30% filled, no combustibles were added to the FHA calculations. Penetrations necessary for routing conduits were resealed in accordance with the accepted procedures. Therefore, the FHA was not impacted.

Throughout the installation ALARA precautions were maintained.

In conclusion, there is no impact on Licensing, the FSAR or the Technical Specifications. These changes did not make an accident/malfunction more likely or worsen the consequences of an accident/malfunction. These changes did not make it possible for a new type of accident/malfunction to occur. These changes did not reduce the plant margin of safety. In accordance with 10 CFR 50.59 and per this safety evaluation, there were no unreviewed safety questions.

DCP No. 1148 Replace Motors in RHR Air Cooling Units

Description and Basis for Change: The two existing motors which drive the fans on RHR air cooling units 1V-AC-11 and 1V-AC-12 were replaced with two new qualified Westinghouse motors which meet environmental qualification requirements. The motors operate the supply fans in the room cooling units in the RHR and core spray pump rooms. The new motors are physically interchangeable with the old motors. Existing electrical power feeds and control schemes were used for the new motors. This change is in response to IE Bulletin 79-01B and its supplements.

Summary of Safety Evaluation: This change was safety-related. The new motors which replaced the existing motors are one-for-one replacements and are physically and functionally interchangeable with the existing motors. The new motors are qualified to NUREG 0588-1979, Category I for Duane Arnold application and, therefore, did not degrade the system. All work was performed in compliance with Duane Arnold technical specifications.

The replacement of the existing motors with new qualified motors with adequate radiation qualification documentation did not change the operation of the RHR air cooling units and therefore:

a) 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, did not increase.

b) The possibility of an accident or malfunction of a different type which was previously discussed in the FSAR was not created.

c) The margin of safety was not reduced by the replacement of the motors.

DCP No. 1151 Replacement of Nonsafety-Related Cables

Description and Basis for Change: Damaged portions of nonsafety-related cables were replaced. The various cables provide annunciation for MSR drain tank level, HPCI room temperature to steam leak detection logic, position indication and solenoid valves for reactor head vent valves, and drywell temperature indication. Associated flexible conduit and terminal boxes were replaced or added.

Summary of Safety Evaluation: The replacement of the cables did not affect the safe operation of the plant and was not safety-related. In accordance with 10 CFR 50.59, there were no unreviewed safety questions. ALARA was considered and the work was completed during a shutdown. The cables were routed in conduits and/or trays. The tray fill did not exceed 30% or previous fill levels if above 30%, therefore the FHA was not impacted. One tray was found 36% filled before this work. As the tray is located in the drywell, which is filled with nitrogren while the plant is operating, the FHA was not affected. The tray is only 8 feet long, and all the cables in this tray are nonsafety-related.

DCP No. 1152 Reactor Level Indication Power Supply s

Description and Basis for Change: Previously, in the event of a total loss of all offsite AC power and both diesel les generators, no reactor vessel water level indication was available in the control room which indicated below a level of +158" TAF. This indication is needed to monitor water s, level during the initiation and operation of any Emergency Core Cooling System (as an example, see Integrated Plant Operating Instruction (IPOI), Vol. C-2.0, Sec. 5, Rev. 1, dated January 4, 1982). This was not previously available since all of the control room level indicators with scales below +158" TAF were powered by plant AC.

Changes were implemented to have level indicator LI-4539 and level transmitter LITS 4539 (Division 1) powered by the RCIC Inverter (Division 1) and level indicator LI-4540 and (Division 2) powered by the HPCI Inverter (Division 2). A minimal amount of load was added to these inverters (refer to Iowa Electric load study). The Division 2 cable was run from panel 1C03 through the cable spreading room to panel 1C05. The Divison 1 cable was routed from panel 1C04 through a 3/4" conduit to tray 1S5A and down another 3/4" conduit to panel 1C05. These are short runs and required a minimal amount of cable pulling. This change was installed during an outage since the modification included circuitry for RCIC, HPCI, and vessel level indication.

Summary of Safety Evaluation: This change was safety-related. This change significantly improved the reliability and availability of level instruments LI-4539 and LI-4540 in that they are now powered from a more reliable power supply. The probability of occurrence or magnitude of any accident or malfunction was decreased, specifically in the case of total loss of AC power, and the possibility of a different type of accident or malfunction occurring was not created. The margin of safety was increased by the more reliable power supply.

Per the load study performed, the additional load on the HPCI and RCIC systems did not reduce the margin of safety. The additional load is practically insignificant to that available from the Topaz inverters. Each inverter has a maximum specification of 125 VA and the level indicators have added only 3 watts to the existing maximum load of 20 watts.

The work was performed in the control room so there were no ALARA concerns. There was no impact on the UFSAR, Technical Specifications, Fire Hazards Analysis, or any other licensing documents. No unreviewed safety questions were identified.

DCP No. 1158A Acid Feed to MUD System (Foundation/Containment)

Description and Basis for Change: A previous change request was initiated to install an acid supply storage tank and associated piping for the Makeup Demineralizer system. This tank, to be located east of the turbine building, requires a concrete foundation with a containment dike in case of tank rupture. To allow immediate installation of the foundation/containment (to avoid cold weather concreting), this change request was issued authorizing the necessary civil/structural work. Specifically, the scope of this change included: 1) excavation and fine grading required for foundation construction, 2) construction of all reinforced concrete foundation and containment dike, including embeds and grating, and 3) application of an acid-resisting coating on the interior surfaces of the containment structure.

Summary of Safety Evaluation: The acid tank foundation/containment structure is a nonsafety-related structure located where it cannot affect safety-related equipment. Specifically:

1. The use or operability of safety-related systems during a DBA is not affected by the structure. This modification had no affect on the initiation or consequences of an UFSAR-evaluated accident.

2. Should the acid tank rupture, the containment structure was designed to hold all expected spillage until such time as spilled acid could be disposed of. In any case, should spilled acid breach the containment structure, no safety-related systems are near enough to be affected. No new accidents are expected due to the installation or failure of this modification.

3. The structure itself was designed with an acceptable margin of safety per the UFSAR and all applicable codes. The weight of the structure and acid tank is relatively light, well distributed, and sufficiently distant from the turbine building to add only negligibly to the turbine building foundation soil pressure. Existing margins of safety were negligibly, if at all affected. Iowa Electric calculations are available.

DCP No. 1160 Install LSA Box Trash Compactor

Description and Basis for Change: The existing compactor (55-gallon hydraulic press) presently in use was inefficient and unable to adequately support an outage. As a result, a large inventory of trash was stored in the radwaste building. This inventory of trash exceeded the levels of combustibles permitted by the fire hazard analysis.

In order to meet Iowa Electric's objective of minimizing the inventory of trash during subsequent outages, the existing compactor was supplemented with an LSA box trash compactor capable of keeping up with the expected amount of LSA trash. Conduit and electrical wire were installed to supply electrical power from nonessential MCC.

Summary of Safety Evaluation: The dry waste to be compacted consists of air filters, paper, rags, contaminated clothing, etc. and results from routine operations and maintenance in contaminated areas. This dry waste is handled manually because of low radioactivity content or minimal contamination levels.

Major emphasis was placed on the handling and storing of compacted waste. The safety objective is that handling and monitoring of the dry radioactive waste meet the requirements of 10 CFR 50, Appendix A, Criteria 60, 63, and 64 while maintaining radiation exposure to operating and maintenance personnel "as low as is reasonably achievable" (ALARA) in accordance with 10 CFR 20.

Five areas of concern were addressed in this safety evaluation to determine if the installation involved an unreviewed safety question. Each area of concern has been addressed below with the above safety objective in mind:

1. Increase of Ambient Radiation Levels in the Truck Bay

   An increase of ambient radiation levels was expected in the immediate area of the compactor during compactor operation. However, enforced administrative controls and operational procedures minimize personnel exposure to within the guidelines of 10 CFR 20. "Closing off" the compactor area to through traffic, temporary shielding, or additional procedural steps assure a greater degree of control from unnecessary radiation exposure. Radiation exposure to personnel operating the LSA box compactor is the same as to personnel operating the existing 55-gallon drum hydraulic press.

2. Potentials of Accidental Release of Radioactive Materials to the Environs and In-plant Area

The 55-gallon hydraulic press's exhaust is ducted to a HEPA filter in the radwaste building exhaust system for airborne radioactive material removal during compacting. The LSA box trash compactor's airborne radioactive materials removal is identical to the 55-gallon hydraulic press. During compacting operation, a self-contained air evacuation/filter system consisting of a fan and a HEPA filter provides negative pressure in the LSA box trash compactor and removes radioactive materials that may be released.

The potential for accidental radioactive material release is greatest during the handling of waste in preparation for compacting. The radioactive material release rate to the environment must not exceed 10 CFR 20 concentration limits for unrestricted areas at the DAEC site boundary.

The offgas retention building exhaust (the location of the LSA box trash compactor) and the radwaste building exhaust (the location of the 55-gallon hydraulic press) are monitored for radioactive airborne release by the airflow monitor in the reactor building stacks. In the event of a high radiation signal detected in the exhaust from the offgas retention building, the reactor building exhaust fans will shut off in the same manner as if detected in the exhaust from the radwaste building, thus mitigating the release of radioactive materials into the environment.

3. Storage of Compacted Dry Radwaste

The LSA box trash compactor's compaction factor is much greater than the 55-gallon hydraulic press's compaction factor ultimately reducing storage requirements during normal operation. A reduced volume of stored radwaste may reduce the level of personnel exposure to radiation because of smaller and fewer areas required for storage. In addition, the trash inventory may be reduced to levels permitted by the fire hazard analysis.

4. Radwaste Container Integrity

The containers that accommodate the recommended LSA box trash compactor are acceptable for storage and disposal and designed in accordance with 49 CFR 173. Each container has been tested for structural integrity and each container is provided with a certificate of compliance for fabrication and testing. These containers meet the same requirements as the 55-gallon containers used for the 55-gallon hydraulic press. Therefore, the failure rate of a container is not increased.

5. Floor Loading Due to Compactor

A calculation was performed on the "floor area" in the offgas retention building. The calculation indicated that the additional floor loading due to the compactor, filled container, and forklift is acceptable.

The five areas of concern in the safety evaluation indicate that the installation of a LSA box trash compactor is nonsafety-related and the results of the five areas of concern indicate that this change did not involve an unreviewed safety question.

DCP No. 1164 Radwaste HIC and Resin Processing Modifications

Description and Basis for Change: The previous radwaste method of processing spent resin was solidification in Hittman 85 cubic foot liners. Due to limitations of this procedure, the radwaste processing system was unable to adequately support a refueling outage. Modification of the system was needed to add the ability to process spent resin into high-integrity containers (HICs).

This change adapted the radwaste solids processing system to use HICs, supplementing the existing solidification system. The following changes allowed HICs to be used by the existing 55-gallon steel drum filling, storage, and handling system:

1. Recommendation of a HIC supplier

2. Purchase and installation of a HIC capper on the existing drum capping machine

3. Purchase and installation of a grapple to lift HICs on the 1.5 ton radwaste crane

4. Addition of compressed air tubing to the shock mounts on radwaste hoppers

5. Addition of a flexible collar on the downcomer pipes from radwaste hoppers

6. Purchase and installation of level switches at radwaste filling stations 2 and 3

7. Purchase and installation of additional fire hose section at hose station 30

Changes 2 and 3 were required due to the physical differences between HICs and steel drums. Change 4 was made to reduce radiation exposure ALARA to personnel inflating the shock mounts on radwaste hoppers. The mounts can be inflated outside the shield labyrinth. Change 5 was necessary to direct the resin into the smaller fill opening on HICs. Change 6 was necessary to signal when the HIC is filled with resin. This cannot be seen with the drum viewers since the HIC walls are opaque and the fill opening is obscured by the downcomer pipe and collar. Change 7 was included based on review of the fire hazards analysis.

Summary of Safety Evaluation: This change was not safety-related. The resin processing modifications adapted the General Electric supplied spent resin processing system to use HICs. The capacity to use 55-gallon steel drums is maintained. The only change is that dewatered resin is placed in HICs instead of steel drums. The previous safety analysis (Updated FSAR) applies since the basic resin processing system is unchanged. Shielding design and semi-remote operation of the system limit radiation exposure to within 10 CFR 20 values. Inadvertent or accidental release of radioactive materials is limited to within the 10 CFR 20 guideline values by operating procedures and containment and storage facilities.

DCP No. 1184 ADS/Core Spray and RHR (LPCI) Interlock Setpoints

Description and Basis for Change: The Emergency Core Cooling System (ECCS) logic on all BWRs contains a low pressure ECCS pump/ADS interlock. Once the ADS system has received both high drywell pressure and low water level signals and the ADS timer has "run-out", the logic requires a pressure signal from the discharge side of one of the low pressure ECCS pumps (either a low pressure core spray (LPCS) or a low pressure RHR pump (LPCI) is running) to initiate ADS. Technical specification changes as requested by Iowa Electric and approved by the NRC are stated below:

Function	Pressure Switches	Old Value	New Value (approved by NRC)
Initiate ADS	Core Spray Pump Discharge Pressure Switches	100+/-5 psig	145+/-20 psig
Initiate ADS	RHR (LPCI) Pump Discharge Pressure Switches	100 psig (min.)	125+/-25 psig

Summary of Safety Evaluation: This change was safety-related. The new lower limit for the ADS/Core Spray discharge pressure interlock is 125 psig whereas Core Spray suction line relief valve setpoint is 100 psig. Therefore, the possibility of false indication of running core spray pump at the time of ADS initiation, due to pressurization of the suction line on valve leakage is precluded. The new upper limit of the setpoint (165 psig) is below the core spray pump head at the maximum runout flow conditions. Thus, the switch setpoints indicate a running pump for all LPCS flow conditions.

Since the RHR suction line can be pressurized to a maximum value of 68 psig and the new lower limit for the ADS/RHR (LPCI) discharge pressure interlock is 100 psig, the possiblity of false indication of a running RHR pump due to pressurization of suction line is precluded. In the event of low reactor water level signals and high drywell pressure, all the four RHR pumps start and pressurize the discharge lines to a value approaching the shutoff head of the pump (248 psig). Since the presence of both the signals (low reactor water levels and high drywell pressure) is a precondition for the ADS to operate and the upper limit of the setpoint (150 psig) is well below the RHR (LPCI) discharge line pressure, the switch setpoints ensure RHR (LPCI) pump running permissive for ADS initiation.

Based on the considerations discussed, it was concluded that this change did not involve any unreviewed safety questions. It was also determined that the changes:

1. Did not involve any increase in the probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously evaluted in the UFSAR;
2 Did not create the possibility of an accident or malfunction of a type different than any previously evaluated in the UFSAR;
3. Did not reduce the margin of safety as defined in the basis of any technical specification.

DCP No. 1187 Mode Switch Shutdown Scram Reset Permissive

Description and Basis for Change: SIL-344, Rev. 1, described an event at an operating BWR. With the reactor mode switch in RUN, a Reactor Protection System (RPS) motor-generator set failed, causing a half scram. When RPS power was restored, the operator observed that the mode switch shutdown scram reset permissive was annunciated (which is abnormal while in the RUN mode). The same event occurred at Duane Arnold on September 27, 1982. The problem was determined to be a relay "race" resulting in the reset permissive alarm. A relay race is a flaw in the relay logic which is dependent on the sequence of a particular relay operation.

A set of normally open K16A and K16B contacts were added to trip system A and B to solve this problem in all mode switch positions. The following will refer to trip system A, but the same applies for trip system B.

A. Circuit operation after power loss: In the original circuit if the power was lost, then restored with the mode switch in the "RUN" position, K16A might energize before K17A. If this happened, the K16A 1-2 contacts would prevent K17A from energizing. With K16A energized and K17A de-energized the bypass circuit around mode switch contacts 9-9C was established (even though 9-9C are closed in "RUN") and the annunciator that indicates the bypass was activated. As the reactor was brought down and the mode switch rotated through the "START UP" and "REFUEL" positions, its contacts 2-2C were closed. Closure of 2-2C energized K17A, which de-energized K16A, which in turn opened the bypass around the mode switch 9-9C contacts. However, if the operator were to rotate the mode switch rapidly through the "START UP" and "REFUEL" positions, it was possible that the switch might be placed in the "SHUTDOWN" position without energizing K17A. This would place the circuits in the configuration that would prevent the de-energization of the manual scram relays (15A and K15C) and, thus, prevent a scram from being automatically initiated when the mode switch was moved to the "SHUTDOWN" positon.

B. In normal circuit operation: When the mode switch (S1) is in the "SHUTDOWN" position, its contacts 1-1C are closed causing K16A to be energized. When K16A is energized, its contacts 1-2 are open preventing K17A from energizing. In this configuration K16A is energized and K17A is de-energized, and a bypass around the 9-9C contacts of the mode switch is formed by contacts 3-4 of K16A and contacts 3-4 of K17A.

The bypass provides for the energization of the manual scram relays K15A and K15C after the protection system is reset. When the mode switch is moved to the intermediate position between "SHUTDOWN" and "REFUEL" mode, switch contacts 1-1C open and 9-9C remain open. When this happens, it is necessary to have a seal-in circuit to keep K16A energized in order to maintain a bypass circuit around the 9-9C contacts until they close in "REFUEL" position. The seal-in circuit to keep K16A energized was provided by the K17A contacts 1-2 and the normally open K16A contacts added in this change. When the mode switch is moved to "REFUEL" position, its contacts 9-9C close and the bypass is no longer needed. At the same time contacts 2-2C close and energize K17A. When K17A energizes, its 1-2 contacts open. As contacts 1-1C of the mode switch are also open, relay K16A is de-energized. With K17A energized and K16A de-energized, both sets of relay contacts that formed the bypass around mode switch contacts 9-9C are open, removing the bypass. By adding normally open K16A and K16B contacts to each trip system the modified circuit implemented this solution.

Summary of Safety Evaluation: This change was safety-related. This change did not constitute any change in the technical specifications or any unreviewed safety questions previously evaluated in the UFSAR. The margin of safety, as defined in the basis of technical specifications, was not reduced by this modification.

The purpose of this change was to eliminate the relay "race" problem which had occurred at Duane Arnold. By adding a spare set of normally open K16A contacts in series with the K17A contacts, this problem was eliminated. This modification did not have any impact on the Fire Hazard Analysis since no new installation of any component was involved. Two cable terminations were changed in the same area of panel 1C15. These cable terminations complied with Iowa Electric standards.

The probability of the occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR was not increased. The possibility of an accident or malfunction of another type was not created. This modifcation makes RPS more reliable.

DCP No. 1188 Diesel Fire Pump Magnetic Switch Replacement

Description and Basis for Change: Possible pitting and/or fusing of the starting contactors for the diesel-driven fire pump was identified. The starting contactors are located on the same mounting bracket as the starting magnetic switches, which provide current-draw coils in parallel with the contactors.

The original vendor for the magnetic switches, Cummins Diesel, recommended replacement of the existing contactors and switches with their newly designed switches. The new switches have a coil which is capable of 3 amperes more current draw. This has reduced the starting current experienced by the contactors because of the parallel arrangement described above. The scope of work was as follows:

A. Field removal of both existing starting circuit magnetic switches and contactors, A and B.

B. Field installation of a new mounting bracket to the block above the cranking motor.

C. Field installation of both new starting circuit magnetic switches, A and B.

Summary of Safety Evaluation: This change was not safety-related. This change did not interface with any safety-related system and did not affect any safety support system. Because this change did not permanently affect the operation of the fire protection system, no operating procedure revisions were needed. This temporarily placed the diesel-driven fire pump out of service. Immediately upon taking the pump out of service, field personnel took the necessary actions to comply with plant technical specifications and related test procedures. The operators were alerted as to the status of the diesel-driven fire pump before commencing and after completing all work.

DCP No. 1197 Replacement Shielding at Torus Room Hatches

Description and Basis for Change: These modifications involved replacement shielding between the torus room and ground floor, reactor building, in hatch openings RB-3 and RB-5. The modifications were divided into three parts: pickpoints above Hatch RB-5, replacement shielding at Hatches RB-3 and RB-5, and baffle shields at the utility ports in the hatch covers.

Prior to the 1983 refueling outage, four 36-inch concrete plugs were located in Hatch RB-5, and a hinged 3/16"-plate cover was located above Hatch RB-3. To facilitate access to the Torus, the plugs in RB-5 were required to be removed. A series of pickpoints were designed above RB-5 to allow pulling and removal of the plugs. The concrete plugs were extremely heavy and difficult to handle and a lighter method of shielding was requested. At the same time, a study determined that the 3/16"-plate shielding at RB-3 did not meet NUREG 0737 requirements. To solve both problems, two sets of plate steel hatch covers meeting NUREG 0737 shielding and UFSAR loading requirements were designed for RB-3 and RB-5. A utility port was designed into each hatch cover to facilitate temporary work in the torus room during operation. A baffle shield was required at each port to ensure acceptability of each hatch cover per NUREG 0737.

Summary of Safety Evaluation: All modifications covered by this change were nonsafety-related. This was based on the following:

1. The replacement steel hatches perform no safety function necessary to the operation or safe shutdown of the plant exce . to remain intact, under load, in the event of a DBE. As such, the hatches are no different thany any other nonsafety-related item above a safety-related item. This "seismic II over I" situation required that the hatches be designed as seismic Category I structures, but did not make the hatches safety-related. Accordingly, the steel hatches and frames were designated as nonsafety-related structures requiring seismic integrity.

2. By definition, the hatch cover shielding must be in place during operation; thus the hatches are expected to be moved only while the plant is in safe shutdown. Since movement of the covers neither affects the safe shutdown condition of the plant nor crosses over spent fuel, no special NUREG 0612 analysis was required. The pickpoints installed above RB-5, used only during shutdown to lift the hatches, also do not affect plant operation and were also designated as nonsafety-related.

In summary, the replacement hatch covers are nonsafety-related seismic Category I structures, and:

1. The covers are structurally equivalent to the plugs they replace. They do not increase the probability of occurence or the magnitude of any FSAR-evaluated accident.

2. The covers and framing have no effect on the operation or the integrity of the plant. They are not expected to cause any different type of accident.

3. The modifications themselves are designed to have an acceptable margin of safety per UFSAR 3.8. Since they replace the much heavier concrete plugs, the existing structural margins of safety in the reactor building are unchanged.

DCP No. 1198 Intermediate Range Monitor Noise Reduction

Description and Basis for Change: A significant number of half scrams had occurred which were attributable to a high noise level signal in the Intermediate Range Monitors (IRMs). The implementation of General Electric SILs 46, 47, 47-S1, and 210 was requested.

SILs 46 and 47 reported occurrences of 100MHz parasitic oscillations in the IRM voltage preamplifier, the amplifier attenuator and the inverter modules. These parasitic oscillations manifest themselves in range correlation difficulties or inverter balancing problems. The true IRM signals can be masked and range correlation may be impossible. In SIL 47-S1, amplifier attenuator resistor values were updated to avoid confusion when trouble-shooting and as a design improvement.

In accordance with General Electric SILs 46 and 47, high permeability ceramic beads were installed on selected transistor leads to eliminate module parasitic oscillations. Since the beads are conductive, they were fixed in place so as to not touch the transistor bodies or printed wiring board conductors. The beads were fixed in place using epoxy or RTV adhesive. After installation of these beads, the module was checked for signal performance. The beads cause the transistor leads to appear as high impedances at the parasitic frequency and effectively prevent oscillation.

SIL 210 reported occurences of carbon biasing resistors generating excessive noise on the input of the voltage preamplifier which is used, in conjunction with the Mean Square Root Voltage/Wide Range Monitor (MSV/WRM), to measure neutron flux density over a range of three decades.

As recommended by SIL 210, voltage preamplifier carbon-type resistors were replaced with metal film resistors. This type of resistor exhibits minimal electron agitation characteristics and results in less electrical noise.

Summary of Safety Evaluation: A review of 10 CFR 50.59 was performed for these safety-related equipment modifications. No changes to the plant technical specifications were required and no unreviewed safety questions were created by the changes. The changes did not affect the Fire Hazard Analysis, FSAR, or ALARA.

DCP No. 1202 Main Steam Line Hanger Repair

Description and Basis for Change: Due to an apparent temporary overload, main steam line hanger MS-8 failed completely; the structural steel pulled loose from the wall. Opposite hanger MS-9 also experienced significant yielding in the structural steel bracket. Hangers MS-8 and MS-9 are located on the main steam line downstream from control valves CV-1 and CV-4, respectively, adjacent to the high pressure turbine.

A knee brace hanger was designed to replace the previous hanger design. The new hanger is considerably stronger and is designed to withstand loads even greater than the temporary loads encounter d just prior to the previous hanger failure.

Summary of Safety Evaluation: Hangers MS-8 and MS-9 are located on a nonsafety-related and nonseismic sections of the main steam lines. The design change did not effect the operation or function of the main steam lines.

DCP No. 1204 Appendix R Modifications to the Automatic Fire Suppression System

Description and Basis for Change: These modifications, in conjunction with other changes, were required to bring the plant in compliance with 10 CFR 50, Appendix R, Section III.G, and the exemptions therefrom granted by the NRC. This change provided for the addition of one deluge sprinkler system and two wet pipe sprinkler systems in Fire Zones 3-B (corridor and waste tank area - hatch), 4-A (HVAC heat exchanger and chiller area), and 12-B (control building HVAC equipment room), respectively. The design of the sprinkler systems were in accordance with the National Fire Protection Association (NFPA) standards.

Summary of Safety Evaluation: These changes were not safety-related and the results of the safety evaluation indicated that they did not involve an unreviewed safety question. These modifications included additional fire suppression systems similar to existing fire suppression systems provided in other plant areas. These changes provided an additional measure of protection for safety-related equipment and components which are part of safe shutdown systems.

Sprinklers are considered passive mechanical components. Therefore, the inadvertent activation of the sprinklers is not postulated to occur concurrent with a loss-of-coolant accident (LOCA). Consequently, these fire suppression systems have no impact on control room habitability under LOCA conditions.

To evaluate the effects of inadvertent actuation of the fire suppression systems installed by this change, the potential impact on safety-related systems due to spray and flooding was considered. Under normal operating conditions, the inadvertent actuation of the suppression systems in the control building HVAC equipment room (Zone 12-B) or the HVAC heat exchanger and chiller area (Zone 4-A) may result in the loss of control building HVAC due to spray. Should evacuation of the control room be necessary due to loss of HVAC, capability to shutdown the reactor will be provided by alternate shutdown. Automatic fire suppression system sprinkler discharge on safety-related equipment and cables located in Fire Zone 3-B does not pose a hazard that will affect safe operation or equipment function. Safety-related cabling is environmentally qualified for 100% relative humidity/wet environment (water spray). There is no safety-related equipment that is accessible to water damage in Fire Zone 3-B sprinkler coverage.

Any water discharged from the suppression system above the control room in the control building HVAC equipment room would not affect the control room. The duct and pipe shaft passing through the control room is sealed airtight and there is an adequate drainage system in the HVAC equipment room. Should evacuation of control room become necessary due to loss of HVAC, capability to shutdown the reactor will be provided by alternate shutdown.

Section 9.5.1.2.3.1 of the UFSAR evaluates the failure of fire protection system piping and the effects of flooding on safety-related equipment. Any reactor building flooding will drain freely to the lower level. This level is divided into watertight compartments equipped with floor drains which alarm on water buildup. The torus compartments contain safety-related equipment such as the isolation valve operators for the torus suction lines. However, the volume of this compartment is such that approximately 2 hours would be required to flood to the level of the operators at the rated flow of the fire pumps, thus providing ample time for operator action to isolate the flooding condition. Further, a pressure decrease in the fire system will alarm in the control room permitting operator action within a short time.

Sprinkler systems located over essential safety systems were seismically supported to preclude damage during safe shutdown earthquake. Pipe support loads were reviewed in accordance with requirements of IE Bulletin 80-11 and due to their low magnitude were found to have insignificant effects on the walls.

The probability of occurrence and the magnitude of the consequences of an accident or malfunction involving the equipment protected by the installation or modification of the fire suppression systems was not increased above that previously evaluated in the UFSAR. The addition or modification of the fire suppression systems did not in any way alter the function or operation of any safe shutdown system. Therefore, an accident or malfunction of a different type than any previously evaluated in the UFSAR, was not created by the addition or modification of the fire suppression system. The addition or modification of the fire suppression systems did not in any way alter the function or operation of any safe shutdown system. The margin of safety as defined in the bases of any technical specification was not affected by the addition of the fire suppression systems.

DCP No. 1209 Turbine Full Arc Admission Modification/Turbine Control Valves

Description and Basis for Change: G.E. Technical Information Letter No. 954, "Nuclear Units with Partial Arc Admission" recommended converting partial arc admission turbines to full arc admission capability. This was based on three (3) forced outages due to first stage bucket or wheel dovetail failures in turbines with partial arc admission at other nuclear plants. Under partial arc admission, these buckets are subjected to their highest dynamic loading. Under full arc admission, the stimulus to the first stage buckets is significantly reduced.

Duane Arnold turbine control valves were adjusted so all 4 valves open equally. This required changes in the electrohydraulic control system and adjustment of the stop pin of one control valve.

Summary of Safety Evaluation: This change involved nonsafety-related components and did not pose any unreviewed safety questions. This change did impact safety-related systems and therefore was evaluated per 10 CFR 50.59. The change was within the scope and intent of the UFSAR, Section 15, and was in compliance with the technical specifications as revised.

The Turbine Control Valve (TCV) modification impacted the load rejection transient discussed in Chapter 15 of the updated FSAR due to the reduction in valve closing time. However, the probability of the load rejection transient was not increased as this modification did not introduce a common mode failure which would initiate a load rejection event. The consequences of a load rejection transient were not increased beyond the limits in the updated FSAR (UFSAR) as the purpose of the change to the technical specifications was to add new Minimum Critical Power Ratio (MCPR) operating limits for Cycle 7, based upon the GE analysis with the modification, and to assure that the UFSAR limits were maintained. Lastly, no equipment identified as safety-related was affected by this change as the TCVs do not perform a safety function. It is noted that no changes were made to the TCV position switches which provide an input to RPS and are safety-related. The probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously analyzed in the FSAR was not increased.

The consequences of a failure of the TCVs is analyzed in the UFSAR (Section 15.0) and the modification did not invalidate the previous assessment by introducing a new event. The possibility of an accident or malfunction of different type than any evaluated previously in the FSAR was not created.

The operating limits which were updated in the technical specification change were to maintain the present margin of safety by restricting the operating domain based upon GE analysis. The margin of safety as defined in the basis of any technical specification was not reduced.

DCP No. 1212 Termination for Backup Scram Valves

Description and Basis for Change. This change provided for the installation of terminal strips and boxes for Backup Scram Valves, SV-1840A and B. The installation of terminal strips and boxes facilitates future maintenance of the valves. The terminal strips eliminate the use of splices which necessitated the shortening of both the solenoid pigtail and control cable each time that the solenoid was disconnected.

Summary of 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 were seismically supported.

As a result, this change did 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 was introduced. The margin of safety, as defined in technical specifications, was not reduced. No changes to the plant technical specifications were required and no unreviewed safety questions were created by these changes. This change was safety-related.

DCP No. 1213 Cable Replacement for Main Steam Line Drain Control Valve

Description and Basis for Change: Damaged portions of cables which terminate at main steam line drain control valve MO-4424 were replaced. A terminal box was added to facilitate any future replacements.

Summary of Safety Evaluation: This change was safety-related. The terminal blocks are fully qualified to IEEE 344-1975 and the blocks and cable are qualified to IEEE 344-1975. Also, the terminal boxes in which the terminal blocks were installed were seismically supported.

As a result, this change did 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 was introduced. The margin of safety, as defined in the technical specifications, was not reduced.

DCP No. 1214 Cable Replacement for Inboard HPCI Steam Supply Valve

Description and Basis for Change: Portions of damaged cables which terminate at inboard HPCI steam supply valve, MO-2238, were replaced. A terminal box was added to facilitate any future replacements.

Summary of Safety Evaluation: This change was safety-related. The terminal blocks are fully qualified to IEEE 344-1975 and the blocks and cable are qualified to IEEE 323-1974. Also, the terminal boxes in which the terminal blocks were installed were seismically supported.

The three questions of 10 CFR 50.59 are addressed as follows:

1) The probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR was not increased since the change involved replacing existing equipment with similar qualified equipment and also the addition of new equipment which is properly qualified.

2) The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR was not created since fully qualified equipment was used for this change.

3) The margin of safety, as defined in the basis of any technical specification, was not reduced, again, since fully qualified equipment was used for this change.

DCP No. 1217 Addition of Pressure Gauge to Indicate Electric Fire Pump Starting Pressure

Description and Basis for Change: A 0-250 psig pressure gauge with isolation valve was installed on the same pressure sensing line as PS3301, to indicate starting pressure of the Electric Fire Pump (1P-48) as required by test procedure. This change was in response to an Iowa Electric QA Audit.

Summary of Safety Evaluation: This modification was not safety-related. The additional pressure gauge did not present additional hazards not evaluated in the Final Safety Analysis Report (FSAR), nor did the addition of the gauge reduce the margin of safety defined in the basis of any technical specification.

DCP No. 1223 Replacement of "D" Main Steam Line Flow Transmitter

Description and Basis for Change: The existing main steam line flow transmitter FT-4411, a Barton model 368, failed and required replacement. An equivalent spare, Rosemount model 1151, obtained from and certified by GE in 1978 was installed in the existing transmitter's place. Instrument piping was also modified.

Summary of Safety Evaluation: This change was safety-related. General Electric provided a Product Quality Certification (PQC) which verified that the new transmitter is equal to or better than the previous transmitter. The new transmitter is designed by GE as "Essential-Passive". This is consistent with the requirements for FT-4411. The instrument is part of the reactor coolant pressure boundary, seismic Class 1, and must maintain passive integrity.

As indicated in UFSAR Section 3.10.1.1, all General Electric instrumentation devices are individually qualified by vibration test for 1.5g (or more) horizontal and 0.5g (or more) vertical. This statement indicates that the original transmitter was qualified to at least these levels. By virtue of the PQC, the new transmitter is qualified to withstand the required ZPA input DBE response of 1.4g horizontal and 0.46g vertical.

The three questions of 10 CFR 50.59 are addressed as follows:

1) The probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR was not increased since the change involved replacing existing equipment with similar qualified equipment.

2) The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR was not created since fully qualified equipment was used for this change.

3) The margin of safety, as defined in the basis of any technical specification was not reduced, again, since fully qualified equipment was used for this change.

DCP No. 1235 VAX Computer Conduit Routing

Description and Basis for Change: Connection of the Health Physics record system to the main VAX computer required two four-inch unscheduled conduits to be routed from the computer room (control building) to access control in the administration building. For installations in a Category 1 building (such as the control building) which pass over safety-related equipment, the integrity of the existing structure must be ensured. Accordingly, installation of conduit and penetrations in the control building were documented herein.

Summary of Safety Evaluation: This change involved a modification to a Category I structure by a nonsafety-related installation. This is analogous to a "Seismic II over I" situation and, in accordance with previous practice, the work was designated as nonsafety-related. This safety evaluation addressed the effect of the authorized work on the integrity of the existing structures.

This work did not affect the safety or operability of any safety-related equipment. The penetrations were firesealed in accordance with established procedure. The penetrations were resealed full depth with grout and did not affect existing radiation zones. There was no ALARA impact. New conduit in the control building was supported such that it will retain its integrity during a seismic event, thus affecting no safety-related equipment nearby. A FHA analysis of the additional combustible material (cable) added to the computer room indicated that the resulting change was negligible. The probability of occurrence or magnitude of the consequences of an FSAR-evaluated accident was not increased nor was the possibility of a new type of accident created.

Neither the boxout and penetration modifications nor core drill reduced the existing structural factors of safety below the limitations of UFSAR Section 3.8. The boxout and penetrations by definition contribute no strength to the structural integrity of the slab, and were included as "openings" in the slab design. Therefore, any modification to these openings, provided no damage, is done to the surrounding slab, has no affect on the margin of safety of the structure. The core drill was prepared in accordance with two Bechtel-prepared references: BLIEG-78-116 and the "Guidelines for the Cutting of Penetrations in Existing Masonry Blockwalls, Poured Concrete Slabs and Walls, and Structural Steel". Core drill size, location and reinforcement cutting limitations were established per these references. Core drills completed in accordance with these documents did not unacceptably affect the margins of safety of existing concrete structures.

All conduit supports were also designed to have an acceptable safety margin per UFSAR Section 3.8 and applicable codes. The small loads imposed on the existing structure by these supports were well within the capacity designed into the structures. Thus, the required margins of safety for the control building, set by UFSAR Section 3.8, are not changed.

DCP No. 1239 Replacement of Feedwater Flow Transmitters

Description and Basis for Change: A survey of the Feedwater (FW) flow transmitter instrument calibration data indicated that the instruments were exhibiting a bad zero drift problem. The error had been as high as 4.75% and was persistently well above the permissible range of 0.5%. Core Thermal Power Subroutine (SR-1A) uses the FW flow transmitter output to calculate the reactor power. Therefore, accuracy of these instruments directly affects the core thermal power and could have unnecessarily limited the plant capacity factor.

General Electric SIL 240 highlighted the pressure transmitter drifting as a generic problem with all BWRs where Barton model strain gage transmitters are installed. The old FW flow transmitters were strain gage type Barton model 368.

Barton 368 transmitters were replaced with Rosemount model 1151 DP transmitters which operate on a different principle and are less susceptible to ambient temperature variations. In SIL 240, GE recommended replacement of Barton transmitters with GE 163C1559 transmitters which are essentially Rosemount model 1151 DP transmitters. GE has qualified these transmitters for Class 1E applications.

Summary of Safety Evaluation: The change was nonsafety-related. The probability of occurrence or the magnitude of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR was not increased since no changes were made in system function. The possibility of an accident or malfunction of a type different than any previously evaluated in the UFSAR was not created. As the new instruments are more reliable, this change enhances plant operation. The margin of safety, as defined in the basis of any technical specification, was not reduced. The new instruments, with a proven performance record, have increased plant safety.

DCP No. 1246 HPCI Discharge Piping Support Modification

Description and Basis for Change: A discrepancy was found regarding an Anchor Darling Corporation (ADC) motor operator weight. Anchor Darling used a figure of 365 pounds in their Seismic Analysis Report Number 1359-1. The ADC motor was weighed at the Duane Arnold Energy Center and was determined to be 435 pounds. Based on this load, a design check was performed and all components were found to be satisfactory for the increase in load with the exception of support number EBB-5-SR-8, which had a factor of safety of less than four (4) for the concrete expansion anchors. The bolts were replaced with larger bolts, thereby increasing the bolt safety factor to the required level.

Summary of Safety Evaluation: This work involved modification of a Category I structure by a safety-related installation. This work positively affected the safety and operability of safety-related piping. The previous support had a design factor of safety of approximately 3.4 which was less than the required design factor of safety for the concrete expansion anchors. As per NRC IE Bulletin Number 79-02, the minimum design factor of safety shall be 4 for wedge and sleeve-type anchor bolts. All other components were found to be acceptable. The modified support safely carries the design loads in the manner originally intended in the piping analysis. Therefore, the probability of occurrence or magnitude of the consequences of an FSAR-evaluated accident was not increased nor was the possiblity of a new type of accident created.

This support was designed to have an acceptable safety margin per UFSAR Section 3.8, IE Bulletin 79-02, and applicable codes. The load imposed on the existing structure by this support is well within the capacity designed into the structure. Thus, the required margins of safety for the reactor building, set by UFSAR, Section 3.8, remain intact.

DCP No. 1247 Snubber Clevis Changes

Description and Basis for Change: Pacific Scientific's Service Bulletin SR83-01 identified a possible defect that could cause a model PSA-1 snubber to go inoperative. As a result, it was decided to change this snubber out in October, 1983 while the plant was shut down. Because of the difficult working conditions under which the snubber was to be reattached to the drywell and reactor head spray piping, new clevis bushings were installed to expedite the changeout. These bushings were machined with lips to help retain the bushings onto the clevis during installation.

Summary of Safety Evaluation: A 10 CFR 50.59 safety review was performed for this work. This was a safety-related change that had no detrimental effect on the safety of the plant. This change did not increase the magnitude of the consequences of a snubber failure and did not pose an unreviewed safety question. No changes to the plant technical specifications were necessary.

DCP No. 1256 Replacement of Lantern Ring in 1P-117C

Description and Basis for Change: During routine maintenance work performed on river water supply pump 1P-117C, the lantern ring was misplaced. This lantern ring had served no purpose, since seal water was never supplied to it. It was decided to replace the nonfunctional lantern ring with an additional ring of packing. This DCP provided a mechanism for formal design review of the replacement of the lantern ring with standard packing ring.

Summary of Safety Evaluation: This change was safety-related. After reviewing pump construction and the intended function of the packing and lantern ring, it was concluded that the substitution of a standard packing ring for the lantern ring would not adversely affect the pump's ability to satisfy the technical specification requirements for head and flow. The only conceivable way in which packing might be seen as adversely affecting pump performance would be if the packing was sufficiently overtightened. However, this is strictly a function of the torque value on the gland follower bolts and not how many layers of packing are involved.

A factor which supports this substitution is the fact that the lantern ring was never used. The intended function of the lantern ring was for sealing fluid flow along the axial direction of the shaft and required a supply of sealing fluid. The sealing fluid connections were never made, thereby making the lantern ring useless. In all probability, the additional layer of packing enhances the ability of the packing box to control leakage along the shaft. The pump manufacturer was contacted and stated that several of their customers chose not to use the lantern ring. He foresaw no problem with our substitution.

Therefore, based on the above considerations, it was concluded that this change did not adversely affect the performance of pump 1P-117C and that an unreviewed safety question did not exist based on the following:

1. This change did not create the possibility for an accident or malfunction of a different type than evaluated previously in the updated FSAR.

2. This change did not increase the probability of occurrence of an accident or malfunction of equipment previously analyzed in the updated FSAR.

3. This change did not increase the consequences of an accident or malfunction of equipment previously analyzed in the updated FSAR.

4. Because of this change, there was no reduction in the margin of safety as defined in the basis for technical specification 3.5.J.

DCP No. 1283 Standby Filter Units Sampling Modification

Description and Basis for Change: This change allowed for a method to sample the activated charcoal cells in the control building standby filter units. A sampling method was needed because Regulatory Guide 1.52 states that representative samples of the charcoal must be periodically analyzed for efficiency. Until recently, canisters which contain representative samples were available for removal from each filter unit and subsequent analysis. The last canister was used in 1983. Means for sampling were required before the next required analysis in May, 1984.

A hole was drilled in the uppermost charcoal bed of each unit as a means to obtain these samples. Some electrical connections were determinated to access the filter unit. Enough activated charcoal was removed from the bed for several samples. One was for analysis in May, 1984 and the others were put into sampling canisters and reinstalled in the filter unit for purposes of future retrieval and analysis. After adding replacement charcoal to the filter, the hole was sealed using a gasket and a metal plate.

Summary of Safety Evaluation: This change was safety-related. The modification did not increase the probability of accidents analyzed in the UFSAR or accidents not previously reviewed. The integrity of the carbon beds was not compromised because the holes were repaired with a plate of sufficient strength and a tight fitting gasket to assure that the air path through the filter was not changed.

This change is important from a safety standpoint in that the function of the SFU is essential in assuring acceptable breathing air to the control room at all times. The SFUs must be tested and proved operable for the plant to be allowed to operate. The addition of new canisters enables the SFUs to operate normally, with the original margin of safety, as defined in the basis of the technical specifications. Therefore, this change did not cause an unreviewed safety question because:

1) The probability of occurrence or the consequences of an accident or malfunction of safety-related equipment previously evaluated in the UFSAR was not increased.

2) The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR was not created.

3) The margin of safety, as defined in the basis of technical specification, was not reduced.

SECTION B - PROCEDURE CHANGES

During 1984, various procedures as described in the safety analysis report were revised and updated. All changes were reviewed against 10 CFR 50.59 by the DAEC Operations Committee. No procedure changes were performed that constituted unreviewed safety questions.

The emergency operating procedures (EOP) were placed in effect in 1984. These site-specific procedures were derived from the BWR Owner's Group EPGs which had received prior NRC review.

All special test procedures performed in 1984 were also reviewed by DAEC Operations Committee. No unreviewed safety questions were found. Summaries of these special tests are listed below.

Test No.	Title/Description
SpTP No. 107	Low-Low Set (LLS) Valve PSV 4401 Open/Close Relay Testing This test was performed to demonstrate that connecting an ohmmeter across pressure switch PS-4544 on the X100 scale would energize solenoid K21A of the channel "A" Low-Low Set Open/Close Relay Logic. The test repeated the actions (with the reactor shut down) which were believed to have caused a reactor scram on the previous day (Reference LER 84-001). This special test was performed January 8, 1984.
SpTP No. 108	"A" LPCI Loop Selection Logic Test A partial test of the "A" LPCI loop selection logic was performed to determine the cause of a blown fuse which caused the "RHR Bus A/B Logic Power Failure" alarm and which may have tripped the reactor recirculation pumps. This special test was performed January 25, 1984.
SpTP No. 109	Procedure for Testing MSIV-LCS Heater 1S-122B This test was performed to determine resistances of the MSIV-LCS heater and to detect possible short circuits on open connections in the heater as a function of temperature. This special test was performed May 10, 1984.
SpTP No. 110	Partial Logic Test for the 1E-5A/5B Feedwater Heaters Drain Line Crossconnect Valve CV-1341 A partial operational test of the control circuitry for the 5A/5B feedwater heater drain line crossconnect valve, CV-1341 was performed. This special test was performed October 10, 1984.

Test No.	Title/Description
SpTP No. 111	Temperature Switch Response in HPCI Steam Leak Detection System on Loss and Regain of AC Power This test was performed to determine the responses of temperature switches in the HPCI steam leak detection system to a loss and subsequent reapplication of AC power over the course of approximately 10 seconds. This special test was performed October 15 and 21, 1984.
SpTP No. 112	This special test was not performed in 1984.
SpTP No. 113	Emergency Light Level Verification This test was performed to verify illumination levels provided by 8-hour emergency light fixtures along access pathways to alternate shutdown locations. This special test was performed while the reactor was shut down on October 28, 1984.
SpTP No. 114	Measurement of Power Range Neutron Monitor System RBM Null-Initiate Signals This test collected data pertaining to the Power Range Neutron Monitor (PRNM) system RBM null-initiate signal. The data was needed for the ARTS Improvement Program. This special test was performed on November 12, 1984.
SpTP No. 115	CV 1361 Internals Test This test was performed to functionally test a possible modification to correct the inoperability of the primary steam valves to the ejector. This special test was performed on November 10, 1984.

SECTION C - EXPERIMENTS

This section has been prepared in accordance with the requirements of 10 CFR 50.59(b). There were no experiments conducted during the calendar year 1984.

Section D Safety and Relief Valve Failures and Challenges

This section contains information concerning relief valve and safety valve failures and challenges for calendar year 1984 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). Note that any instance in which a main steam relief or safety valve was manually cycled open, for surveillance testing or other reasons, is included for your information. No valves were cycled open for surveillance testing in 1984. There were no (0) safety valve failures or challenges during 1984. There were no (0) relief valve failures during 1984. There was one (1) relief valve challenge during 1984. This event is described below.

Date	Event Description
01/07/84	Relief Valve PSV-4401 lifted due to an error in the performance of a low-low set instrument functional test. This event contributed to a reactor scram as reported in LER 84-001. The valve was closed after 75 seconds.

DHB

PRIORITY ROUTING

First		Second	
RA		RC	
ORA		FIC	
DRP		SGA	
RS		RL	
DRSS		OL	
DRMA		OI	
		PAO	

FILE

right

Iowa Electric Light and Power Company

March 1, 1985
DAEC-85-0183

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
1984 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, 1984 thru December 31, 1984.

Very truly yours,

Daniel L. Mineck

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

DLM/WRK/ko*

Enclosure

cc: Director of Inspection and Enforcement
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Washington, D. C. 20555

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