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AUTH.NAME AUTHOR AFFILIATION
FRANZ,J.F. Iowa Electric Light & Power Co.
RECIP.NAME RECIPIENT AFFILIATION
MURLEY,T.E. Office of Nuclear Reactor Regulation, Director (Post 870411

SUBJECT: Forwards 1991 repts on Plant Design Changes,Procedures Changes,Experiments & Fire Plan Changes.

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Iowa Electric Light and Power Company

February 28, 1992
NG-92-0482

JOHN F. FRANZ, JR.
VICE PRESIDENT, NUCLEAR

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
1991 Annual Report of Facility Changes, Tests, Experiments, and Safety and Relief Valve Failures and Challenges

File: A-118e

Dear Dr. Murley:

In accordance with the requirements of Appendix A to Operating License DPR-49, 10 CFR Section 50.59(b), and NUREG-0737 (Item II.K.3.3), please find enclosed the subject report covering the calendar year 1991. In addition, a summary of changes to the DAEC Fire Plan implemented during 1991 is included in our report.

Please contact this office if you have any questions regarding this matter.

Very truly yours,

John F. Franz, Jr.

John F. Franz, Jr.

JFF/CJR/pjv

cc: C. Rushworth
L. Liu
L. Root
R. McGaughy
C. Shiraki (NRC-NRR)
A. Bert Davis (Region III)
NRC Resident Office
Commitment Control Nos. 910048, 910049

JE47

SECTION A - PLANT DESIGN CHANGES

This section contains brief descriptions of and reasons for plant design changes completed during the calendar year 1991 and summaries of the safety evaluations for those changes, pursuant to the requirements of 10 CFR Section 50.59(b).

The basis for inclusion of a Design Change Package (DCP) in this report is site closure of the package at the Duane Arnold Energy Center (DAEC) in the calendar year 1991. It is noted that portions of some DCPs listed were partially closed in previous years.

DCP 1458 HPCI Transient Data Monitoring System

Description and Basis for Change

The High Pressure Coolant Injection (HPCI) System has historically not performed as well at the DAEC as compared to the industry. In an effort to analyze and improve the HPCI system performance, a transient data monitoring system was installed. The HPCI Transient Data Monitoring System monitors the following parameters:

1. Turbine Speed
2. Pump Discharge Flow
3. Pump Discharge Pressure
4. Stop Valve Balance Chamber Pressure
5. System Oil Pressure
6. Stop Valve Continuous Position
7. Governor Valve Continuous Position
8. EG-M Output
9. Ramp Generator Output
10. Flow Controller Output
11. Steam Line High Differential Pressure

The eleven parameters are recorded on a single 12-channel recorder housed in the HPCI Transient Data Monitoring System Panel (1C471) which is located in the southeast corner of the Reactor Building. The recorder is normally on standby and starts recording automatically when the HPCI system receives an initiation signal.

The monitoring signals for the pump discharge flow and pressure, EG-M output, ramp generator output, and flow controller output come from existing safety-related instrument loops. A voltmeter was installed on the HPCI system which also monitors the EG-M output signal. The voltmeter supplies Operations personnel with information concerning this output signal. Should an erroneous voltage signal appear, it would be noted prior to the next surveillance, and appropriate actions could be taken.

The monitoring signal for the turbine speed comes from the existing turbine speed indication loop. Two pressure transmitters were added to monitor the stop valve balance chamber pressure and the system lube oil pressure. A differential pressure transmitter was added to monitor steam line high differential pressure. In addition, Linear Variable Differential Transformers (LVDTs) were mounted on the stop valve and the governor valve to continuously monitor the position of each valve.

Summary of Safety Evaluation

The HPCI Transient Data Monitoring System only affects the HPCI system. The HPCI system is provided to mitigate possible accidents and is not an accident initiator. This modification did not tie into or impact any other system that could increase the probability of occurrence or consequences of an accident important to safety. Failure of the HPCI system has previously been analyzed in the UFSAR and is not a limiting event.

Five Quality Level I analog signal isolators were added to existing HPCI safety-related instrument loops. This effectively inserted a resistor in each of the existing loops. These resistors can cause possible failure of the HPCI system if they were to short. However, since the probability of a short circuit due to the addition of the new isolators is low, the overall system reliability has not been reduced.

The voltmeter added to the system is fused to effectively isolate the voltmeter from the EG-M output signal. The EG-M output signal is short-circuit protected and would not be damaged by a short in the voltmeter before the fuse could clear the fault. The current output of the EG-M signal is sufficiently large to blow the fuse should a short circuit condition exist. Therefore, the addition of the voltmeter will not adversely impact HPCI reliability.

The existing portion of the speed signal that is used by the monitoring system is for indication only and is not required for the operation of the HPCI system. The tachometer board located in the EG-M is isolated from the other components by two transformers.

The pressure transmitter, tubing, piping, and valves added to monitor the stop valve balance chamber pressure increased the volume of the balance chamber by approximately 6.5%. The balance chamber at the top of the stop valve was designed to provide a steam cushion for the valve disc when opening, and a controlled steam flow path to pressurize the downstream side of the valve. The addition of instrumentation did not change the function of this balance chamber. The additional volume changed the time response of the chamber during the pilot valve opening. However, the beginning pressure (1,000 psig) and ending pressure (100 psig) were not changed because the steam inlet and outlets were not altered. The increase in volume had an insignificant impact on the 1.5 second depressurization transient of the balance chamber from 1,000 psig to 100 psig and did not affect the overall performance of the stop valve.

The three pressure transmitters added to monitor the stop valve balance chamber pressure, the system lube oil pressure, and the steam line high differential pressure were seismically mounted. The pressure transmitters do not perform any safety function. However, the pressure retaining components of the transmitters were proof-tested to ensure the HPCI system pressure boundaries will remain intact during any expected condition. Because of the low probability of a failure to the new transmitters that would adversely affect HPCI, the overall reliability of the system has not been reduced.

The two LVDTs that were installed do not perform a safety function. The amount of weight they added to the system is insignificant. The overall size and strength of the monitored devices is such that no mounting failures of the LVDTs could cause binding.

This modification did not affect the operation of any other safety-related system nor did it change the operation of the HPCI system for any evaluated accident.

As described in the Technical Specification (TS) bases, the HPCI system is provided to ensure the reactor core is adequately cooled in the event of a small break in the nuclear system which results in a loss-of-coolant without rapid depressurization of the reactor vessel. The HPCI system is required to operate until the reactor vessel pressure is below the pressure at which Low Pressure Coolant Injection (LPCI) or Core Spray system operation maintains core cooling. The addition of the HPCI Transient Data Monitoring System did not affect the operation of the HPCI system or reduce the margin of safety as defined in the Technical Specification bases.

This modification did not involve an unreviewed safety question or a change to Technical Specifications.

MM-279 <u>Removal of XL3 Alarm Contacts from Fire Indicating Unit #1</u>

<u>Description and Basis for Change</u>

The XL3 fire panel provides the Control Room with information on system status for the fire protection systems in the Control Room, Control Building HVAC, the Air Compressor Building, the Data Acquisition Center, the Security Building, the Main Generator Exciter, and the Low Level Radwaste Building.

The XL3 was equipped with a set of normally open contacts that would close when any alarm causing device goes into alarm. This set of contacts was tied in parallel with the Zone Indicating Units associated with Fire Indicating Unit (FIU) #1. This set of contacts was also paralleled with a surge suppressor.

This arrangement did not allow proper operation of trouble alarms at the FIU due to the leakage current through the surge suppressor. This modification removed the XL3 panel alarm contacts from the FIU alarms to allow proper operation of the FIU alarms. The XL3 is still equipped with an audible alarm and LED display to alert the control room operators of system alarms.

This modification also updated electrical drawings to show the addition of a printer to the XL3 fire protection system. An existing receptacle was also relocated to the west wall of the Control Room backpanel area to eliminate the need for an extension cord to power the XL3 printer.

<u>Summary of Safety Evaluation</u>

The XL3 is a self-contained continuous detection, fire protection system that provides the control room with information on system status for various fire protection systems.

This modification did not change the operation of the XL3 system or FIU # 1. It removed the XL3 system alarm contacts from the alarm circuit associated with FIU #1 and added a receptacle for the XL3 printer. This allows proper operation of the alarms associated with FIU #1 and eliminates the need to use an extension cord to power the XL3 printer.

This modification did not increase the probability of occurrence or the consequences of an accident or the malfunction of equipment important to safety previously addressed in the FSAR; it did not create the possibility of an accident or malfunction of a different type than previously evaluated in the FSAR.

The margin the of safety as defined for Fire Protection Systems in Technical Specifications was not changed or decreased by this modification. The system will continue to provide fire detection and system trouble alarms.

The modification did not involve an unreviewed safety question or a change to Technical Specifications.

PMP 0015 Welding and Hydrolazer Receptacles

Description and Basis for Change

The welding units located inside the East entrance to the Machine Shop had temporary 480VAC welding power. This modification installed two permanent 480VAC welding receptacles in the vicinity of these welding units. These receptacles are powered from panel 1L09. The circuit breaker (CB) is the same CB which furnished the temporary power to the welders. The Hydrolazer unit located in the railroad airlock also had temporary 480VAC power provided for the unit. A permanent 480VAC power receptacle was installed in the railroad airlock. This permanent power is also fed from a CB in panel 1L09.

Summary of Safety Evaluation

None of the accidents previously evaluated in the FSAR were affected by the modifications to the welding receptacles or the Hydrolazer receptacle. The receptacles are fed from a lighting panel (1L09) which does not feed any equipment important to safety. A fault in the welding receptacles and the Hydrolazer receptacle will only affect 1L09 circuitry. There would not be any impact on equipment important to safety which could cause an increase to the radiological consequences of any previously evaluated accident.

The margin of safety has not been reduced because these modifications did not affect any safety-related equipment or any equipment important to safety. A margin of safety for the welding receptacles and the hydrolazer receptacle is not defined in the Technical Specifications.

This modification did not involve an unreviewed safety question or a change to Technical Specifications.

NCR 91-037 Description

Non-Conformance Report (NCR) 91-037 was initiated as a result of a review of Bill of Materials (BOM) files. During the review, it was discovered that the air pressure being supplied to the solenoid valves (SV5704A and B, SV5718A and B) which control the drywell well water isolation control valves was set at 90 psi, while the manufacturer recommends a maximum air pressure of 75 psi. Correction of the problem requires changing out the solenoids to a different model. Currently DAEC has no qualified replacements on site.

A review of the history of the currently-installed solenoids showed they have been installed since 1982, and passed numerous quarterly tests. Due to proper operation of the solenoid valves for over nine years, the reliability of the instrument AC power supply, and that the air supply is safety-related, Design Engineering recommended that the valves continue to be operated until replacements can be installed.

Summary of Safety Evaluation

Well Water is used to provide drywell cooling, which supports normal power operation by maintaining drywell temperature and pressure within the acceptable limits. The Well Water system is not needed or relied upon during any accident or malfunction as described in Chapter 15 of the FSAR, nor is it required by the Nuclear Safety Operational Analysis (NSOA).

The design function of the drywell well water isolation valves is to close on a Group 7 signal. They can also be manually isolated in the event the well water piping were to break with no isolation signal present. The solenoid valves in question serve to provide air to the isolation valve operators during containment isolation conditions. Air pressure is supplied via the safety-related Heating & Ventilation instrument air compressors.

Per the manufacturer the solenoid valves operating pressure rating of 75 psi was determined by testing in which valves failed at 83 psi when the coil was energized with 102 VAC. Higher voltages will raise the pressure at which the valves fail. At the DAEC, the solenoids are energized with 120 VAC instrument AC and have operated satisfactorily with 90 psi air supply for over 9 years. In order to fully evaluate the condition the solenoid valves are in, the potential for a reduced voltage power supply must be evaluated. 120 VAC instrument AC is normally lined up so that it is powered from the 125 VDC battery system through the inverters. The instrument AC voltage will not degrade during accident conditions due to the design of the inverters. These inverters are designed to swap to an alternate AC supply if the output voltage drops to 108 VAC. However, the inverters are also designed to give a continuous 120 VAC output from a battery voltage range of 140 to 105 VDC. At 105 VDC input the inverters will isolate the output until a power supply is available. Due to the design of the instrument AC power supply, the voltage is not expected to drop below 120 VAC even under accident conditions.

Allowing these solenoid valves to continue to be used until replacements can be installed is justified since there is significant operating history with the solenoids being operated at 90 psi and the fact that no failures have occurred. This operating history provides assurance that these valves will perform the intended design function, if needed. Allowing these solenoids to operate at the higher pressure does not alter or modify the consequences of the failure of these valves. The failure of these solenoids is no different than a failure of instrument air, since both will cause the valves to fail open. If the isolation valves failed open, the containment boundary would be maintained by the well water piping. However, the air supply is safety related and is designed to maintain pressure under events described in Chapter 15 of the FSAR. Because we are assured of a continuous air supply and that the power supply will not degrade below the normal operating voltage during accident conditions, allowing them to operate at 90 psi does not increase the consequences of a malfunction of equipment important to safety evaluated previously in the FSAR.

This condition did not involve an unreviewed safety question.

NCR 91-071 Well Water (Domestic Water Supply)

Modification of the Well Water System (Domestic Supply) was performed under a Corrective Maintenance Action Request (CMAR). New valves and filters were added to the system. The modification affected P&ID M144 and several Field Sketches along with the Operation Instruction for

Well Water. To adequately control changes, this modification should have been processed under design control per our QA program as a Plant Modification (PMP). Consequently, the NCR was written and a safety evaluation was prepared.

Summary of Safety Evaluation

This modification added new valves and filters to the Well Water system. The Well Water system's operation has no effect on the operation of any safety system in the plant. The relocation of valves and the addition of new valves and filters had no impact on the operation of the system.

No accidents evaluated in the SAR involve the Domestic Water System. There is no equipment important to safety in the Domestic Water System. The Domestic Water System has no effect on equipment evaluated in the SAR, nor on systems evaluated in the SAR which could cause an accident.

This modification did not involve an unreviewed safety question or a change to Technical Specifications.

Core Spray Minimum Flow Valve

Description

This is the evaluation of the effect of removing the automatic function of the Core Spray minimum flow valve, MO2124. The automatic function of the valve is being removed because Flow Indicating Switch (FIS)2131 is inoperable. FIS2131 has contacts in the Core Spray minimum flow valve logic which open MO2124 when system flow is less than 300 gpm and closes MO2124 when system flow is greater than 600 gpm. The valve will be able to be operated from the control room using the valve's handswitch.

The minimum flow valve will be maintained in the open position during all plant operations except when isolation of that line is required or increased Core Spray flow to the vessel is required. Actions will be taken to ensure that this valve remains open including the installation of a Warning tag which will state: "The Auto Function is disabled. Leave OPEN unless closure is needed for Primary Containment Isolation or to increase Core Spray flow to the vessel."

Summary of Safety Evaluation

The Core Spray system does not initiate any accidents previously evaluated in the FSAR, nor does this modification affect the function of the Core Spray system such that it could initiate an accident. As shown in the NSOA, the function of the Core Spray system is to mitigate the consequences of the following accidents and transients:

a. Loss of Offsite Power
b. Control Rod Drop Accident
c. Loss of Coolant Accident (LOCA)-Inside Containment
d. LOCA-Outside Containment
e. Loss of Control Room Habitability

The ability of the Core Spray system to mitigate the above accidents and transients and perform its intended function was verified during post-modification testing. The testing ensured that the Core Spray system will deliver the required flow rate and pressure as defined in TS 4.5.A with the minimum flow valve open. Satisfactory demonstration of the Core Spray system's ability to mitigate the accidents and transients described above ensures that removing the automatic function of the minimum flow valve will not increase the consequences of a previously evaluated accident.

The containment isolation function of the minimum flow valve will also be ensured. Removing the automatic functioning of the valves allows the valve to be remotely isolated from the control room in the event the minimum flow line requires isolation.

As described in UFSAR section 6.3.2.3, the purpose of the minimum flow line is to prevent the core spray pump from overheating when pumping against a closed discharge valve. While it is desirable to maintain the minimum flow valve closed during system operation when flow is greater than 600 gpm, an orifice in the minimum flow line restricts flow in that line and helps ensure that the system is still capable of meeting its design flow and pressure requirements. In addition, during extenuating circumstances, the control room operators will be able to close the minimum flow valve if required to increase Core Spray flow to the vessel.

Administrative controls will be implemented to ensure that the minimum flow valve remains open. By maintaining the minimum flow valve open, the function of the minimum flow line is maintained. Additionally, operating the Core Spray system with the minimum flow valve open will not affect any other components in the Core Spray system or any other system important to safety. Operating the Core Spray system with the minimum flow valve open will still provide adequate cooling to the core. The system will still be injecting into the core through its design flow path.

This modification did not involve an unreviewed safety question or a change to Technical Specifications.

During 1991, 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. 2 safety evaluations were written to support changes to plant procedures during 1991. Summaries of these procedure changes and their safety evaluations are provided below. No procedure changes were made that involved unreviewed safety questions or changes to Technical Specifications.

All Special Test Procedures (SpTPs) performed in 1991 were reviewed by the DAEC Operations Committee. No unreviewed safety questions were found to exist. Summaries of these special tests and their safety evaluations are also found below.

TEST/PROCEDURE	TITLE/DESCRIPTION
ODAM	Offsite Dose Assessment Manual The Offsite Dose Assessment Manual (ODAM) is being revised to 1) relocate the radiological effluent specifications from the TS into new ODAM section 6.0 (RTS-231), 2) correct discrepancies identified in revision 3 to the ODAM, 3) incorporate changes to improve the clarity of the ODAM and 4) revise the reporting and detection levels for I-131 in water to achieve consistency with Standard Radiological Effluent TS. The proposed changes are consistent with the recommendations of NRC Generic Letter 89-01 and maintain the level of radioactive effluent controls required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36, and Appendix I to 10 CFR Part 50 and do not adversely affect the accuracy or reliability of effluent dose or setpoint calculations. Summary of Safety Evaluation The proposed change relocates the radiological effluent procedural details from their existing location in the TS to the ODAM. The NRC has stated, in Generic Letter 89-01, that the details are not required by the Commission's regulations to be included in the TS. The proposed change only affects the programmatic controls for radioactive effluents and radiological environmental monitoring and does not affect the function or operation of the equipment. Programmatic controls added to the TS ensure that programs are established, implemented, and maintained to ensure that operating procedures are provided to control radioactive effluents

consistent with the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50. The relocated specifications still address the limiting conditions for operation, their applicability, remedial actions, associated surveillance and reporting requirements for the equipment and instrumentation which comprise radioactive effluent and environmental monitoring programs. The changes to the detection and reporting levels for I-131 in water only correct discrepancies. The corrected values are conservative and are consistent with standard radiological effluent TS.

The change did not affect the purpose or function of any equipment which could cause an accident previously evaluated in the SAR. The consequences of a malfunction of equipment are unchanged. The affected equipment and instrumentation continue to function as designed and as required by the governing regulations. The governing regulations must still be followed, the surveillance requirements must still be performed within the required time limits, and equipment operability must still be maintained.

This procedure change did not involve an unreviewed safety question.

IPOI 8,
Section 3.2.3 <u>Conduct of Evolutions Which Have the Potential for Draining the Vessel</u>

The purpose of this procedure change is only to provide more specific guidance on what constitutes "Operations with the Potential for Draining the Vessel" (OPDRVS) as used in TS 3.5.G.3(b).

<u>Summary of Safety Evaluation</u>

TS 3.5.G.3(b) provides guidance on Emergency Core Cooling System (ECCS) and diesel generator (DG) operability when irradiated fuel is in the vessel and the reactor is in Cold Shutdown or Refueling. A review of the NSOA, Operating States A (refuel) and B (shutdown/startup) indicates that three accidents (Fuel Handling, LOCA - inside containment, and LOCA - outside containment) must be evaluated. The probability of these accidents is not increased because this procedure change does not specifically affect the operation, design or function of any equipment which could cause any of the three accidents. The change does provide specific guidance on preventing actions which have the potential for draining the vessel.

The bases for TS section 3.5.G discuss the general requirements for ECCS and diesel operability during OPDRVS but do not specifically define what activities constitute OPDRVS. This procedure change only provides better and more specific guidance on what actions constitute OPDRVS in order that the required ECCS and diesel equipment remain operable. The guidance provided in IPOI 8, Section 3.2.3, is based on existing regulatory guidance and conservatively requires emergency equipment to be operable when the conditions of the IPOI are not met.

The procedure change does not specifically affect the operation, design or function of any equipment. It only better defines what actions constitute OPDRVS as used in TS. ECCS and diesels must still be operable during OPDRVS. Any work being performed on equipment which has the potential to drain the vessel must still be performed in accordance with approved standards and work control processes. TS requirements governing operability of equipment which mitigate the consequences of draining the vessel are unaffected.

This procedure change did not involve an unreviewed safety question or a change to Technical Specifications.

SpTP 169 <u>Cable Spreading Room Leakage Test Using A Door Fan</u>

The purpose of this special test was to determine the leaktightness of the Cable Spreading Room. A door-fan was installed at door 407 to introduce administration building air into the Cable Spreading Room. The airflow through the door-fan was measured along with the room pressure developed. This was repeated for different airflow rates so that a leakage vs pressure curve could be developed.

This Special Test provided data for an Engineering Evaluation of the Cardox System for the Cable Spreading Room. This data will be used to help predict the maximum pressure likely to be developed during a Cardox actuation based on our current configuration and a proposed configuration. The proposed modification will provide a controlled vent path to limit the pressure buildup, which will reduce the infiltration of carbon dioxide into the Control Room.

During the test the Cable Spreading Room ventilation was turned off and the ventilation dampers closed. The Cardox system was isolated and a continuous fire watch was established. One of the doors to the cable spreading room was then blocked open and a door fan installed. The door fan pressurized the room to varying pressures and the airflow was monitored. The maximum test pressure was two inches water gauge- significantly less than the ducting design pressure of 10 inches water gauge. The ducting was assumed to be the limiting factor in the pressurization.

Summary of Safety Evaluation

This special leak test of the Cable Spreading Room did not impact any plant systems that could cause any accident previously evaluated in the FSAR. Per the NSOA, the Control Building HVAC is required to mitigate the major accidents. Activities in the Cable Spreading Room cannot cause a rod drop accident, a fuel handling accident, or a LOCA.

This test did not subject the Cable Spreading Room to pressures near its design limits and did not result in the loss of fire protection for that area. This test had no adverse impact on equipment important to safety.

The Cardox system was inoperable during the test but was replaced with a continuous fire watch to ensure that a fire would not impact the ability of plant safety systems to mitigate the accident. The Technical Specifications require that a continuous fire watch be substituted for the automatic Cardox system if the Cardox system is removed from service.

In the event of an accident with a radiological release sufficient to actuate the Control Building radiation monitors, the Control Building HVAC would isolate automatically. The Cable Spreading Room is not intentionally part of the control room habitable zone. It does normally provide some resistance to unfiltered inleakage into the control room. In the event of a Control Building isolation, the procedure contained direction to abort the test and close the doors to the Cable Spreading Room. The isolation setpoint for the Control Building HVAC is 5 mr/hr. The conditions inside the Administration Building would have to degrade to that level to impact the Operator's thirty day dose.

The performance of this procedure did not involve an unreviewed safety question or change to Technical Specifications.

SpTP 170 Feedwater Tracer Test

This test performed an on-line calibration check of feedwater flow elements FE1581 and FE1626 using a radioactive sodium (Na^{24}) tracer. The results of this test may be used to revise the flow coefficients used to determine feedwater flow as it is used in the core thermal power calculations.

Feedwater flow measurement at Duane Arnold is done by a flow venturi in each flow train. The sum of these flows is used in the plant thermal power calculations. Any error in the measurement of feedwater flow will create an almost equal error in the thermal power calculated and therefore affect the amount of electricity the plant is allowed to produce. A discrepancy between the feedwater flows determined by the flow venturis and those determined through the use of the tracer may indicate a drift in the characteristics in the existing nozzles such that the flow coefficients previously determined for them may need to be revised. This test was performed by injecting a radioactive sodium tracer (Na^{24}) in the form of sodium nitrate into a drain line just upstream of the feedwater regulating valves and drawing samples through the Turbine Building Sample system downstream of the high pressure feedwater heaters. Additional samples were taken from the high pressure heater shell side drains to verify that no tube leakage was taking place.

The vendor who performed the test provided the injection and sampling equipment and arranged for the supply of radioactive tracer.

This procedure applied to the installation of the vendor equipment, performance of the radioactive sodium tracer test, and vendor equipment removal. This procedure referred to specific vendor procedures for details of the equipment installation, removal, and test performance.

Summary of Safety Evaluation

The tracer material was 1.0 g $NaNO_3$ irradiated to make approximately 500 mCi of Na^{24}. The injection solution was made up of the tracer material dissolved in 700 g of demineralized water to which a small amount of stable $NaNO_3$ was added. The test involved a 35 minute injection into the 'A' feedwater line followed by a 35 minute injection into the 'B' feedwater line.

During the test, approximately 340 mCi of radioactive sodium in the form of sodium nitrate Na^{24} was injected into the feedwater system. The elevation in reactor coolant Na^{24} is calculated to be approximately 2.37E-3 μCi/g. This is a factor of eight higher than the Na^{24} activity levels normally achieved at Duane Arnold but is comparable to the activity levels typically found in BWRs. Following the test, the activity levels were expected to return to near normal within 12 hours with normal decay and cleanup. Overall sodium levels would temporarily increase from 1 ppb to about 4 ppb and nitrate concentrations would be less than 8 ppb, assuming hydrogen water chemistry is in effect. These concentrations would not result in any noticeable increase in main steam line radiation levels or offgas flow. Vessel conductivity was expected to increase from 0.065 μS/cm to less than 0.081 μS/cm. The maximum pH change would be from 7.0 to 7.27. These changes were considered small and well within accepted operational chemistry guidelines. The demineralized water used in injecting the tracer into the feedwater would likely be air saturated, resulting in a final feedwater oxygen concentration increase of 1.6 ppb. In conclusion, the test would not aggravate general corrosion of IGSCC within the reactor or plant secondary. The effects of such a small quantity of sodium nitrate on the components of reactor vessel, fuel cladding and components of feedwater system would be negligible. The introduction of Na^{24} would cause only negligible increases in the reactor coolant activity. The introduction of Na^{24} and associated valve lineup would not affect the response of any systems required to respond to accidents described in the FSAR. The introduction of Na^{24} into the feedwater and reactor coolant would have no effect on equipment important to safety. The test would result in approximately 500 gallons of high purity water being drained to radwaste. This would represent a small fraction of the amount of waste normally processed daily.

The major ALARA concern relative to the use of Na^{24} was the minimization of personnel exposure during the unloading of the cask, and preparation of the injection solution and counting standards. The vendor had considerable experience handling this material and projected a total whole body exposure for their personnel of approximately 150 mRem. Vendor and IE Health Physics personnel rehearsed several critical operational steps before actual use of the radiotracer. The rehearsal allowed the HP/ALARA personnel to familiarize themselves with the operations associated with tracer transfers and the preparation of the injection solution.

The performance of this special test did not constitute an unreviewed safety question.

SpTP 172 <u>Reactor Water Cleanup System Leak Detection Channel Verification</u>

During the performance of a monthly surveillance test in October, 1991, temperature indications being supplied by various temperature elements in the Reactor Water Cleanup (RWCU) equipment area were being read and evaluated at Panel 1C21 in the Control Room. This evaluation consisted of comparing indications from two temperature elements at the same physical location and verifying that the indicated temperatures are within +/-5 degrees F of each other.

It was found that indication on TI-2745 (point 2) did not agree with the temperature point module indication for TS-2742C. In addition, the indication on TI-2745 (point 3) did not agree with the temperature point module indication for TS-2742B. Efforts to calibrate the point modules to bring them into agreement were not successful. Further evaluation revealed that wiring internal to 1C21 was consistent with appropriate wiring drawings. However, this configuration did not agree with the STP and labeling on Panel 1C21.

The purpose of this Special Test was to resolve this discrepancy and verify the channel configurations within the RWCU portion of the Steam Leak Detection system so that the appropriate documents could be revised. This was accomplished by individually cooling the subject temperature elements and observing the indication response at 1C21. Testing encompassed all channels within RWCU leak detection, including six ambient temperature channels and six differential temperature channels.

Since cooling of the above temperature elements would most likely result in corresponding PCIS Group 5 isolation signals, the Reactor Water Cleanup system was secured and isolated prior to commencement of this test.

Summary of Safety Evaluation

As discussed in UFSAR Section 7.3, the purpose of this portion of the Steam Leak Detection System (SLDS) is to detect steam and/or water leakage from the RWCU system and to then initiate isolation of the system to maintain the integrity of the nuclear system process barrier. Since the RWCU system was isolated prior to commencement of this test, the test had no detrimental effect on the performance of the RWCU system portion of the SLDS.

The role of the SLDS in mitigating the consequences of a RWCU leakage event was not reduced by this test. All proposed testing was conducted within the operating range of the equipment in question. This test consisted of varying the temperature input to a number of temperature elements and observing their response. Since the normal function of these elements is to respond to temperature changes, the possibility of creating an equipment malfunction within the SLDS different from those already analyzed did not exist in this test.

The test did not degrade either SLDS reliability or the reliability of the RWCU system isolation function since all isolation valves were in the closed position during the test. In addition, there is no other equipment in this area (RWCU Pump and Heat Exchanger rooms) that is important to safety that could have been affected by this test.

The malfunction of any of the equipment related to this test (including SLDS temperature elements, SLDS logic, PCIS logic, RWCU system isolation valves) would not result in increased consequences since the affected equipment was in an isolated (safe) condition during the test. An equipment malfunction occurring during this test would not cause any of the RWCU system isolation valves to reposition.

Technical Specification Basis 3.2 for the SLDS system is that it should initiate action to mitigate the consequences of accidents which are beyond the operator's ability to control. The action to be initiated by the portion of SLDS that was evaluated in this test was the isolation of RWCU. Since RWCU was isolated for the duration of this test, the margin of safety provided in this technical specification basis was not reduced.

The implementation of Special Test Procedure #172 did not constitute an unreviewed safety question.

SpTP No. 173 Standby Filter Unit Backflow Test

The purpose of this test was to measure flow through the idle unisolated Standby Filter Unit (SFU) train during operation of the redundant train and verify that the operating train could deliver 1000 cfm +/- 10%. This testing became necessary since it was found that backdraft dampers intended to prevent backflow through the idle SFU train had not been installed.

Summary of Safety Evaluation

The effects of this test on various UFSAR accident analyses were considered. The probability of events resulting in a nuclear system pressure increase remained unchanged since this testing did not affect MSIV, bypass or stop valve logic, safety valve setpoints or generator trip/load reject logic. The probability of an RPV coolant inventory decrease was not increased since the test did not involve opening a pressure boundary, nor did it affect feedwater operation or emergency injection system logic.

The DAEC NSOA discusses four accidents for which the SFUs are required to operate in order to mitigate the consequences- rod drop, fuel handling, pipe break inside containment, and pipe break outside containment. During a portion of the test, the idle SFU train was unisolated which could render both trains unable to perform their design function if operator actions were not taken. The lineup could be restored to normal at any time from the control room with minimal effort. Therefore, the effects of an accident were not changed.

Consideration was given to the probable increase in fan motor current due to the backflow of air. The increase in motor current would be negligible, therefore the probability of a motor malfunction was not increased. Should a motor failure have occurred during testing, overload protection would have operated to protect the motor and the circuit.

This testing was confined to the SFU trains of the control building ventilation systems. Safety-related components not associated with the SFUs were not operated or manipulated in any way. Consequently, this testing did not create the possibility of an unanalyzed accident.

As discussed previously, during the course of this testing, the SFU was placed in a configuration for which its ability to deliver 1000 cfm +/- 10% was not assured (with idle train isolation valves open). This was not an operability concern, nor did it contribute to a reduction in the margin of safety as defined in TS. System operability and the safety margin were maintained since operator action could be taken to restore the idle SFU train to its normal shutdown line-up. Direction was provided in the SpTP on how to restore the system.

The performance of this procedure did not involve an unreviewed safety question.

SECTION C - EXPERIMENTS

This section has been prepared in accordance with the requirements of 10 CFR Part 50.59(b). No experiments were conducted during calendar year 1991.

SECTION D - SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES

This section has been prepared in accordance with the requirements of Technical Specification 6.11.1.e. There were no safety/relief valve failures or challenges in calendar year 1991.

SECTION E - FIRE PLAN CHANGES

The information contained in this section identifies, briefly describes and provides assurance that changes made to the DAEC Fire Plan during the calendar year 1991 did not alter our commitment to the NRC guidelines contained in "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

Revision No.	Description of Change
24	This revision removed redundant sections and made miscellaneous editorial changes. Technical Specification requirements were added to the DAEC Fire Plan.
25	This revision defined the DAEC Fire Brigade as six Shift Fire Brigades, and modified the fire drill requirements accordingly. It also added a Quality Assurance review.
26	This revision changed the wording in the operability requirements to incorporate RTS-238, which revised the flow and discharge pressure requirements for the annual fire pump surveillance. It also made minor editorial changes. Reference to surveillance procedure numbers was deleted from paragraph 6.3.2.