**Commonwealth Edison**
Quad Cities Nuclear Power Station
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TKT-85-38

July 1, 1985

Mr. Edson G. Case, Deputy Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Case:

Enclosed please find a listing of those changes, tests, and experiments completed during the month of June, 1985, for Quad-Cities Station Units 1 and 2, DPR-29 and DPR-30. A summary of the safety evaluation is being reported in compliance with 10 CFR 50.59.

Thirty-nine copies are provided for your use.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

T. K. Tamlyn

T. K. Tamlyn
Services Superintendent

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Enclosure

cc: B. Rybak



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## Special Test 1-79

Special Test 1-79 was completed on June 13, 1985. The purpose of this test was to determine and document the relationship between air flow through the Standby Gas Treatment System train and differential pressure across the air flow orifices.

### Safety Evaluation

1. The probability of an occurrence or the consequence of an accident, or malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report is not increased because Technical Specification 3.7.B will be met at all times during the testing. Both units will be in Cold Shutdown and no fuel moves will be made.

2. The probability for an accident or malfunction of a different type train previously evaluated in the Final Safety Analysis Report is not created because the test does not modify the design of the Standby Gas Treatment System, and operability will be verified after the testing.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specification 3.7.B will be met at all times.

Modification M-4-2-83-47

## Description

Modification M-4-2-83-47 installed two 8.5 minute Auto Blowdown timers and an Auto Blowdown Inhibit keylock switch. The new timer is a result of NUREG 0737, Item II.K.3.18, and the inhibit switch is a result of the new Emergency Operating procedures. The timers start on a low-low level signal from the 287-72A through D Yarway Reactor level instruments. If Reactor level remains at or below the low-low level setpoint for 8.5 minutes, the RHR and Core Spray pumps auto-start and the Reactor will blowdown. If Reactor level goes above the low-low level setpoint before the 8.5 minutes is up, the timers auto reset. If a blowdown has occurred, the Auto Blowdown timer reset button must be depressed to reset the 8.5 minute timers. The Auto Blowdown Inhibit keylock switch will prevent an auto blowdown from occurring. When the keylock switch on the 902-3 Panel is moved to inhibit, an Auto Blowdown Inhibit alarm will be received. The Target Rock and Electromatics will open at their pressure setpoints even if the auto blowdown has been inhibited.

## Evaluation

The basis for the Technical Specification on the Auto Blowdown System is to provide vessel depressurization enabling Core Spray or LPCI protection against small pipe break independent of HPCI. This function is unchanged. The system will limit fuel cladding temperatures to well below cladding melt point and will assure that core geometry will remain intact. The circuitry changes allow Auto Blowdown for more event sequences. No single failure can prevent Auto Blowdown from depressurizing the Reactor vessel and the two channels are independent.

## Modification M-4-1(2)-80-9

### Description

This modification was originally written to replace all Safety Related hydraulic snubbers with mechanical snubbers manufactured by Pacific Scientific Company. Seven supports on Unit One and two supports on Unit Two were replaced on this modification. The remainder of the snubbers will be replaced on another modification that is implementing the piping support requirements of Bulletin 79-02 and 79-14.

### Evaluation

The design requirements of the pipe supports replaced in this modification are not changed. The mechanical snubbers will improve reliability of pipe support systems.

## Modification M-4-2-84-24

### Description

This modification inserted an Intermediate Open (IO) limit switch in series with the close signal on Limitorque Motor Operated Valves 1(2)-1402-25A, 1(2)-1402-25B (Core Spray System), and 1-2301-3 (HPCI System). This modification was initiated to prevent stem damage due to possible valve "hammering" in the closed position with a close signal present.

### Evaluation

This modification in no way affects the original design function of the Limitorque Operators or their associated systems. Valve operation is affected only in the close direction, and then only to prevent damage. Since all valves open to perform their safety functions, the probability of valve failure during an accident is not changed.

## Modification M-4-1/2-85-8

### Description

This modification was initiated to document Station Nuclear Engineering's (SNED) approval of the procurement of four Electromatic Relief Valves from Dresser Industries. This approval was required due to a material change in the new valves that were procured. This modification did not affect the installation of these valves into the system. No piping modifications were required by the initiation of this modification.

### Evaluation

The function of the Electromatic Relief Valves and the system remain unchanged, the margin of safety is not reduced. The new design valves will increase the reliability of the valves; thus, the probability of valve failure is less than or equal to the old design valves (based on an engineering evaluation).

## Modification M-4-2-85-12

### Description

The Reactor Building to Suppression Chamber Vacuum Breakers are controlled by Solenoid valves SO 2-1601-50A and SO 2-1601-50B. Due to requirements of 10 CFR 50.49, Environmental Qualification of Electrical Equipment, the existing Versa Solenoid valves were replaced with qualified ASCO, Model Number NP8344A73V valves.

### Evaluation

The qualified Solenoids perform the same function as the original Solenoids. The installation of this modification provides improved reliability during a "harsh" environment condition.

## Modification M-4-2-85-19

### Description

The HPCI drain pot containment isolation valve, CV 2-2301-34, was replaced with a new valve. Repairs to the existing valve were not possible due to deterioration of the stellite seat surface. An exact replacement was not possible. The replacement is a Kerotest lift Check Valve. The replacement was an ASME, Section XI code repair.

### Evaluation

The new valve and the old valve are both lift Check Valves, and should perform equally well in this application.

## Modification M-4-2-84-25

### Description

This modification changed the logic which trips the Drywell coolers. Prior to this modification, the Drywell coolers would only trip on a bus undervoltage. This would be undesirable because without the Drywell coolers, pressure in the Drywell could increase causing an unnecessary RHR and Core Spray pump initiation.

This modification changed the Drywell cooler trip circuitry to allow the coolers to continue to operate on a Core Spray initiation, to decrease Drywell pressure, or a loss of off-site power without an actual LOCA. With loss of off-site power and Core Spray and RHR initiation, the Drywell coolers would be shed to prevent overloading the Diesel Generators.

### Evaluation

The Drywell coolers will be shed on a loss of off-site power and Core Spray initiation to prevent an overload of the Diesel Generators. Core Spray and RHR pump logic remains unchanged. This modification will prevent an inadvertent Core Spray initiation on loss of off-site power.