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RAR-89-69

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Director of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D. C. 20555

Enclosed please find a listing of those changes, tests, and experiments completed during the month of September, 1989, for Quad-Cities Station Units 1 and 2, DPR-29 and DPR-30. A summary of the safety evaluations are being reported in compliance with 10CFR50.59 and 10CFR50.71(e).

Thirty-nine copies are provided for your use.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION

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RAR/LFD/vmk

Enclosure

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IE47
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Procedure Change QOS 2300-1

This revision provides clarification of IST requirements, system startup steps and provides for shutdown of the drywell-torus differential pressure control system during HPCI testing if drywell pressure becomes excessive.

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 this revision should decrease the probability of an accident by clarifying certain steps in the test procedure. Also, this change should ensure that high drywell pressures are avoided during HPCI testing by allowing shutdown of the drywell-torus dp control system.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because the basic method of system testing remains unchanged, therefore, no new possibility of an accident or malfunction is created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because operation of the HPCI system and drywell-torus control system remains within the requirements of Technical Specifications, therefore, the margin of safety is not reduced.

Safety Evaluation #89-335
Technical Specification Proposed Change, Section 3.6/4.6

This change adjusts the pressure-temperature operating limits for Quad Cities Unit 1 and 2 reactor vessels by updating Figure 3.6-1 and make the limits valid through 16 effective full power years. This is necessary to comply with Reg. Guide 1.99 Revision 2 (NRC Generic letter 88-11).

Removes the limitation that the reactor vessel be vented unless the reactor vessel temperature is equal to or greater than the minimum reactor pressurization temperature curve (Figure 3.6-2, DPR-29, and Figure 3.6-1, DPR-30). Additionally, these figures will be removed from the Technical Specifications.

An administrative change to correct the reactor vessel speciman withdrawal dates in table 4.6-2.

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 the pressure-temperature operating limits are adjusted to incorporate the initial fracture toughness conservatism present when the reactor vessel was new. GE's analysis (NEDO-21778-A) shows that for a control rod drop accident transient in the conditions identified for venting, that no operator actions are needed to alter the vessel conditions. For water levels as great 780 inches above the vessel bottom, a maximum vessel pressure rise of 15.8 psi was calculated. The venting requirement was a result of a postulated pressure spike of sufficient magnitude that would place the vessel in a condition that violates 10CFR50 Appendix G. GE's analysis shows that this requirement was overly conservative and restrictive. Early withdrawal of the specimens simply provided irradiation effects at a lower fluence level. There remains sufficient vessel specimens to support the requirements of Appendix H.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because the new pressure-temperature operating limits are merely an update of the old limits, no physical changes are being made. The removal of the venting requirement is only an adjustment to an overly restrictive limit which has been shown not to be needed (NEDO-21778-A). The maximum pressure spike was calculated to be 15.8 psi and as a result GE states that no operator action is needed to alter vessel conditions such as opening the vessel head vent. No new or different kind of accident is created as a result of removing the reactor vessel specimen early. The vessel specimen was subjected to a slightly lower fluence level but provides information on irradiation effects of the vessel material. There are sufficient vessel specimens remaining in the vessel to satisfy the requirements of Appendix H.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the new pressure-temperature operating limits are actually restoring the margin of safety to a level similar to when the reactor vessel was new and the fracture toughness slightly greater. Removing the venting requirements still includes an adequate margin of safety as shown by GE's analysis (NEDO-21778-A). The calculated maximum pressure rise as a result of a CRDA was 15.8 psi and thus, GE states no operation action to alter vessel conditions is needed. The margin of safety is not reduced by the early removal of the reactor vessel specimen. The specimen was used to determine the irradiation effects on the reactor vessel material. There are still enough specimens remaining to support the requirements of Appendix H.

Safety Evaluations #89-432 and #89-439
Reactor Recirculation and Reactor Water Cleanup
System Decontamination

During the Unit 1 Refuel Outage decontamination of piping associated with the Reactor vessel was performed, the Reactor Water Cleanup Piping was performed with fuel in the vessel and the vessel head removed. The decontamination chemicals did not enter the vessel during this process.

The Recirculation Pump Suction and Disharge Piping was also decontaminated. This was done with the fuel removed from the vessel. The vessel head was in place but not tensioned. Water level in the vessel was maintained below the core area of the vessel. The decontamination chemicals were flushed from the vessel prior to reloading fuel.

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 since metallurgy effects are minimal because the solvent corrosion rates are less than the original allowances. 304 stainless steel coupons were placed in the decontamination flow path and analyzed upon completion of the project for assurance of the actual corrosion rates. Water purity effects are minimal because the reactor coolant were returned to a conductivity and a TOC level that is acceptable to station chemistry.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because the effects of residual solvent in the system was determined to be negligible. Reactor Coolant is cleaned and returned to a conductivity and a TOC level which is acceptable to the station chemistry staff. Station radiation protection procedures were followed throughout the decontamination. During resin transfer to the solidification truck, the affected areas of the reactor building was evacuated. Access into the drywell during the process was strictly controlled by station health physicists. The level of the solvent in the recirculation system risers and annulus was continuously monitored. Since SMAD has reviewed the material/solvent interface for materials within the core and has accepted the solvent for use, the consequences of a failure in the level controls causing a spill into the core are negligible.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the decontamination project was performed in accordance with the existing Technical Specifications. The reactor was maintained in the shutdown or refuel mode with all interlocks in the shutdown or refuel position.

Safety Evaluation #89-437
Process Control Program for CNSI Cement Solidifcation

This provides the Process Control Program for CNSI to process LOMI decon solution on bead resin using Formula II.

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 the solidification of decon spent resins does not involve plant systems and will not increase the probability of an occurrence or the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the FSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because this procedure does not contradict FSAR Section 9. This procedure assures that the solidification is done according to a pre-approved Process Control Program.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this is in accordance with Tech Spec 6.9 and ensures this margin of safety is incorporated.

Safety Evaluation #89-445
FSAR Correction

Change FSAR Table 5.2.5 to better describe power to close 1601-21, 22, 23, 24, 56 and 60 from spring to air.

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 this safety evaluation is for a FSAR correction and does not involve any equipment, procedure, design function or operating method changes.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because the probability of human error due to misinterpretation of the FSAR is reduced,

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications are not affected.

Safety Evaluations #89-522 and #89-523

Reduce the number of temperature switches from 16 to 4 and change the trip setpoint from 185°F to 155°F on the Unit One RCIC and HPCI Turbine Area High Temperature Isolation system.

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 reducing the number of temperaure elements will not degrade the integrity of the leak detection system. Decreasing the trip level setting will reduce response time and maintain radiation releases within acceptable limits. Therefore, the probability of an occurrence or consequence of an accident is not increased.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because the modified system will still maintain one-out-of-two taken twice trip logic and separation criteria for electrical power supplies. The system will still ensure isolation in the event of an actual steam line break but should preclude spurious isolations due to small localized steam leaks. Therefore, there is no possibility for an accident or malfunction created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change requires a revision to Technical Specifications. However, the change will not reduce the effectiveness of the steam leak detection system. The modified system should increase the reliability of RCIC and HPCI by reducing the probability of sporadic isolations. Therefore, the margin of safety has not been reduced.

M-4-1(2)-84-21A and B
Safety Evaluation #89-468
HPCI and RCIC Area High Temperature
Isolation System

This modification involves decreasing the number of temperature elements from 16 to 4 and reducing the trip level setting from $\leq 200°F$ to $\leq 170°F$. The current system consists of four groups of switches at four different locations. Each group of four switches at one location is arranged in a one-out-of-two taken twice trip logic. This has resulted in spurious system isolations due to minor steam leaks at the turbine bearings. The modified system will consist of two groups of switches at two different locations. The four switches will then be arranged in a one-out-of-two taken twice trip logic. The trip level setting will be reduced to maintain system response time and limit radiation release in the event of a steam line break.

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 an analysis was performed by General Electric and a calculation performed by Impell to evaluate the HPCI and RCIC area temperature monitoring systems and proposed modification. The calculations determined that reducing the number of temperature elements will not degrade the integrity of the leak detection system. Decreasing the trip level setting will reduce response time and maintain radiation releases within acceptable limits. Therefore, the probability of an occurrence or consequence of an accident is not increased.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because the modified system will still maintain one-out-of-two taken twice trip logic and separation criteria for electrical power supplies. The system will still ensure isolation in the event of an actual steam line break but should preclude spurious isolations due to small localized steam leaks. Therefore, there is no possibility for an accident or malfunction created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the modification requires a change to Technical Specifications. However, the change will not reduce the effectiveness of the steam leak detection system. The modified system should increase the reliability of HPCI and RCIC by reducing the number of sporadic isolations. Therefore, the margin of safety has not been reduced.

Modifications M-4-1-84-027A, B, C and D

Description

General Electric identified that a potential existing of a 'b' contact bounce problem in their HGA relays during a seismic event. After review, certain important to safety HGA relays were exchanged for HFA relays. In some cases the wiring was moved from an HFA to an HGA to free up the HFA for use. No circuit logic was altered - function and operation of the system was unaffected. The modification covers the HPCI, RCIC and Core Spray systems.

Evaluation

1. The probability of an occurrence or the consequence of an accident, of malfunction of equipment important to safety as previously evaluated in the Final Safety Analysis Report is not increased because the HFA type relays, which have a higher seismic rating than the HGA relays, will now be used in place of HGA relays in safety circuits. Thus reliability is increased and the probability of a malfunction is reduced.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because this is a one-for one exchange of the function performed by existing relays. Therefore, no new malfunction is created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because since the seismic rating of the replacement HFA relay exceeds original equipment ratings, the margin of safety is not reduced.

Modification M-4-1-85-26

Description

The existing General Electric CFD Diesel Generator differential current protection relay was replaced with a new seismically qualified Westinghouse SA-1 type differential relay in order to satisfy OPEX 84-75S1. The new relay is in the same physical location (at the 4KV switchgear) as before. The relay continues to provide a trip signal to the lockout relay to disconnect an internally faulted Diesel Generator from its 4KV switchgear bus.

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 existing differential relays are being replaced with seismically qualified relays, therefore the probability of an occurrence or an accident, or malfunction of equipment important to safety as evaluated in FSAR is not increased.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because new relays have the identical system interfaces as the existing relays, therefore the possibility for an accident or malfunction of a different type than previously evaluated in the FSAR does not exist.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because new relays will provide improved reliability during seismic events, therefore the margin of safety as defined in the basis of Quad Cities Technical Specification is not reduced. The presently installed relays are non-seismic.

Modification M-4-1-87-074A and 74B

Description

These modifications replaced restricting orifice 1-3241-53A on the 'A' and 1-3241-53B on the 'B' feedwater flush line with spectacle flanges. The spectacle flange consists of a blank plate and a large bore orifice. Blank plates will be installed during normal operation. The large bore orifice will be used for flushing operations. The original restrictive orifice was sized extremely small to form a pressure barrier between the feedwater piping and the condenser since flushing was originally designed to be done using a feedwater pump. However, the restrictive orifice did not allow enough flow to provide adequate flushing of the system.

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 the feedwater flush lines are not mentioned in Section II of the FSAR which deals with the feedwater system. Since the original conditions and assumptions made in the FSAR have not been changed, the probability of an occurrence or the consequence of an accident, or malfunction of equipment important to safety as previously evaluated in the FSAR is not increased.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because this modification does not interfere with any safety-related equipment and would not fall outside any single failure event or design basis accident which has already been analyzed in the FSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because feedwater flush lines do not interact with any systems described in the Technical Specifications. Therefore, the margin of safety is not reduced.

Modification M-4-0-89-064
Safety Evaluation #89-467
Install RACS Video Capture System Modification

A RACS (Redundant Access Control System) Video Capture System will be installed to further enhance the security perimeter intrusion assessment by CCTV (Closed Circuit Television). The potential exists for a CAS (Central Alarm Station) console operator to miss an intruder on CCTV, when an intrusion alarm comes up. This RACS Video Capture System can freeze a video frame inside one second of the intrusion. The video frame, through a pair of dedicated monitors, provides the CAS console operator the reaction time capability of the existing electronic devices. A video printer will provide a hard copy of that video frame to assess and document any human intrusions.

The RACS Video Capture System consist of six devices (two monitors, one video printer, two video digitalizer boards, one video switch board) and two manual switches (for transferring the communications lines). The first three devices mentioned operate at 120V and will be connected to the existing security system UPS.

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 the reliability of the CCTV and the Perimeter Intrusion Detection System will be enhanced by the addition of this new RACS Video Capture System. However, this would have no bearing on the probability or consequence of an accident or malfunction of equipment important to safety, since analyses take no credit for this security system.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because this modification does not alter the description of any equipment or systems important to safety as previously evaluated in the FSAR/UFSAR. Installation of the new PACS VCS involves non-safety-related equipment which will be located remote from any safety-related system.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this modification does not alter or affect any equipment described in the Technical Specification. Therefore, the margin of safety will not be reduced.

Modification M-4-1(2)-89-152
Safety Evaluations #89-519 and #89-520

Description

This modification is being installed as a corrective action per Potential Significant Event Report PSE-89-006, titled "New Fuel Bundle Drop While in Fuel Pool". The PSE occurred on September 21, 1989 at Quad Cities Unit 1. See PSE-89-006 for details.

The modification will install an additional electrical interlock that will prevent raising the hoist on the fuel moving machine while the hoist is loaded unless the grapple is fully closed and in the engage position.

The modification will be contained in the G.E. fuel moving panel located on the refuel bridge.

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 the modification will add an additional feature to the interlock system to enhance the safe movement of fuel.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report is not created because this modification will add an additional interlock protection to an evaluation condition.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this modification will increase the margin of safety while moving fuel.