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MAY 13 2005

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Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUPPLEMENTAL REPORT OF CHANGES, TESTS AND EXPERIMENTS
HOPE CREEK GENERATING STATION
DOCKET NO. 50-354

By letter dated March 31, 2005, PSEG Nuclear forwarded a summary of changes, tests, and experiments pursuant to the requirements of 10CFR50.59(d)(2). The letter also stated that a supplementary report would be provided by May 15, 2005 conveying the review results for earlier design changes which may have been reviewed against either the three criteria of 10CFR50.59(a)(2) which were in effect prior to March 13, 2001, or the eight criteria of 10CFR50.59(c)(2) subsequent to that date.

Attached are the results of that review.

Should you have any questions, please contact Ralph Donges at (856) 339-1640.

Sincerely,

Christina L. Perino
Regulatory Assurance Director

Attachment

IE47

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SUMMARY OF CHANGES TESTS AND EXPERIMENTS
HOPE CREEK

Revised P&IDs M-16-1 and M-23-1 to Reflect Condensate Demineralizer Sample Element Details (H00-058)

This Safety Evaluation supported an as-built document only change DCP (80019543) that changed the depiction of the Condensate Demineralizer System sampling lines as shown on UFSAR Figure 10.4-4, Sheet 1 of 2.

The basis for the two Condensate Demineralizer Sample Points was not specifically described in the UFSAR, but the lines and valve position were shown on two P&IDs, one of which is a UFSAR Figure. Neither the details of the location of the Condensate Demineralizer effluent sample points nor the fact that there were two sample points on each Condensate Demineralizer outlet were discussed.

The Condensate Demineralizer Sample Points, the sample piping, and the licensing basis for the procedure for Condensate Demineralizer operation and sampling were not changed by this proposal. Only the depiction and the normal position of the alternate sampling valves were changed. This change also allowed deleting the red blocking tags that limited chemistry department operation of the sample valves. The proposal did not change the Technical Specifications and no unreviewed safety question was involved.

Re-alignment of SW Dewatering Pump Isolation Valve (H00-044)

This change evaluated re-alignment of Service Water dewatering pump discharge isolation valve 1-EA-V618 from normally open to normally closed. Due to the physical location of the RACS heat exchangers being below the elevation of the Service Water system piping as it enters and leaves the Reactor Building, a dewatering tank and pump was provided. This allowed for pumping water from the Service Water side of the heat exchangers back to the Service Water system piping to be drained to either the cooling tower basin or to the river. The relief valves and drains from the SACS heat exchangers also drain to the Service Water dewatering tank. This design minimizes the need to discharge non-radioactive brackish river water into building sumps and into the liquid radwaste processing system.

A check valve is designed on the discharge of the Service Water dewatering pump to prevent service water effluent from the RACS heat exchanger from filling the dewatering tank when the Service Water system is in service. Historically, this check valve has required extensive maintenance to keep it operational. Therefore, discharge isolation valve 1-EA-V618 was being maintained closed and controlled by the tagging system. This, however, was inconsistent with procedures and programs for controlling plant equipment. Tagging was intended to provide personnel safety, and was not an alternate method of configuration control.

Because it was operationally preferred to operate the dewatering system in manual, this change re-aligned the 1-EA-V618 to normally closed when the system is not in service for dewatering.

Potential Operation of the Auxiliary Boiler System with Low Levels of Radioactivity (H00-041)

This safety evaluation supported the potential operation of the auxiliary boiler system with low levels of radioactivity. UFSAR Section 9.5.9 describes the design bases of the auxiliary boiler system. The auxiliary boiler system is designed and normally operated to supply low pressure, uncontaminated, saturated steam to various start-up and plant service functions. Operating the auxiliary boiler system with low levels of radioactive contamination changed the facility as described in the SAR.

The use of a potentially contaminated auxiliary boiler system is administratively controlled to prevent uncontrolled release of radioactivity to the environment. The administrative controls, including caution tags and a sampling and monitoring program, were established while the system was decontaminated, while corrective actions to address the suspected cause of contamination were taken, and until the system was declared free from contamination.

By reference to engineering evaluation H-1-FA-NEE-1364, the safety evaluation reviewed the operation of the potentially contaminated auxiliary boiler system using the criteria in NRC IE Bulletin 80-10, "Contamination of Non-radioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to the Environment".

The safety evaluation determined that the potential consequences, under normal operating and postulated accident conditions, were well below the applicable regulatory limits and As Low As Reasonably Achievable (ALARA).

The safety evaluation concluded that the proposal did not require a Technical Specification change and did not present an unreviewed safety question.

SUMMARY OF CHANGES TESTS AND EXPERIMENTS
HOPE CREEK

Main Turbine Vibration Probes (H00-021)

The change added vibration probes to each of the 12 Main Turbine Generator journal bearing covers and provided for the remote readout of signals produced. Existing plant processes and systems were unaffected by the change.
This change did not involve an unreviewed safety question because the proposed equipment did not interface with any existing process or equipment. The change did not introduce a new, credible accident or malfunction nor did it affect any accident analysis contained in the UFSAR.

Reactor Well Shield Plug Removal Before Cold Shutdown (H00-018)

The proposal involved removal of the reactor well cover plugs before the Main Turbine is taken offline. It was proposed that the upper layer of plugs be removed with the reactor power at 20% or less and that the lower layer of plugs be removed in cold shutdown.

Engineering evaluation H-1-KE-MEE-1349 documented the results of an analysis of issues relating to removing reactor well shield plugs before cold shutdown. The engineering evaluation assessed internally generated missiles, missiles generated by natural phenomena, IPEEE results for turbine missiles, heavy load risk, pipe rupture protection, plant ventilation, fire protection, post accident operator access to plant areas, DBA radiological consequences, environmental qualification, rad monitoring channel setpoints and detection capability.

The proposal did not require a change to the Technical Specifications and no unreviewed safety question was involved.

Abandonment of Construction Fire Water Pumps and Construction Service Water Pumps (H00-017)

During the construction of the Hope Creek station, fire water and fresh water (non-potable) needs were supported by a construction diesel engine driven fire pump, an electric driven fire pump and two service water pumps. Based on a review of UFSAR Figure 9.5-133 historical revisions, it is evident that the construction pumps were considered to be temporary.

After the construction phase, blind flanges were intended to be installed to isolate the construction equipment from the permanent plant equipment. However, the flanges were never installed and as an interim measure, the construction equipment was mechanically and electrically isolated and controlled by the tagging program using red blocking tags.

The safety evaluation supported removing Notes 23 and 25 from UFSAR Figure 9.5-133 (P&ID M-22-0) and realigning isolation valves in the flow path to the construction fire water and construction service water pumps to normally locked closed in lieu of installing blank flanges as originally intended. The evaluation concluded that no change to the Technical Specifications was required and that no unreviewed safety question was involved.

Refueling Platform Upgrade (H02-005)

This design change replaced the existing control systems on the Refueling Platform, H-1-KE-10-S-238, with an industry-proven, reliable Programmable Logic Controller (PLC) unit. Along with the PLC controls, the platform was also fitted with an IBM compatible computer allowing semi-automatic and automatic positioning of the bridge, trolley and hoist. The associated peripheral control devices, such as encoders, motor drives, etc., were replaced with reliable, industry-proven, off-the-shelf products. The 480 VAC circuit breaker and scheduled feeder cable were upgraded due to the increase in electrical loading.

The Refueling Platform controls and hardware replaced by this modification were non-safety related components. The Refueling Platform itself is Seismic Category 1, passive essential as described in section 9.1.4 of the UFSAR.

The effect of the proposed change on the design functions was that the logic for these functions would now be accomplished via a PLC as opposed to through the use of relays which was the method at that time.

Although this modification did not adversely affect a design function, use a new or revised evaluation methodology, involve a test or experiment not described in the UFSAR, affect the design basis limit for a fission product barrier, or involve a change to the Technical Specifications, it did involve a digital upgrade which was held to fundamentally alter the existing means of performing or controlling design functions. Since the modification now allowed Refueling Platform controls to be operated in semi-automatic and automatic modes, a full evaluation was performed.

A Failure Modes and Effects Analysis (FMEA) was performed on the new system. No new failures were identified and the identified system failures were all bounded by failures that were already present with the existing hardware.

The safety evaluation concluded that no change to the Technical Specifications was required and that no unreviewed safety question was involved.

Radiation Consequences of Removing Reactor Well Shield Plugs at Hot Shutdown (H01-013)

At that time, HCGS removed the upper layer of reactor cavity shield plugs at 20% power or less and removed the bottom layer of shield plugs at cold shutdown. Calculation H-1-KE-MDC-1898 proposed to remove the bottom layer of shield plugs at hot shutdown.

The expected effects of removing the bottom layer of shield plugs at hot shutdown were (1) shorter refueling outage times, (2) minimally increased dose rates on the refueling floor (eventually returning to previous levels as time passes), and (3) increased design basis accident dose and dose rates if the accident occurs after the bottom layer is removed.

Calculation H-1-KE-MDC-1898 showed that post-accident control room, site boundary, and LPZ doses were not affected by this change. This calculation also showed no change in the normal refueling floor dose rates. However, the calculation showed that the post-accident refueling floor dose and dose rate would increase, possibly having an effect on equipment qualification.

The total integrated gamma dose for refueling floor safety related equipment having post-accident functional requirements was below the specified levels listed on the Equipment Evaluation Summary Sheets. Therefore, the removal of the bottom layer of shield plugs at hot shutdown would not impact the environmental qualification of equipment on the refueling floor.

The evaluation concluded that no change to the Technical Specifications was required and that no unreviewed safety question was involved.

Replacement of SACS Fuel Pool Cooling Inlet and Outlet Cross-tie Valves (H00-004)

The rubber-seated butterfly valves in the SACS spent fuel pool cooling heat exchanger inlet and outlet lines were replaced with metal-seated Atwood and Morrill Tricentric valves. To ensure a tight seal in both directions, the closing torque was increased. The change included changes to the gearbox, gear ratio, drive sleeve, adaptor spline, seismic plate and spring pack. The control circuit was also changed from limit seated to torque seated.

The safety evaluation concluded that no change to the Technical Specifications was required and that no unreviewed safety question was involved.