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MAR 1 4 2003

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LRN-03-0116

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

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

Pursuant to the requirements of 10CFR50.59(d)(2), this correspondence forwards a summary of changes, tests and experiments implemented at Hope Creek during the period March 1, 2001 through February 28, 2003. The report includes changes reviewed either in accordance with 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.

Should you have any questions, please contact Carl Berger at (856) 339-1432.

Sincerely,

G. Salamon
Manager Nuclear Safety and Licensing

CLB
Attachment

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SUMMARY OF CHANGES TESTS AND EXPERIMENTS
HOPE CREEK GENERATING STATION

DESIGN CHANGES

80000916, Service Water Intake Structure Public Address System Upgrade (50.59 Evaluation H2000-038)
This modification expands the existing public address system within the Service Water Intake structure by adding handsets, speakers and associated components. The expanded PA System is compatible with the existing system. All added components are seismically supported to prevent their failure from impacting adjacent safety related equipment.

The design and licensing basis for the PA System is to provide onsite communication between various plant locations as described in UFSAR Section 9.5.2. This modification does not alter the design and licensing basis. There is no increase in the probability or consequences of any accident or malfunction, no new accidents or malfunctions created and the margin of safety is not reduced. Therefore, this modification does not represent an Unreviewed Safety Question.

80016651, Chiller Motor CRIDS Point Deletion (50.59 Evaluation H2001-006)
This deletes a plant computer point from the Control Room Chilled Water System chiller compressor motor. The CRIDS point monitors the motor winding temperature from an RTD that has failed. The temperature monitoring is for indication only, and does not initiate any automatic or operator action. All available spare RTDs on the motor do not work and cannot be immediately replaced because they are physically embedded into the motor windings. This chiller runs at approximately 15-20% of rated capacity, and therefore motor RTD temperature indication is not a critical parameter for system trending.

This modification does not alter the design and licensing basis. There is no increase in the probability or consequences of any accident or malfunction, no new accidents or malfunctions created and the margin of safety is not reduced. Therefore, this modification does not represent an Unreviewed Safety Question.

80007100, Add a Time Delay to the RCIC Steam Supply Pressure Low Isolation Signal (50.59 Evaluation H2000-46)
This modification adds a four second time delay to the RCIC isolation control scheme to eliminate spurious RCIC system isolations during system warm-up due to condensation flashing in the horizontal flow venturi. A four second delay is longer than the observed field data for low steam supply pressure due to the flashing of the condensate. The time delay is implemented by replacing existing control relays with time delay relays, inserted into the existing relay sockets.

A Technical specification change is not required since the Hope Creek Technical Specifications do not specify a required time response for the low steam supply pressure isolation. The UFSAR shows “NA” for time response for this parameter.

Therefore, there is no impact on the design basis. There is no increase in the probability or consequences of any accident or malfunction, no new accidents or malfunctions created and the margin of safety is not reduced. Therefore, this modification does not represent an Unreviewed Safety Question.

ENGINEERING CALCULATIONS

H-1-KE-MDC-1898, Rev. 0, Radiation Consequences of Removing Reactor Well Shield Plugs Before Cold Shutdown (50.59 Evaluation H2001-0013)
HCGS currently removes the upper layer of reactor cavity shield plugs at 20% power or less and removes the bottom layer of shield plugs at cold shutdown. This calculation evaluates removal of the bottom layer of shield plugs at hot shutdown. Removal of the bottom layer of shield plugs at hot shutdown will cause an increase in both the normal and accident radiation level on the refueling floor. However, control room, site boundary and LPZ accident doses are unchanged. The radiological effects of change have been evaluated and satisfy the criteria for less than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The shield plugs also function as a barrier for turbine missiles. The activity will take place at hot shutdown when the turbine is off-line and not capable of generating missiles, and will therefore have no impact on this design function. The heavy load drop event was evaluated to confirm that no breach of the drywell head boundary could occur.

No change in Technical Specifications is required and there is not more than a minimal increase in the frequency or consequences of an accident or in the likelihood of occurrence or consequences of a malfunction. The analysis does not create the possibility of a new accident or malfunction with a different result, design basis limits of fission product barriers are unaffected and the methods of analysis are consistent with the UFSAR. Therefore, the change may be implemented without obtaining a License Amendment.

HCT.8-014, Rev. 0, Hope Creek, Evaluation of Compensatory Action Required due to Degraded Rod Block Monitor Channel, (50.59 Evaluation H2002-002)
This change institutes a compensatory action for withdrawal of any control rod due to the degraded condition of Rod Block Monitor (RBM) channel "A" caused by the failure of local power range monitors (LPRM). Specifically, a configuration of LPRMs for RBM channel "A" in four strings is outside the bounds of the standard reload licensing evaluation for the Rod Withdrawal Error (RWE) transient. The degraded channel "A" has no impact on rods which are fully withdrawn. Thus, it could not be assumed that the remaining single "C" level LPRM is sufficient for the RBM to perform its design function.

The activity is to add an administrative control on MFLCPR (Maximum Fraction to the Limiting Critical Power Ratio) so that the MFLCPR is confirmed to be less than 0.89 prior to any control rod withdrawal. This value of MFLCPR is conservative for both fuel types and implementation of this compensatory action would ensure that specified acceptable fuel design limits are not exceeded, even if RBM channel "A" failed to perform its design basis function.

No change in Technical Specifications is required and there is not more than a minimal increase in the frequency or consequences of an accident or in the likelihood of occurrence or consequences of a malfunction. The administrative controls do not create the possibility of a new accident or malfunction with a different result, design basis limits of fission product barriers are unaffected and the methods of analysis are consistent with the UFSAR. Therefore, the change may be implemented without obtaining a License Amendment.

H-1-AB-MDC-1854, Rev. 0, Steam System Piping Break Outside Containment; H-1-BG-MDC-1859, Instrument Line Pipe Break; H-1-AE-MDC-1868, Feedwater Line Break Accident Outside Primary Containment Analysis Reconstitution (50.59 Evaluation H2001-009)

Certain accident radiological consequence analyses described in UFSAR Section 15 could not be retrieved. This activity reconstitutes those analyses and updates the UFSAR to be consistent with documented radiological consequences, including incorporation of the analysis results. They are concerned with the radiological consequences of postulated accidents and do not affect design functions of any structures, systems, or components.

No new methods of analysis were used. The TACT5 program from the HABIT computer code has been previously approved by the NRC for a similar application. No change in Technical Specifications is required and there is not more than a minimal increase in the frequency or consequences of an accident or in the likelihood of occurrence or consequences of a malfunction. The reconstituted analyses do not create the possibility of a new accident or malfunction with a different result and design basis limits of fission product barriers are unaffected. Therefore, the change may be implemented without obtaining a License Amendment.

HCA.8-0008, Rev. 0, HCGS Cycle 10 Core Design and 1.4% Power Uprate (50.59 Evaluation H2001-010)

This analysis supports an upgrade of the Hope Creek Generating Station (HCGS) Cycle 10 reload analysis by 1.4% power to 3339 MWt. The reload core was previously evaluated to support full power operation at 3293 MWt. The core design continues to meet design and licensing requirements.

NRC approved Westinghouse BWR Reload Methodology, which has been determined applicable to HCGS, was utilized to perform the evaluations and analyses. The reactivity characteristics of reload core meet shutdown margin design and standby liquid control system capability requirements consistent with existing UFSAR assumptions.

All relevant UFSAR Anticipated Operational Occurrences (AOOs) have been evaluated relative to Specified Acceptable Fuel Design Limits (SAFDLs) and demonstrate compliance with requirements. The values of the Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR) and Maximum Planar Linear Heat Generation Rate (MAPLHGR) limits will be in the Core Operating Limits Report (COLR) to ensure compliance with the Safety Limit MCPR and fuel rod thermal mechanical SAFDLs.

The design basis accidents were reviewed, with a conclusion that acceptance criteria are met consistent with existing UFSAR assumptions.

There is not more than a minimal increase in the frequency or consequences of an accident or in the likelihood of occurrence or consequences of a malfunction. The analysis does not create the possibility of a new accident or malfunction with a different result, design basis limits of fission product barriers are unaffected and the methods of analysis are consistent with the UFSAR. All design basis parameters related to core reload design continue to be satisfied. However, implementation of this change is contingent upon approval of license amendment request LCR H00-05, granted as Amendment 131, for the power uprate itself.

HCA.8-0009, Rev. 0, Hope Creek Cycle 11 Core Design (50.59 Evaluation H2001-011)
This analysis supports the Hope Creek Generating Station (HCGS) cycle 11 reload core. The cycle 11 core is designed to operate at a rated power of 3339 MWth.

Westinghouse CE Nuclear Power LLC, using NRC approved methods, performed the reload licensing analyses using generic methodology determined applicable to HCGS in Cycle 10.

The reactivity characteristics of the cycle 11 core have meet shutdown margin design and standby liquid control system capability requirements consistent with existing UFSAR assumptions.

All UFSAR Anticipated Operational Occurrences (AOOs) affected by the cycle 11 reload core have been evaluated relative to the Specified Acceptable Fuel Design Limits (SAFDLs) and demonstrate compliance with requirements. The values of the Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR) and Maximum Planar Linear Heat Generation Rate (MAPLHGR) limits will be in the Core Operating Limits Report (COLR) to ensure compliance with the Safety Limit MCPR and fuel rod thermal mechanical SAFDLs.

The design basis accidents were also reviewed relative to the cycle 11 reload core design with a conclusion that acceptance criteria are met consistent with existing UFSAR assumptions.

No change in Technical Specifications is required and there is not more than a minimal increase in the frequency or consequences of an accident or in the likelihood of occurrence or consequences of a malfunction. The analysis does not create the possibility of a new accident or malfunction with a different result, design basis limits of fission product barriers are unaffected and the methods of analysis are consistent with the UFSAR. Therefore, the change may be implemented without obtaining a License Amendment.

UFSAR CHANGES

HCN 00-57, Rev 0, Change to UFSAR Description of Suppression Pool Cooling (50.59 Evaluation H2000-057)
This change notice modifies the description of the suppression pool cooling mode of Residual Heat Removal (RHR) in chapters 6.2 & 7.3 so they are consistent with each other and aligned with latest industry insights.

The design basis of the system is to remove heat from the suppression pool. Industry information suggests changes in operational sequences to avoid possible water hammer. Since the sequence is not critical, these changes do not affect the design function.

This UFSAR change does not increase in the probability or consequences of any accident or malfunction, no new accidents or malfunctions created and the margin of safety is not reduced. Therefore, this change does not represent an Unreviewed Safety Question.

PROCEDURES

Hope Creek ODCM, Rev. 19, Offsite Dose Calculation Manual, (50.59 Evaluation H2001-008)
The Offsite Dose Control Manual (ODCM) has been revised to more accurately reflect the previous relocation of the Radioactive Effluent Technical Specifications (RETS) to the ODCM in accordance with Technical Specification Amendment 121. The revision includes a requirement to analyze liquid and gaseous effluent grab samples at the lower limits of detection rather than for gross radioactivity; modification of Site Related Ingestion Dose commitment Factors and Pathway Dose Factors for Atmospheric Releases; and incorporation of more recent information of isotopic distribution in liquid effluents. This revision also changes to the default

setpoints for the liquid radiation monitoring instrumentation and changes the dose conversion factor applied to the Simplified Liquid Effluent Dose Calculation.

The ODCM revision does not reduce the accuracy or reliability of dose calculations or setpoint determinations and remains within the Hope Creek design and licensing basis. There is no increase in the probability or consequences of any accident or malfunction, no new accidents or malfunctions created and the margin of safety is not reduced. Therefore, this ODCM revision does not represent an Unreviewed Safety Question.

HC.RE-RA.ZZ-0008(Q), Rev. 3, Shutdown Margin Test: Local Criticals (50.59 Evaluation H2001-012)
This procedure has been revised to govern local criticality tests at Hope Creek with a fully loaded reactor core. The local criticality testing involves achieving criticality by pulling control rods in a sequence that does not comply with the Banked Position Withdrawal Sequence (BPWS) constraints, which requires a supplemental analysis to demonstrate that fuel failure would not occur in the event of a Control Rod Drop Accident (CRDA).

This testing is not currently described in the UFSAR, nor is it bounded by any other tests. All systems involved in the testing will be operated within the requirements of the Technical Specifications and approved system design. The procedure contains prerequisites to assure that analyses are completed prior to the testing to demonstrate that the design function of the fuel and fuel clad are not affected.

No change in Technical Specifications is required and there is not more than a minimal increase in the frequency or consequences of an accident or in the likelihood of occurrence or consequences of a malfunction. The analysis does not create the possibility of a new accident or malfunction with a different result, design basis limits of fission product barriers are unaffected and the methods of analysis are consistent with the UFSAR. Therefore, the change may be implemented without obtaining a License Amendment.

HC.OP-SO.CG-0001(R), Rev. 19, Condenser Air Removal System Operation (50.59 Evaluation H2002-001)
This operating procedure revision allows use of the mechanical vacuum pumps at reduced flow rates, with known or suspected fuel damage, during shutdown and startup. If high radiation is detected in the main steam lines, the pumps are normally tripped and the suction valves would automatically close. In order to allow continued operation of the mechanical vacuum pump and prevent the south plant vent radiation monitor from alarming, the suction valves will be throttled by jacking, which defeats their automatic closure capability.

No change in Technical Specifications is required and there is not more than a minimal increase in the frequency or consequences of an accident or in the likelihood of occurrence or consequences of a malfunction. The change in operation does not create the possibility of a new accident or malfunction with a different result, design basis limits of fission product barriers are unaffected and the methods of analysis are consistent with the UFSAR. Therefore, the change may be implemented without obtaining a License Amendment.