December 8, 2000

Mr. R. P. Powers
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: D. C. COOK NUCLEAR POWER PLANT- NRC INSPECTION REPORT 50-315/00-21(DRS); 50-316/00-21(DRS)

Dear Mr. Powers:

On December 4, 2000, the NRC completed a team inspection of modified Unit 1 safety-related systems and components required for accident mitigation. The team also inspected evaluations of changes, tests, and experiments at both Units. The enclosed report documents the inspection findings which were discussed on November 17, 2000, and December 4, 2000, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The team reviewed selected procedures, and records, observed activities and interviewed personnel. Further, the team noted that your staff has made progress in preparing the plant for restart as measured by the completion of sufficient corrective actions to allow the NRC to close seven Restart Action Matrix items.

Based on the results of this inspection, the team identified one issue of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the D.C. Cook facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Geoff Grant, Director
Division of Reactor Projects

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: Inspection Report 50-315/00-21(DRS);
50-316/00-21(DRS)

cc w/encl: A. C. Bakken III, Site Vice President
J. Pollack, Plant Manager
M. Rencheck, Vice President, Nuclear Engineering
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/00-21(DRS); 50-316/00-21(DRS)
Licensee:	American Electric Power Company 1 Cook Place Bridgman, MI 49106
Facility:	D. C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	October 23 - December 4, 2000
Inspectors:	M. Holmberg, Reactor Engineer (Team Lead) A. Dunlop, Reactor Engineer J. Lennartz, Senior Resident Inspector P. Lougheed, Reactor Engineer T. Scarbrough, Office of Nuclear Reactor Regulation D. Schrum, Reactor Engineer W. Scott, Reactor Engineer
Approved by:	J. Jacobson, Chief, Mechanical Engineering Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas) reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 05000315-00-21, IR 05000316-00-21, on 10/23-12/4/2000, American Electric Power Company, D. C. Cook Nuclear Power Plant Units 1 and 2. Permanent Plant Modifications, Evaluation of Changes, Tests, or Experiments.

The inspection was conducted by inspectors based in the Region III office, a Senior Resident Inspector, and a staff member from the Office of Nuclear Reactor Regulation. The inspection identified one Green finding, which was a Non-Cited Violation. The significance of most/all findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

Green. During review of a design change, the team identified improperly set relief valves installed in two Unit 1 motor operated valves, which was considered a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control."

The licensee had not yet operated Unit 1 with this design change and for the current plant mode, operability of these valves was not required. Therefore, this finding was determined to be of very low safety significance. This issue was considered more than minor, because if it was left uncorrected, it could have impacted the function of these valves, which affect safe operation of the plant at power (Section 1R17.1.b.2).

TBD. During review of a design change, the team identified that the licensee had not verified the set point of relief valves installed in two Unit 2 motor operated valves. The operation of Unit 2 during the current operating cycle, with this inadequate design change, is an Unresolved Item pending review of the licensee's evaluation for past operability of these valves (Section 1R17.1.b.2).

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems and Barrier Integrity

1R02 Evaluations of Changes, Tests or Experiments (IP 71111, Attachment 2)

.1 RAM Item 7.1 - Evaluate the Licensing Basis Program Controls for Safety Evaluations

a. Inspection Scope

Issues requiring inspection/resolution prior to restart of the Cook Plant have been identified in the Restart Action Matrix (RAM) approved by the NRC Manual Chapter 0350 Oversight Panel. The team evaluated the licensee's program controls for 10 CFR 50.59 safety reviews (RAM Item 7.1). This review included licensee procedural controls and personnel training and qualification requirements associated with performing safety evaluations in accordance with 10 CFR 50.59. The review also included verification of actual training and qualifications for a selected sample of individuals who performed 10 CFR 50.59 screenings and evaluations.

The team's review included in excess of 20 safety evaluations performed pursuant to 10 CFR 50.59. These evaluations related to permanent plant modifications, setpoint changes, procedure changes, changes to the updated final safety analysis report or technical specification bases, and changes that involved special tests. The team also reviewed in excess of 40 screenings where the licensee had determined that a 10 CFR 50.59 safety evaluation was not necessary. The team reviewed these screenings to confirm that a complete 10 CFR 50.59 evaluation was not necessary.

b. Findings

The licensee's program for evaluating changes, tests, and experiments in accordance with 10 CFR 50.59 was described in plant procedure PMP-1040-SES.001 "Safety Screenings/Evaluations." The team determined that the licensee's program met the current industry guidance and was in conformance with the requirements of 10 CFR 50.59. The licensee's safety evaluation training program was described in TS-C-CS44 "Enhanced 10 CFR 50.59" and TS-O-0003 "Perform Safety Screenings/Evaluations (50.59)." This procedure provided controls of adequate depth and scope to ensure that the requirements of 10 CFR 50.59 were understood by the licensee's staff participants. For a sample of 10 CFR 50.59 preparers or reviewers, the team confirmed that each had attended the required training classes and had the necessary experience and background required by the licensee's program.

The completed 10 CFR 50.59 evaluations were of good quality and adequately addressed the basis of why an NRC review prior to implementation was not required. Further, the safety screenings reviewed provided an adequate basis to justify why a full 10 CFR 50.59 safety evaluation was not necessary.

No findings of significance were identified. Therefore, based on reviews of 10 CFR 50.59 evaluations and screenings that demonstrated the adequacy of the licensee's safety evaluation program controls, RAM Item 7.1 was closed.

1R17 Permanent Plant Modifications (IP 71111, Attachment 17)

.1 Review of Recent Permanent Plant Modifications

a. Inspection Scope

Issues requiring inspection/resolution prior to restart of D. C. Cook have been identified in the RAM approved by the NRC Manual Chapter 0350 Oversight Panel. The team reviewed 19 design changes and supporting calculations to evaluate the licensee's resolution of RAM Items 2.1, 3.1, 3.2, 4.1, 5.1 and 6.1. These changes involved modifications of safety related systems used in accident mitigation. The team's review focused on the system design requirements, licensing bases, and confirming that the system safety functions were not adversely affected by the modifications.

b. Findings

b.1 RAM Item 2.1 - Evaluate Modifications to the Unit 1 Recirculation Sump to Address Previous Deficiencies

1-DCP-436 Remove the Internals of the Containment Air Recirculation/Hydrogen Skimmer (CEQ) Room Floor Drain Check Valves

In this design change, the internals from three check valves in drain lines for the Unit 1 CEQ fan rooms were removed to eliminate a postulated failure of the check valves to open. These drain lines ensure that containment spray run-off collected in the fan rooms reach the annulus sump, and then flow into the lower containment sump to contribute additional sump water inventory following a loss of cooling accident (LOCA). These drains also ensure that the post LOCA flood up levels in the fan rooms do not reach levels which could affect safety related equipment needed for accident mitigation. The Performance Assurance department had previously identified errors in a supporting calculation which determined the maximum flood level of the CEQ fan rooms. Based on a draft of the revised calculation, the team determined that these errors had been appropriately resolved. Further, these errors did not change the original conclusion that safety related equipment would not be affected by the maximum possible CEQ room flood up level. The team also confirmed that the in-process flow testing documented in job order 00307068 demonstrated that the CEQ drain lines were not blocked.

1-DCP-634 Modify CEQ Fans to Start on a Phase A Containment Isolation Signal

In this design change, the CEQ fan start logic was modified to delay the start of all the CEQ fans. This logic change reduced the fan start actuation from 9±1 minutes to 120±12 seconds to increase the quantity of ice melt from the ice condenser during small and medium break LOCAs to ensure that sufficient water is available in the sump. The team reviewed supporting calculations and found them acceptable. The team also reviewed the proposed post-modification test and confirmed that the test would adequately verify the initial design requirements for the CEQ fan start logic and time delay.

1-DCP-728 Modification to Containment Flood up Overflow Wall Open 300 Square Inch Penetration in Unit 1 Crane Wall for Flood-up Overflow

For certain small break LOCA scenarios, there was insufficient containment recirculation sump water inventory to support continued operation of the emergency core cooling system (ECCS) pumps. This deficiency was caused by the design features of the internal lower containment that allowed water to be diverted from, and not made available to, the containment recirculation sump. This design change corrected the deficiency, by installing five 10 inch diameter penetrations in the containment flood up overflow wall separating the pipe annulus region from the reactor coolant system loop compartment. These holes allow water to freely flow between these areas to ensure that the pipe annulus inventory is available for the ECCS pumps during the recirculation phase following a LOCA. The team confirmed that the licensee had used conservative inputs in the supporting analysis.

1-DCP-4684 Install New Containment Water Level Switches

In this design change, new containment water level switches were installed that augment the existing containment water level indicating system. These new switches improved the accuracy of the containment water level system such that containment water level could be determined within+/-1.25 inches. This was necessary to allow operators to accurately evaluate key sump level values such as, minimum level for switch over to sump recirculation mode, or maximum containment flood up level. The team reviewed associated calculations which established the setpoints for the new containment water level switches and also reviewed the proposed post-modification test procedure for the level switches. The team considered that the modification test procedure methodology would adequately confirm the intended design requirements.

12-DCP-867 Extend Containment Six-Inch Diameter Recirculation Sump Vent Pipe to Elevation 614 Feet

In this design change, the recirculation sump vent line was extended by approximately 12 inches (from elevation 613' to 614') in order to ensure that the vent would remain above the maximum flood up level in containment. The maximum flood level increased due to recalculation of the maximum possible ice melt following a LOCA. The team confirmed that appropriate inputs had been considered in the current calculation which established the maximum containment water level. The team also reviewed the construction drawing for installation of the sump vent line extension and confirmed that it was installed above the maximum containment sump water level.

1-DCP-678 Refueling Water Storage Tank Overflow Modification

In this design change, the storage capacity of the refueling water storage tank (RWST) was increased by modifying the configuration of the overflow and vent piping. The increased storage capacity ensured that an adequate amount of water would be delivered to the containment sump before the operators switched from the injection mode, to the sump recirculation mode, following a LOCA. The team confirmed that appropriate inputs had been considered in a sample of the supporting calculations. However, the team identified discrepancies associated with the completed modification testing as discussed below.

The in-process testing described in section 4.2 of this design change, required filling the RWST to 98.3 percent and verifying that the tank did not overflow prior to receiving the high level alarm. The completed in-process testing documented in a job order C0204675 indicated that the Unit 1 RWST had been filled to only 97.9 percent indicated tank level. Based on conversations with licensee staff, fill of the tank was secured at this level after receiving the high level alarm. However, the confirmatory check to ensure that the tank had not overflowed prior to the high level alarm point was not completed. The licensee initiated condition report (CR) 00313083 to enter this issue into the corrective action program. This discrepancy did not affect the functionality of this modification and therefore, the team considered it a minor issue.

The functional testing discussed in section 4.3 of this design change, required leak testing per the American Society of Mechanical Engineers (ASME) Code. Based on review of completed job order C0204675, the licensee had performed visual inspection of the structural attachment welds to look for leakage. This leakage test was not actually required by the ASME Code and did not serve as a useful check, because the tank level was insufficient to cover all of the welds. The licensee staff subsequently determined that a leak test of the modified overflow piping welds was not required, and appropriately designated the flow path test required by IWC 5222(D), Section XI of the ASME Code. Completion of this test would have served to demonstrate that the RWST overflow line was not blocked. For example, if the purge dams and cleanliness barriers used during installation had been left inside the vent line, then the flow path test would serve to identify this error. However, no documentation existed to indicate that a flow path test was performed as a post modification test. Licensee staff stated that the flow path test requirement was met by closeout cleanliness inspections and the recent completion of 01-OHP 4030.STP.008R “ECCS Check Valve Test,” in which large amounts of water were drawn from the RWST. However, if the overflow vent line had been blocked during this test, licensee staff stated that the RWST would have likely collapsed due to the vacuum drawn in the tank. Fortuitously, this did not occur, and the team agreed that this test met Code requirements for a flow path test. The licensee initiated CR 00319038 to enter this discrepancy into the corrective action program. This discrepancy did not impact the functionality of this modification and therefore, the team considered it a minor issue.

Because of these issues, the licensee initiated CR 00320066 to ensure a more comprehensive review of station modification testing practices would occur.

No findings of significance were identified. Therefore, based on review of the design changes which corrected deficiencies in plant design affecting the containment sump, this issue was adequately resolved and RAM Item 2.1 was closed.

b.2 RAM Item 3.1 - Evaluate Modifications to Resolve Operability of Unit 1 Motor Operated Valves (MOVs)

The team reviewed design operating margins for the 112 Unit 1 MOVs in the Generic Letter (GL) 89-10 program. The team selected eight of these MOVs for a more detailed review. This detailed review included; design change packages, thrust calculations, differential pressure calculations, set-up calculations, seismic/weak link calculations, and Electric Power Research Institute MOV Performance Prediction Methodology engineering evaluations. The team also discussed with licensee personnel, several areas of the MOV program, that remain to be addressed, and which require further NRC review for final closeout of the GL 89-10 program.

The team concluded that Unit 1 MOVs in the GL 89-10 Program were operable, except for the recirculation sump to residual heat removal/containment spray pumps suction valves, 1-ICM-305 and 1-ICM-306. For these valves, the team identified that the licensee had not adequately resolved a pressure-locking concern which could have affected the operability of these valves as discussed below.

1-DCP-4705 Unit 1 MOV Modifications - ECCS

This design change modified the recirculation sump to residual heat removal/containment spray pumps suction valves, 1-ICM-305 and 1-ICM-306 to address a pressure locking concern. The concern involved a post-accident scenario, in which hot recirculation sump water on one side of these parallel disc gate valves would heat water trapped in the valve bonnets locking the valves in a closed position. To correct this vulnerability, this design change provided an equalizing line installed from the bonnet of the valve to a connection on the upstream piping. This line included a relief valve and two isolation valves. The relief valve was intended to prevent a buildup of pressure in the bonnet of the valve by relieving trapped pressure to the upstream piping. The team identified that this design change did not require functional testing of the relief valves. Instead, the licensee relied on the vendor to set and test the relief valves at the design setpoint of 20±1 pounds per square inch gage (psig) as stated in the purchase order. The team’s questions prompted the licensee to identify that these valves had not been preset or tested at 20 psig by the vendor. Through bench testing of a spare relief valve the licensee determined that these valves could be set as high as 290 psig. The, as found set pressure, for the Unit 1 relief valves 1-SV-344E and 1-SV-344W was 160 psig and 240 psig respectively. At the 240 psig lift set pressure, the licensee was unable to confirm that the modification would perform as designed to alleviate sufficient trapped water for the valves to be opened.

The licensee had not operated Unit 1 with this design change installed and for the current plant mode, operability of these valves was not required. Therefore, this finding was determined to be of very low safety significance (GREEN) and within the licensee’s response band. Failure to verify the set points of the relief valves is a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," as measures were not provided for verifying or checking the adequacy of the equalizing line design change. However, because of the very low safety significance of the item and because the licensee has included this item in their corrective action program (CR 00321040), this violation is a Non-Cited Violation (NCV 50-315/00-21-01). This issue was considered more than minor, because if it was left uncorrected, it could have impacted the function of these valves, which affect safe operation of the plant at power. The licensee subsequently removed, reset, and tested the relief valves at the appropriate design setpoint of 20 psig. The team confirmed that these valves had been properly tested and reinstalled.

One finding was identified which potentially impacted operability of two Unit 1 MOVs (1-ICM-305/306). Based on completion of corrective actions for valves 1-ICM-305/306, and data demonstrating operability for each of the Unit 1 MOVs, RAM Item 3.1 was closed.

2-DCP-4371 Pressure Locking Modification for Valves for 2-ICM-305/306 and Operator Changes for 2-ICM-306, and 2-NMO-151/152/153

In this design change, the licensee had modified the recirculation sump to residual heat removal/containment spray pumps suction valves, 2-ICM-305 and 2-ICM-306 to address the same pressure locking concern as discussed for the Unit 1 valves above. The team's questions prompted the licensee to identify that the relief valves for this modification had also not been preset or tested prior to installation, which affected operability of these valves. At 6:00 p.m. on November 17, 2000, the licensee declared the Unit 2 valves inoperable and commenced a Unit 2 shutdown. The licensee determined that by draining at least 2 gallons of water from the bonnet area of each recirculation sump valve, pressure locking could not occur. The licensee subsequently drained 4.5 gallons of water from each valve bonnet to restore operability, and Unit 2 was returned to full power. At the conclusion of this inspection, the licensee was periodically draining water from the valve bonnets to ensure operability of the Unit 2 recirculation sump valves. Operation of Unit 2 during the current operating cycle, with this inadequate design change, is an Unresolved Item (URI 50-316/00-21-02) pending review of the licensee's evaluation for past operability of these valves.

b.3 RAM Item 3.2 - Evaluate Modifications to Address Unit 1 Compressed Air System Operability

1-LDCP-4656 Add Air Flow Device in the Control Air Line to Containment

In this design change, a rotometer type flowmeter was installed in the control air line feeding the containment building to measure air flow. The licensee modified this system to address a concern for not being able to detect a severed air line during a LOCA. A severed air line would add additional pressure to the containment and this modification allowed operators to identify and correct this condition prior to an excessive containment pressure increase. The team reviewed an associated calculation for containment pressure increase and the post-modification test to confirm that the flowmeter would function as designed.

1-DCP-4559 Install Additional Compressed Air Bottles to Increase Back-up Air Volume to Pressurizer Power Operated Relief Valves

In this design change, four additional compressed air bottles were installed to increase the back-up air volume available to cycle the pressurizer power operated relief valves

(PORVs). The compressed air bottles are part of the reactor vessel over pressure mitigation system which prevents a reactor coolant system transient from exceeding design pressure and temperature limits. In the event of a loss of control air, the back up compressed air bottles supply enough air to provide ten minutes of PORV operation. The team confirmed that the calculation supporting the ten minute air demand for PORV operation was adequate.

<u>1-DCP 279 Replace Hoses for Pressurizer PORVs 152 and 153</u>

In this design change, the ½ inch diameter pneumatic air supply hoses for the PORVs were replaced with 3/8 inch diameter hoses. This change was necessary to slow the valve stroke time. During surveillance testing of the Unit 1 PORVs, the licensee had identified that the PORVs closed faster than assumed in the Low Temperature Over Pressure Protection Analysis, which affected the reactor coolant system pressure response during a mass injection or heat-up event. This change corrected the PORV stroke time and ensured that the backup air supply would last 10 minutes for cycling the PORVs as discussed above. The team reviewed post modification tests to confirm that the slower opening and closing stroke times of the PORVs were within required specifications.

<u>1-DCP-548 Unit 1 CD Emergency Diesel Generator Starting Air Compressor/Piping Installation</u>

In this design change, the starting air compressor was replaced with a seismically qualified safety-related compressor. This design change included replacement of associated air system piping and valves. These components were replaced because the original components were non-safety-related/non-seismic and were required to support operation of the safety related emergency diesel generators. This design change was one of four basically identical design changes performed on the four emergency diesel generators, with only minor differences in pipe routing and support installation. The team reviewed associated calculations and confirmed that the new compressors were seismically qualified and that the new air compressors did not result in room heat-up temperatures above the design maximum. The team also reviewed the post-modification tests for the new compressor and the inservice leak test for the newly installed check valves. The team confirmed the tests met the initial design requirements and showed acceptable operation of the compressor and check valve.

<u>1-DCP-185 Replace Emergency Diesel Generator Starting Air Receiver Pressure Switches</u>

In this design change, the pressure control circuits and switches for the emergency diesel starting air receiver were modified. This change was necessary to ensure that the diesel generator air receiver tank pressure could be maintained within its design values. The replacement pressure switches reduced the instrument deadband and eliminated the possibility for the air receiver tank safety valves to actuate before the switches reached the reset pressure of 245 psig. The team confirmed that appropriate inputs had been considered in a sample of supporting calculations, and reviewed the post modification test which demonstrated the integrity of the starting air system logic.

No findings of significance were identified. Therefore, based on the design changes reviewed, the issues affecting the operability of the compressed air systems were adequately resolved and RAM Item 3.2 was closed.

b.4 <u>RAM Item 4.1- Evaluate Modifications to Resolve Operability of the Unit 1 Auxiliary Feedwater (AFW) System, Associated with High Energy Line Breaks (HELB)</u>

<u>1-DCP-4595 Seal the Turbine Driven and Motor Driven Auxiliary Feedwater Pump Rooms</u>

In this design change, the AFW pump rooms were sealed to protect AFW equipment from steam entering from the turbine building or from the steam supply line to the turbine driven AFW pump during a postulated HELB event. Sealing the rooms resulted in the need to add room coolers to each AFW area to maintain acceptable temperatures. These room coolers were also installed under this change. The team confirmed adequate implementation of this design change during a walk down of these rooms. The team also confirmed that an adequate calculation was performed demonstrating acceptable room temperatures in the event the room coolers were lost due to a postulated plant fire.

<u>1-LDCP-4795 Modify Unit 1 Auxiliary Feedwater Room Fire Damper Installation</u>

In this design change, the mounting configurations of four AFW pump room fire dampers were modified to match the manufacturer's required condition of installation. This change ensured that the dampers met requirements for a three-hour fire barrier and structural requirements for seismic Class I components. This change also corrected an oversight in the original design which had not accounted for thermal expansion during a fire. The team confirmed that adequate functional testing of fire dampers was specified in procedure 12-PPP-4030.066.21 and that this modification did not affect the ability to seal the AFW pump rooms in the event of a HELB.

No findings of significance were identified. Therefore, based on the design changes reviewed, the issues affecting the operability of the Unit 1 AFW system associated with HELB were adequately resolved and RAM Item 4.1 was closed.

b.5 <u>RAM Item 5.1- Evaluate Modifications Completed to Resolve the Unit 1 Electrical System Fuse and Breaker Coordination</u>

<u>1-DCP-4690 250 VDC Fuse Replacement Project</u>

In this design change, existing 60 Amp, and below, fuses in the Unit-1 250 VDC system that did not have adequate voltage ratings for the system operating voltage ranges, were replaced. This change ensured that these safety related fuses were capable of isolating overloads or interrupting faults, and provided predictable system operation. The team also reviewed design information transmittal (DIT)-B-01382-00 which recommended the type and characteristics of the replacement fuses. No post modification test was specified for this Unit-1 modification based on equivalency with the identical Unit-2 modification 2DCP-4392, "250 VDC Fuse Replacement Project," which had been functionally tested. The team reviewed Unit 1 condition reports on fuses and confirmed that no 250VDC fuse failures had occurred since installation of this modification.

No findings of significance were identified. Therefore, based on the design change reviewed, the issues affecting the Unit 1 electrical system fuse and breaker coordination were adequately resolved and RAM Item 5.1 was closed.

b.6 RAM Item 6.1- Evaluate Setpoint Changes for the Unit 1 Refueling Water Storage Tank (RWST) Level to Account for Measurement Error and Instrument Uncertainties

12-DCP-0853 Modification To ILS-950 and 951

In this design change, the Unit 1 and 2 sensing line connections for the refueling water storage tank (RWST) level instruments were relocated. This change resolved the original deficiency in the location of the RWST level sensing tap on the tank outlet pipe, which would cause a substantial level error, that could impair the ability of operators to accurately determine the appropriate point to change from injection to recirculation phase following a LOCA. To avoid level errors caused by flow effects in the original instrument tap location, this modification relocated the instrument taps from their original location on the outlet piping to the tank wall. The team reviewed supporting setpoint changes, calculations and the post modification test, which demonstrated the integrity of the modified piping.

No findings of significance were identified. Therefore, based on the design change completed and setpoint changes reviewed, the issues affecting the Unit 1 RWST level/ level setpoints were adequately resolved and RAM Item 6.1 was closed.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (IP 71152)

.1 Corrective Action Process Review

a. Inspection Scope

In conjunction with the baseline inspection, the team reviewed a sample of licensee corrective action documents to verify that when issues within the plant modification and 10 CFR 50.59 processes were identified, they were appropriately characterized and entered into the licensee’s problem identification and resolution program. During this review, the team also assessed whether the corrective actions were appropriate to prevent recurrence.

b. Findings

No findings of significance were identified.

4OA3 Event Follow up

.1 Review of Licensee Event Reports (LERs)

a. Inspection Scope

The team reviewed LERs associated with the areas reviewed under this baseline inspection.

b. Findings

b.1 (Closed) LER 50-316/98005-01: Potential for HELB to degrade component cooling water system.

(Closed) LER 50-316/98007-01: HELB effects on AFW system.

(Closed) LER 50-315/98058-01: Postulated HELB could result in condition outside the design bases for auxiliary feedwater.

In October 1999, the licensee recognized, based on the large number of HELB deficiencies that a programmatic problem existed. The team reviewed the three LERs listed above and noted that supplements had been written to these HELB scenario-specific LERs to close them to LER 50-315/99026-00 which addressed the broader HELB programmatic deficiencies. LER 50-315/99026-00 was previously inspected and found to be acceptable as documented in NRC Inspection Report 50-315/316/2000007. The risk associated with these postulated HELB events was evaluated and documented in NUREG-1728, "Assessment of Risk Significance Associated With Issues Identified at D.C. Cook Nuclear Power Plant."

During this inspection, the team reviewed modifications to eliminate the vulnerability of the Unit 1 AFW system to HELB events. No findings of significance were identified. Therefore, based on these reviews and documented NRC evaluation of the risk significance, these LERs were closed.

b.2 (Closed) Licensee Event Report 50-315/2000-001 00: Stress loads for the Ice Condenser Basket Assembly Greater than Allowed by Safety Analysis Report. The licensee discovered that the original design analysis did not consider the dynamic forces that could occur on the ice basket due to the slotted clevis bracket design and that a minimum ice basket weight was necessary to prevent the baskets from being over-stressed during a design basis accident. The team reviewed the structural calculation and associated 10 CFR 50.59 analysis, as well as the licensee procedures that prescribed a minimum ice basket weight and no findings of significance were identified. This LER is closed.

4OA6 Management Meetings

Exit Meeting Summary

The team presented the inspection results to Mr. Powers, and other members of licensee management at the debriefing meeting held on November 17, 2000, and in a final phone exit meeting held on December 4, 2000. The licensee acknowledged the finding presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Powers, Senior Vice President, Nuclear
M. Rencheck, Vice President Engineering
C. Bakken, Site Vice President
W. Kropp, Director Regulatory Affairs
S. Lacey, Director Engineering
D. Garner, Director Nuclear Fuels and Safety Analysis
S. Greenlee, Director Design Engineering
R. Godley, Director Plant Engineering
T. Noonan, Director Performance Assurance
M. Barfelz, Engineering
N. Jackiw, Regulatory Affairs
J. Rasor, Modification Manager
R. Ebright , Engineering Programs Manager
K. Eslinger, MOV Testing
A. Gort, MOV Coordinator
J. Kinsey, MOV Production Manager
L. Lorati-Thurston, MOV Project Manager
H. Pitts, MOV Engineering Manager
C. Swanner, MOV Engineering

NRC

B. Bartlett
J. Maynen

INSPECTION PROCEDURES (IPs) USED

IP 71111.02	Evaluations of Changes, Tests, or Experiments
IP 71111.17	Permanent Plant Modifications
IP 71152	Identification and Resolution of Problems
IP 71153	Event Follow up

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/00-021-01(DRS)	NCV	Failure to properly set relief valves installed in Unit 1 motor operated valves during a design change
50-316/00-21-02(DRS)	URI	Operation of Unit 2 with an inadequate design change for the motor operated recirculation sump suction isolation valves.

Closed

50-315/00-021-01(DRS)	NCV	Failure to properly set relief valves installed in Unit 1 motor operated valves during a design change
50-316/98005-01	LER	Potential for high energy line break to degrade component cooling water system
50-316/98007-01	LER	High energy line break effects on auxiliary feedwater system
50-315/98058-01	LER	Postulated high energy line break could result in condition outside the design bases for auxiliary feedwater
50-315/2000-001-00	LER	Stress loads for the ice condenser basket assembly greater than allowed by the safety analysis report
Item 2.1	RAM	Evaluate modifications to the Unit 1 recirculation sump to address previous deficiencies with inadequate Inventory, dead ended compartments and sump venting
Item 3.1	RAM	Evaluate modifications completed to resolve operability of the Unit 1 MOVs
Item 3.2	RAM	Evaluate Modifications Completed to Address Unit 1 Compressed Air System Operability
Item 4.1	RAM	Evaluate modifications completed to resolve operability of the AFW system, associated with postulated high energy line breaks
Item 5.1	RAM	Evaluate modifications completed to resolve the Unit 1 electrical system fuse and breaker coordination
Item 6.1	RAM	Evaluate setpoint changes for the Unit 1 RWST Level to account for measurement error and instrument uncertainties
Item 7.1	RAM	Evaluate licensing basis program controls for safety reviews in accordance with 10 CFR 50.59

Discussed

None

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
CR	Condition Report
DIT	Design Information Transmittal
DRS	Division of Reactor Safety
DCP	Design Change Package
ECCS	Emergency Core Cooling System
GL	Generic Letter
HELB	High Energy Line Break
IP	Inspection Procedure
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MOV	Motor Operated Valve
NCV	Non-Cited Violation
PORV	Power Operated Relief Valve
psig	Pounds per Square Inch Gage
RAM	Restart Matrix Item
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
URI	Unresolved Item

LIST OF DOCUMENTS REVIEWED

Calculations

SD-990618-003	Containment Free Volume, Revision 1
3195-129	Volume Calculations, Revision 0
TH-97-16	D. C. Cook Containment Flood-up, Revision 1 and Draft, Revision 2
MD-12-RWST-001-N	Maximum Differential Pressure for RWST Vent Path, Revision 2
1-2-I9-03	Calculation 7 - Refueling Water Storage Tank Minimum Technical Specification Volume, Revision 0
MD-02-DR-001	Verify Capacity of Rerouted CEQ Room Floor Drains, Revision 1
SD-000429-023	Evaluation of the Reinforced Concrete Containment Flood Up Overflow Wall Inside the Unit 1 Containment Building, Revision 1
MD-12-HV-020-N	Heat Gain Calculation and Maximum/ Minimum Temperature Determination for the Emergency Diesel Generator Room 1,2AB and 1,2CD, Revision 1
SD-000501-002	Seismic Qualification of Unit 1 Emergency Diesel Generator, Ingersoll-Rand Model H7100, and Auxiliary Equipment Including Coalescing Filter, Pressure Gauge, and Discharge Safety Relief Valve, Revision 1
SD-000501-003	Evaluation of Unit 1 Emergency Diesel Generator Air Start Compressors for Missile Effects, Revision 0
SD-990826-003	Ice Condenser Basket Design, Revision 0
WCAP-8304	Stress and Structural Analyses and Testing of Ice Baskets (Proprietary), May 13, 1974
WCAP-8887	Ice Basket Stress Analysis - D. C. Cook (Proprietary), March 1977
1-2-I9-03, Calc. 3	RWST Level Setpoints, Xmtr Calibration

NEID-12DCP0853-001	Heat Loss Associated with the New Piping/Tubing Installed Under DCP 0853, Revision 0
SD-000429-011	Qualification of Mounting Details for Flood Level Switches 1-NLI-330, 333, 340 and 341, Revision 0
SD-000620-002	Seismic Qualification of Containment Water Level Switch 1NLI-330, 331, 340, and 341 - GEMS Sensor Model LS-57761 Revision 0
SD-000608-001	Qualification of Conduit and Junction Box Supports associated with DCP No. 1-DCP-4684, Revision0
MD-1-HV-041-N	Unit 1 CEQ Fan (1-HV-CEQ-1/2) Motor Bhp Evaluation, Revision 0
1-2-UNC-339, CALC2	Setpoint Calculation for RWST Level Alarms, RHR Pump Trip Interlock, and Operations Points, Revision 1
1-2-I9-03, CALC7	Refueling Water Storage Tank Minimum Technical Specification Volume, Revision 0
1-E-N-ELCP-250-001	Unit-1 250 VDC System Coordination Study, Revision 0
MD-12-CA-002-N	Pressurization of Containment Due to Release of PORV Compressed Air Bottles During Design Basis LOCA, Revision 2
MD-12-CA-003-S	Control Air Distribution Header Pressure Loss, Revision 0
MD-12-CA-004-S	Determination of Available Pressurizer PORV Strokes Using the Auxiliary Air Supply, Revision 1
MD-12-HV-033-N	TDAFW Pump Room Temperature Under Station Blackout Conditions, Revision 1
TH-00-05	Auxiliary Feedwater Pump Room Heat-Up Temperatures, Revision 0
MD-1-CCW-013-N	Torque and Thrust Setup Calculation for 1-CCM-430 and 1-CCM-432, Revision 2, 10/30/00
MD-12-CTS-131-N	EPRI PPM Evaluation for 1/2-IMO-210, 1/2-IMO-211, 1/2-IMO-220, 1/2-IMO-221, Revision 0, 8/23/00
MD-01-CTS-140-N	EPRI PPM Evaluation for 1-IMO-220 and 1-IMO-221, Revision 1, 11/8/00
MD-12-ESW-073-N	EPRI PPM Evaluation for 1-WMO-713, 1-WMO-717, 2-WMO-714, 2-WMO-718, Revision 1, 8/18/00
MD-01-ESW-081-N	Torque Setup Calculation for, Revision 2, 11/2/00
MD-01-RHR-018-N	Torque and Thrust Setup Calculation for 1-IMO-330 and 1-IMO-331, Revision 2, 11/9/00
MD-01-RHR-019-N	Torque and Thrust Setup Calculation for 1-IMO-340 and 1-IMO-350, Revision 2, 11/10/00
MD-12-RHR- 110-N	EPRI PPM Evaluation for 1/2-IMO-340 and 1/2-IMO-350, Revision 1, 9/15/00
MD-12-RHR- 111-N	EPRI PPM Evaluation for 1-IMO-330 and 1-IMO-331, Revision 1, 10/4/00
MD-12-SI-001-N	EPRI PPM Evaluation for 1/2-IMO-262 and 1/2-IMO-263, Revision 2, 9/21/00
MD-12-SI-004-N	EPRI PPM Evaluation for 1/2-IMO-270 and 1/2-IMO-275, Revision 1, 9/15/00
MD-01-SI-008-N	Torque and Thrust Setup Calculation for 1-ICM-260 and 1-ICM-265, Revision 1, 11/9/00
MD-01-SI-011-N	Torque and Thrust Setup Calculation for 1-IMO-262 and 1-IMO-263, Revision 2, 11/9/00

MD-01-SI-012-N	Torque and Thrust Setup Calculation for 1-IMO-270 and 1-IMO-275, Revision 2, 10/6/00
MD-01-SI-016-N	EPRI PPM Evaluation for 1-ICM-260 and 1-ICM-265, Revision 0, 9/25/00
SD-990825-005	Seismic Weak Link Thrust Calculation for 1/2-ICM-250 & 251, 1/2-ICM-255 & 256 and 1/2-ICM-260 & 265, Revision 3, 9/15/00
SD-990825-006	Seismic Weak Link Thrust Calculation for 1-IMO-210, 1-IMO-211, 1-IMO-220, 1-IMO-221, 2-IMO-210, 2-IMO-211, 2-IMO-220, & 2-IMO-221, Revision 3, 8/23/00
SD-990825-016	Seismic Weak Link Thrust Calculation for 1-IMO-340, 1-IMO-350, 2-IMO-340, 2-IMO-350, Revision 3, 10/23/00
SD-990825-019	Seismic Weak Link Torque Calculation for 1-WMO-713, -717, and 2-WMO-714, -718, Revision 3, 8/16/00
SD-990825-023	Seismic Weak Link Thrust Calculation for 1/2-IMO-330, 1/2-IMO-331, Revision 4, 11/3/00
SD-990825-035	Seismic Weak Link Thrust Calculation MOV(s): 1/2-IMO-270, 1/2-IMO-275, Revision 2, 9/6/00
SD-990825-041	Seismic Weak Link Torque Calculation for 1/2-IMO-262/263/312/322, Revision 3, 9/22/00
SD-990825-042	Seismic Weak Link Torque Calculation for 1/2-CCM-430, 431, 432 and 433, Revision 2, 8/11/00

Design Changes and Field Change Notices

1-DCP-678, Refueling Water Storage Tank Overflow Line Modification, Revision 0a
1-FCN-678-R0-01,02
1-FCN-678-R0A-01
1-DCP-0634 Modify CEQ Fans to Start on a Phase A containment Isolation Signal, Revision 0
1-DCP-436, Remove the Internals of the CEQ Room Floor Drain Check Valves, Revision 3b
1-FCN-436-R0-01,02,03
1-DCP-728, Modification to Containment Flood-up Overflow Wall, Revision 0
1-FCN-728-R0-01,02,03,04,05,06,07,08,09,10,11,12,13
1-DCP-548 Emergency Diesel Generator 1CD Starting Air Compressor Replacement
12-DCP-867 Extend Containment Six-inch Diameter Recirculation Sump Vent Line to Elevation 614', December 11, 1997
12-DCP-853, Modification To ILS-950 and 951,Revision 0
1-DCP-4690, 250 VDC Fuse Replacement Project, Revision 3
12-DCP-185, Emergency Diesel Generator Starting Air Compressor Pressure Switch Replacement, Revision1a
1-DCP-4684, Install New Containment Water Level Switches, Revision 0a
1-DCP 279, Change Size of Air Hose for PORVs 152 and 153, Revision 0
1-DCP 4559, Install Additional Compressed Air Bottles to Increase Back-up Air Volume to Pressurizer Power Operated Relief Valves, Revision 0
1-DCP 4595, Seal the TDAFP rooms, Common Hallway, and MDAFP Rooms. Modification of Auxiliary Feedwater Pump Rooms Ventilation System, Revision 0a
1-LDCP 4656, Add Air Flow Device in the Control Air Line to Containment, Revision 0
1-LDCP 4795, Modify Unit 1 Auxiliary Feedwater Room Fire Damper Installation, Revision 0
2-DCP-181 Pressure Locking Modification for Valves 2-IMO-330/331 and 2-NMO151/152/153, Revision 0, 11/4/99

1-DCP-4705 Unit 1 MOV Modifications-ECCS, Revision 0, 7/31/00
1-DCP-4705 Unit 1 MOV Modifications-ECCS, Revision 0A, 8/30/00
1-LDCP-4812 Add a Spacer to the Operator Spline Adapter to Ensure Operator Engagement, Revision 0, 8/16/00
1-LDCP-4818 Revise Overall Gear Ratio (OAR) on CTS Valves 1-IMO-210, 211, 220, & 221, Revision 0, 9/5/00
2-DCP-4371, Pressure Locking Modification for Valves for 2-ICM-305/306 and Operator Changes for 2-ICM-306, and 2-NMO-151/152/153, Revision 0

Drawings

OP-1-5128-0 Flow Diagram Reactor Coolant Unit No. 1
INT-1-SI-48
INT-1-5353-DEMO Tank Area Piping Arrangement Plan-West Unit No. 1
INT-OP-1-5144 Flow Diagram Containment Spray Unit No. 1
INT-1-5468 Containment Unit 1 Waste Disposal System Reactor Cooling Drain Tank Piping
INT-1-3185 Grating Details for Vent Openings on the Crane Wall
INT-1-3179A Unit 1 - Containment Building Opening Shielding Details For Flood Up Overflow Wall
INT-1-2-3179A Unit 1&2 Containment BLDG Reactor Control Cable Tunnel to El. 612"-0" Flood Up Overflow Structure Pedestal for Pressurizer Relief Tank Reinforcing
INT-1-2-3185
1-DR-467, Detail Isometric of Containment Recirculation Sump Vent Line, Revision 2

10 CFR 50.59 Evaluations

1999-1484-00 2-DCP-4260, Rev 0 Modification to Surge Line Whip Restraints
1999-1208-01 2 DCP-679 Modification to Containment Flood up Overflow Wall
1999-0339 Unit 1 Control Room Gas Tracer Test, June 04, 1999
2000-0216 Change Updated Final Safety Analysis Report Table to Agree with Seal Water Line Resistance Calculation, March 15, 2000
2000-0262 Emergency Core Cooling System Recirculation Leakage Test, February 21, 2000
2000-0265 Ice Condenser Ice Basket Uplift Analysis, May 09, 2000
2000-0457 Updated Final Safety Analysis Report Change, May 07, 2000
2000-0512 Replace Essential Service Water Pump Casing and Impeller and Install Pump Column Seismic Supports, March 16, 2000
2000-0806 Updated Final Safety Analysis Report Changes from pH Calculations, May 03, 2000
2000-1018 Changes to Procedure on Placing Emergency Core Cooling Systems in Standby, May 19, 2000
2000-1069 Increase Design Basis Lake Temperature, July 07, 2000
2000-1217 Loss of All Residual Heat Removal Cooling, July 24, 2000
2000-1372 Allowable Containment Leak Rate, June 15, 2000
2000-1444 Revise Auxiliary Building Ventilation Fan Control, July 14, 2000
2000-1473 Addition of Bypass Check Valves in Response to Generic Letter 96-06, July 20, 2000

2000-1529 Auxiliary Feedwater Suction from Essential Service Water, July 21, 2000
2000-1552 Unit 1 Motor Operated Valve Modifications - Component Cooling, August 16, 2000
2000-1577 Modification of Essential Service Water Strainer Backwash, August 08, 2000
2000-1616 Emergency Diesel Generator Starting Air Compressor Replacement, August 18, 2000
2000-1644 Unit 1 Loss of Coolant Accident/ Loss of Power Anti-pump Reset Modification, August 15, 2000
2000-1650 Replace Emergency Diesel Generator Throttle Closing Cylinders, August 18, 2000
2000-1745 Control Room Normal and Emergency Damper Modification, September 08, 2000
2000-1790 Changes to Operations Procedures to Delete Down Power for Safety Injection Accumulator Filling, September 15, 2000
2000-1771-00 Revise Overall Gear Ratio (OAR) on CTS Valves 1-IMO-210, 211, 220, & 221, 9/5/00
2000-1573-00 Unit 1 MOV Modifications-ECCS, 7/31/00
2000-1573-01 Unit 1 MOV Modifications-ECCS, 8/30/00

10 CFR 50.59 Screenings

2000-1683-00 Add a Spacer to the Operator Spline Adapter to Ensure Operator Engagement, 8/16/00
2000-1377-00 1-DCP-678, Rev 0 Refueling Water Storage Tank Overflow Modification
2000-0025 Ice Condenser Ice Weight Data Analysis, January 08, 2000
2000-0031 Updated Final Safety Analysis Report 5.3.4.9 Design Criteria for Ancillary Equipment, April 28, 2000
2000-0041 Criteria for Conducting Infrequently Performed Tests or Evolutions, June 15, 2000
2000-0064 Updated Final Safety Analysis Report Ice Condenser Rewrite, March 07, 2000
2000-0120 Updated Final Safety Analysis Report Ice Condenser Rewrite, March 29, 2000
2000-0143 Updated Final Safety Analysis Report Section 7.2 Cable Tray Fire Wrap, February 10, 2000
2000-0162 Emergency Diesel Generator Fuel Oil Supply System, February 09, 2000
2000-0166 Operation of the Residual Heat Removal System Procedural Changes, January 26, 2000
2000-0174 Procedures for Both the East & West Residual Heat Removal Train Operability Tests - Shutdown, February 03, 2000
2000-0178 Ice Condenser Basket Weighing Surveillance, January 28, 2000
2000-0212 Updated Final Safety Analysis Report Sections 6.2.2 & 6.2.3, February 04, 2000
2000-0225 Minor Changes to Ice Condenser Description, March 15, 2000
2000-0241 Design Evaluation for Use-as-is Determination, February 03, 2000
2000-0310 Unit 2 Emergency Core Cooling Systems Flow Balance -Boron Injection System Surveillance, February 10, 2000
2000-0343 Revise Updated Final Safety Analysis Report Table 11.3-3, February 21, 2000
2000-0344 Use-as-is Determination for Reactor Vessel Closure Stud Nuts, March 07, 2000
2000-0528 Residual Heat Removal Pump Discharge High Pressure Alarm, March 22, 2000
2000-0555 Inservice Testing Program Described in the Updated Final Safety Analysis Report, March 21, 2000

2000-0584 Unit 1 Containment Heating, Ventilation and Air Conditioning Duct Support Seismic Upgrade, March 21, 2000
2000-0597 Use-as-is Determination for Centrifugal Charging Pump Cladding, March 30, 2000
2000-0605 Refueling Water Storage Tank Level Setpoints, April 06, 2000
2000-0616 Technical Specification Bases Changes for 3/4.6.1.4 &3/4.6.1.5, April 27, 2000
2000-0618 Maximum 15 Percent Flow Blockage Through the Ice Bed Flow Area, March 31, 2000
2000-0783 East/ West Essential Service Water System Test, April 06, 2000
2000-0857 Use-as-is Evaluation of Styrofoam in the Unit 1 Containment, April 25, 2000
2000-0909 Safety Injection and Steam Line Isolation Initiating Signals, April 29, 2000
2000-0951 Incorporation of Questions and Answers, Update of Residual Heat Removal Flow and Revision of Text to Agree with Figure, May 01, 2000
2000-0986 Updated Final Safety Analysis Report 14.1.6 & 14.1.8, May 03, 2000
2000-1007 Filling and Venting the Reactor Coolant System, May 10, 2000
2000-1190 Multiple Rod Drop Analysis 1.00 Computer Program, May 31, 2000
2000-1262 Commitment Change, May 30, 2000
2000-1809 Emergency Diesel Generator 1AB Starting Air Compressor Modification Test Procedures, September 09, 2000
2000-1832 Loss of Offsite Power/ Loss of Coolant Accident Anti-Pump Reset Relay Modification Tests, September 13, 2000
2000-1895 Evaluation and Modification of Unit 1 Residual Heat Removal System Tap Lines and Tieback Supports, September 13, 2000
2000-1954 Unit 1 East Essential Service Water Outlet Strainer Backup Air Test, September 25, 2000
2000-1970 Modifications to the Reactor Coolant Pump Thermal Barrier Component Cooling Water Discharge Piping, September 27, 2000
2000-2082 Weight Addition to Unit 1 Containment Spray Pump Motors, October 09, 2000
1999-0131-01 Emergency Diesel Generator Starting Air Compressor Pressure Switch Replacement
2000-1681-00 Install New Containment Water Level Switches
1999-1265-00 Install New Containment Water Level Switches.
1999-1478-02 250 Volt DC Fuse Replacement Project
2000-1653-00 250 V Fuse Replacement Project
2000-1841-00 Transfer of Cold Leg Recirculation
2000-1587-00 CEQ Fan Start Logic Modification
1999-1152-00 CEQ Fan Start Logic Modification
1999-1602-01 Modification of Auxiliary Feedwater Pump Rooms Ventilation System
2000-1521-00 Addition of Air Flow Indicator in Control Air Line to Containment
2000-1589-00 Install Additional Compressed Air Bottles to Increase Back-up Air Volume to Pressurizer Power Operated Relief Valves (PORV's), 1-DCP-4559
2000-1973-00 Installation of Unit 1 AFP Room Coolers including ESW Orifices

<u>Procedures</u>

12 EHP 5040 MOD.006	Design Change Packages, Revision 4
12 MHP 5021.001.03	Safety Valve Bench Testing, Revision 8
12-EHP-5074.MOV.001	Motor-Operated Valve Program, Revision 1, 11/11/00
12-EHP-5074.MOV.002	Motor-Operated Valve Setpoint Control, Revision 0, 3/29/00

PMI-5074	Motor-Operated Valve Program, Revision 0, 10/6/99
01-OHP 4023.ES-1.3	Transfer to Cold Leg Recirculation, Revision 5
01-OHP 4023.ECA-1.1	Loss of Emergency Coolant Recirculation, Revision 5
01-DCP-548-TP.1	Emergency Diesel Generator 1CD Starting Air Compressor Test 1-QT-142-DC1, September 26, 2000
01-DCP-548-TP.2	Emergency Diesel Generator 1CD Starting Air Compressor Test 1-QT-142-DC2, September 26, 2000
12-EHP-4030.STP.262	Ice Condenser Surveillance and Operability Assessment, Revision 0
PMP-1040-SES.001	Safety Screenings/ Evaluations, Revision 12
TS-C-CS44	Enhanced 10 CFR 50.59, Revision 5
TS-C-CS44A	Day 4 - Enhanced 10 CFR 50.59 Training, Revision 1
TS-C-CS45	10 CFR 50.59 Requalification Training, Revision 0
TS-O-0003	Perform Safety Screenings/ Evaluations (50.59), Revision 6

<u>Miscellaneous Documents</u>

DIT-S-00821-00 Assessment of Operability for 2-ICM-305/306 under Pressure Locking Conditions, 11/17/00
DIT-S-00824-01 Evaluation of Quantity of Water to Drain from the Internal Volume of 2-ICM-305 and 2-ICM-306 to Ensure Sufficient Volume for Remaining Water to Prevent Pressure Locking of the Valves, 11/18/00
DIT-S-02828-00 Evaluation of Bonnet Volume of 2-ICM-305/306, 11/17/00
DIT-B-00834-01 List of MOVs Modeled in the Cook Probabilistic Risk Assessment (PRA), 11/13/00
DIT-B-00011-04 Accident Analysis Input Assumptions for Sump Water Level Analysis
DIT-B-00016-00 Drain Path from Lower Containment Annulus to Active Containment Sump
DIT-B-00296-01 Additional New Containment Integrity Peak Pressure and Temperature Analysis Results
Completed Job Order No. 00307068 dated November 14, 2000
Completed Job Order No. C0053181 dated April 16, 2000
Completed Job Order No. C0204675-03 Production approved July 28, 2000
Completed Job Order No. C0204675 dated September 11, 2000
Completed Job Order No. C00322036, dated November 23, 2000
Completed Job Order No. C00322024, dated November 28, 2000
01-DCP-634-TP.1 CEQ Fan Logic and Time Delay Functional Test, Revision 0
12IHP5030.EMP.010 Emergency Battery Light Units, Revision 4
PMI-5025 Environmental Qualification Program, Revision 6
SA-2000-ENP-026 Unit 1 Motor Operated Valve Program Implementation (GL 89-10), 10/17/00
FO-00-J-089 Review of CEQ Floor Drains Calculation, dated October 20, 2000
FO-00-H-047 Review of 12-EHP-5074.MOV.001, 8/9/00
FO-00-H-049 Polishing/Grinding on MOVs in U1, 8/14/00
FO-00-H-070 Review of AEP Calc MD-01-CVCS-050-N, 8/16/00
FO–00-H-081 Overhaul of MOV 1-IMO-222, 8/18/00
FO-00-H-083 Review of Calculation SD-990825-008, 8/21/00
FO-00-H-099 Refurbishment of 1-IMO-310, 8/23/00
FO-00-H-126 Review of A/R A0202000 regarding 1-IMO-202, 8/30/00

FO-00-I-127	MOV Diagnostic VOTES Testing of Valve 1-FMO-212, 9/29/00
FO-00-J-069	Review CR 00-8804, 10/17/00
FO-00-J-028	Installation of Test Equipment on 1-CCM-454, 10/6/00
FO-00-J-123	Follow up of CR 00243168 Actions, 10/24/00
FO-00-K-005	Follow up of CR 99-06150, 10/31/00
FO-00-K-012	Review of "NRC Margin Matrix-CNP U1 GL 89-10 Summary", 11/2/00
FO-00-K-062	Review of CR 00283001 Actions, 11/1/00

NRC Margin Matrix-CNP U1 GL 89-10 Summary, 11/15/00
EAP 00-365, MISC/MOV Engineering Action Plan, 11/10/00.
EAP 00-556, MOVs Engineering Action Plan, 11/10/00.
Receipt Inspection Report for Relief Valves, 1-DCP-4705, 8/26/00
DCC-NOSS-106-QCN, Analytical Basis for Environmental Qualification of Equipment, Revision 2
Donald C. Cook Nuclear Plant Fire Protection Program Manual, Revision 2
Purchase Order NU04 0000010595
Dedication Plan No. PV-0152, 7/27/00

Condition Reports

00241020, 00244092, 00250025, 00273049, 00273073, 00278056, 00286075, 0029036, P-00-00247, P-00-05746, P-99-29677, P-99-29296, P-99-29039, P-00-01926, P-00-07535, P-00-08008, P-00-08854, P-00-09122, P-00-09347, P-00-09957, P-00-10912, P-00-11103, P-00-11171, P-00-11407, P-00-09197, P-00-01078, P-00-05856, P-00-11171, P-00-09180, P-99-29063, P-00-01785, P-99-16855, P-00-01908, P–00-05185, P-00-07067, P-00-09523, P-00-09586, P-00-09691, P-00-09696, P-00-10300, 00243168, 00245055, 00248008, 00251083, 00250019, 00251083, 00253033, 00255082, 00256107, 00271081, 000286076, 0291001, 00321085, P-00-01406, P-00-02317, P-00-02988, P-00-04445, P-00-05034, P-00-08804, P-00-08893, P-00-08946, P-00-09235, P-00-09540, P-00-10065, P-00-10234, P-00-10454, P-00-11190, P-00-11538, P-99-24925, P-99-25159, P-99-25166, 97-3277

Condition Reports (As a Result of this Inspection)

00313083	Testing performed for Unit 1 RWST overflow line was not performed according to DCP 678 requirements
00315060	Minor non-conservative differences observed between two calculations of net free volumes inside containment
00319038	The pressure test method to satisfy the inservice inspection requirements associated with the RWST over flow line modification was not clearly documented
00320066	Three instances of inadequate post-installation DCP testing/documentation indicates the need to determine the apparent cause, extent of condition, and whether process improvements are warranted
00321040	The test data for both 1/2-SV-344W to prove the safety is set for 20 psi could not be located
00320057	Typo in Calculation SD-990825-019
00321053	Minor, Non-Technical Issues in MOV Design Change Packages
00320063	Administrative Error in MOV Setup Calculation MD-01-SI-008-N, R1

INITIAL DOCUMENT REQUESTS

I. <u>Information Needed for in Office Preparation Week</u>

The following information is needed in the Region III Office by Monday, October 23, 2000, or sooner, to expedite reviews during the onsite inspection week (November 6-17, 2000). The Team will review the information requested below and submit additional selected items from the lists provided to your staff during the week before the onsite inspection. We request that any additional items selected from the lists be available and ready for review on the first day of inspection (November 6, 2000).

a. Permanent Plant Modifications

(1) <u>List</u> (with short description) of Unit 1 permanent plant modifications *Completed - List provided September 25, 2000.*

(2) Provide copies of (with the latest revision/change and identify any pending changes) the following design change packages:

DCP 436 “Remove the internals of the CEQ room floor drain check valves” (NRC reviewer M. Holmberg)
DCP 634 “Modify CEQ fans to start on a phase A containment isolation signal” (NRC reviewer B. Scott)
DCP 728 “Modification to containment flood-up overflow wall open 300 square inch penetration in Unit 1 crane wall for flood-up overflow” (NRC reviewer M. Holmberg)
LDCP 4684 “Containment water level switches for the control indication” (NRC reviewer B. Scott)

DCP 548 “U1 CD EDG starting air compressor/piping installation” (NRC reviewer P. Lougheed)
LDCP 4656 “Add air flow devise in the control air line to containment” (NRC reviewer D. Schrum)
DCP 4559 “Install additional compressed air bottles to increase back-up air volume to pressurizer power operated relief valves“ (NRC reviewer D. Schrum)

DCP 4595 “Seal the TDAFP rooms, common hallway and MDAFP rooms” (NRC reviewer D. Schrum)
DCP 4795 “Modify Unit 1 auxiliary feedwater room fire damper installation” (NRC reviewer D. Schrum)

DCP 4690 “Unit 1 250 VDC fuse replacement” (NRC reviewer B. Scott)

DCP 678 “Refueling water storage tank overflow line modification” (NRC reviewer M. Holmberg)

For each design change identified above, the documentation is to include (as applicable) a copy of the;
(a) design change description and supporting calculations;
(b) 10 CFR safety evaluation or safety evaluation screening;
(c) set point change documentation and supporting calculations;
(d) operating and emergency procedure revisions/changes resulting from the design changes;
(e) equivalency evaluations or commercial grade dedication for materials used in the design changes;

(f) ASME Code repair records (e.g. NIS-2 forms, nondestructive examination records, suitability evaluations, pressure testing...);
(g) drawings affected by applicable design changes (1/2 size) (e.g. control room, construction, inservice inspection..)
(h) affected system and component original and current design specifications;
(i) list of applicable vendor manuals;
(j) post modification test (copy of the completed test (if available) or copy of the proposed test) and the requirement/analysis/basis supporting the acceptance criteria in the post modification test. Also, provide a detailed schedule of this post modification testing (as applicable) which will occur between November 6-16th, 2000.
(k) all non-conformance reports associated with the installed modification.

(3) List (with short description) of the analyses/calculations that resolve Unit 1 electrical system fuse and breaker coordination (RAM item 5.1). Additionally, provide a list of work orders/job orders (with short description) that installed new electrical fuses or protective devices as a result of these analysis. Provide a copy of the engineering corrective action plans for the fuse breaker coordination program area. (NRC reviewer B. Scott).

(4) List (with short description) of setpoint changes for the Unit 1 RWST level instruments with a cross reference list (with short description) of supporting calculations (RAM item 6.1). Additionally, provide a list of operating and emergency procedure changes affected by these setpoint changes. Provide a copy of the engineering corrective action plans for the instrument uncertainty program area. (NRC reviewer B. Scott).

(5) List (with short description) of condition reports (open and closed) issued to address plant permanent modification issues/concerns. [1] Additionally, provide copies of investigations and corrective actions taken for CRs 00-9197 & 00-9180 associated with nuclear instrument setpoint errors and DIT-B-1355 associated with flow switch setpoint errors.

(6) Copy of modification procedure(s) and post modification testing procedure.

(7) Motor operated valve (MOV) matrix for both rising stem and quarter turn. (Attached is a suggested matrix with the information needed. The last version of the matrix given to the NRC Team during the Unit 2 restart MOV inspection was in a similar format. The information in this matrix should be as complete as possible knowing that some valves may not be complete at the time of the inspection. (NRC reviewer A. Dunlop)

b. Changes, Tests, or Experiments

(1) List (with short description) of all 10 CFR 50.59 completed evaluations involving: (a) changes to facility (modifications); (b) procedure revisions; (c) tests or non-routine operating configurations; (d) changes to the USFAR; (e) calculation. [1]

(2) List (with short description) of all 10 CFR 50.59 screenings that have been screened out as not requiring a full evaluation involving: (a) changes to facility (modifications); (b) procedure revisions; (c) tests or non-routine operating configurations; (d) changes to the USFAR; (e) calculations. [1]

(3) List (with short description) of condition reports generated because of problems associated with 10 CFR 50.59 evaluations. [1]

(4) Copies of procedures that specify how 10 CFR 50.59 evaluations and screenings are performed.

(5) Copies of procedures describing the 10 CFR 50.59 program including; 10 CFR 50.59 FSAR updates, safety evaluation training and qualification requirements (include copies of required training outlines/agendas).

(6) List (with short description) of special tests or experiments and non-routine operating configurations in the last two years (if any).

(7) Copies of restart closure packages and action plans that resolved/ address deficiencies associated with the 10 CFR 50.59 program.

C. General Information

Site phone book and personnel who will serve as points of contact for the Team.
Current Engineering Organization Chart

[1] Provide information requested going back to January of 1998.

II. Information Request to be Available on First Day of Inspection (November 6, 2000)

a. We request that the following information be available to the Team once they arrive onsite. Copies of these documents do not need to be solely available to the Team as long as the inspectors have ready access to them.

- Updated Final Safety Analysis Report
- Technical Specifications (TSS)
- Latest IPE/PRA report
- Vendor manuals
- Equipment qualification binders
- The latest 10 CFR 50.59 FSAR update submittal

b. Copies of the following documents:

1. Copies of additional permanent plant modifications or set point changes with supporting documentation as requested by the Team prior to the on-site inspection.
2. Provide copies of Q. A. audits, self-assessments and outside organization audits conducted in the areas of 50.59 evaluations and screenings. Also include corrective action documentation/status of identified findings (last 2 years).
3. Copies of any self-assessments and associated condition reports generated in preparation for the inspection.

c. Motor operated valve (MOV) related information (RAM item 3.1).

1. Any recent PA or other self assessment of the MOV program completed since the last NRC MOV inspection for Unit 2 restart.
2. Information and documentation for resolution of issues from Unit 2 restart inspection report.
3. MOV Program Document.
4. Plant procedures for MOV sizing, setting, testing, evaluating test data, preventative maintenance, and trending.
5. List of condition reports on MOVs initiated since the Unit 2 MOV restart inspection.
6. MOV risk ranking.

III. Information Request to be Provide during the Inspection (November 6-17, 2000)

- Copies of any condition report generated as a result of the Team's findings during this inspection.

- Copies of the list of questions submitted by the Team members and the status/resolution of the information requested (provide daily during the inspection to each Team member).

NOTE: If you have any questions regarding the requested information please contact Mel Holmberg at the Region III NRC Office, Phone (630) 829-9748.

SUGGESTED INFORMATION REQUEST MATRIX FOR RISING STEM MOVS

Valve Number	Valve Size/ Type	ANSI Press. Class	Valve Vendor	Safety Funct.	MEDP Close/ Open	Test D/P Close/ Open	% of MEDP Close/ Open	Measured Valve Factor	Load Sensitive Behavior (%)	Stem Frict. Coeff.	Open Margin (%)	Close Margin (%)	Available Valve Factor (Close)	Available Valve Factor (Open)	Applied Valve Factor	Basis for Closure

Basis for Closure:

1.) Dynamic test performed at design-basis differential pressure/flow conditions.
2.) Dynamic test performed at less than design-basis differential pressure/flow conditions; results are linearly extrapolated to design-basis conditions.
3.) Non-dynamically tested; grouped with other similar valves that have been dynamically tested in the plant.
4.) Non-dynamically tested; grouped with other similar valves that have been tested by the industry.
5.) Prototype testing.
6.) Large calculated margin.
7.) Other.

Notes:

1.) Each member of a valve group (or family) should be listed together on consecutive rows so that the group's basis for closure can be easily reviewed.
2.) Put "N/A" in applicable columns if valve was not dynamically tested.
3.) The Load Sensitive Behavior column should contain the measured load sensitive behavior (if dynamically tested) or the margin that was applied to the design-basis thrust calculation, which ever is larger.
4.) The Stem Friction Coefficient column should contain the measured value from the dynamic test, if applicable. Preferred data point: Close-flow isolation; Open-flow initiation.
5.) Open Margin equals the difference between the actuator's capability (at degraded voltage, etc.) and the minimum required open thrust (including any needed margins [such as SFC] and uncertainties [such as consideration of unseating loads]).
6.) Close Margin equals the difference between the thrust available at the current torque switch setting and the minimum required close thrust (including any needed margins and uncertainties). If valve is limit seated, calculate margin in a similar manner as for the open direction.

$$\text{non-d/p tested} \quad \frac{CST_{Bias\ ROL} * (1-\sqrt{\text{Diagnostic uncertainty}^2 + TSR^2 + ROL_{Ran}^2}) - MRT_{Hi}}{MRT_{Hi}}$$

$$\text{d/p tested} \quad \frac{CST_{d/p} * (1 - \sqrt{\text{Diagnostic uncertainty}^2 + TSR^2}) - MRT_{NOM}}{MRT_{Nom}}$$

7.) Available Valve Factor (Close) = (Thrust$_{CST}$ * [1 - (LSB + Uncertainties)] - Packing Loads - Stem Rejection) / (Disc Area * Design-Basis $\triangle$P)
8.) Available Valve Factor (Open) = (Thrust$_{Avail}$ * [1 - (LSB + Uncertainties)] - Packing Loads + Stem Rejection) / (Disc Area * Design-Basis $\triangle$P); LSB may be accounted for differently: e.g., through application of an open stem friction coefficient assumption that has been justified to account for the potential change in stem friction coefficient that occurs under dynamic conditions.
9.) Applied Valve Factor refers to the valve factor that is used by the design-basis thrust calculations.

Modification	Information needed to complete review	Requesting Inspector
DCP-867 Extend top elevation of recirc vent sump pipe	Provide calculations supporting the maximum containment sump level and all other supporting calculations upon arrival at site.	Lougheed
DCP 548 U1 CD EDG starting air compressor/piping installation	Provide all supporting calculations and copy of the completed post modification test upon arrival at site.	Lougheed
DCP 436 Remove the internals of the CEQ room floor drain check valves	Provide the following documents as soon as possible: a current copy of the containment bypass log (12 EHP 6040 PER.154), AR 00286075, FO - 00-J-089, DIT-B-00296, calculation MD-01-DR-001-N, Copy of JOA documenting the in process flow testing for 1-DCP-436.	Holmberg
DCP 728 Modification to containment flood-up overflow wall open 300 square inch penetration in Unit 1 crane wall for flood-up overflow	Provide the following documents as soon as possible: calc SD-000429-023, DIT B-00011-04 and the current revision of drawing OP-1-5128-20.	Holmberg
DCP 678 Refueling water storage tank overflow line modification	Provide the following documents as soon as possible: copy of the current revision of 01-OHP-4023-ES-1.3 and 01-OHP-4023-ECA-1.1 also identify the changes to these procedures which occurred as a result of DCP 678, calculations MD-12-RWST-D01-N, TH-97-16 and 1-2-I9-03 calc 7, 1-E-S-DCP-CBL-00, documentation for in process flow test of section 4.2, functional leak test of section 4.3 and safety evaluation/screening for 2-DCP-729.	Holmberg
DCP 279 Replace hoses for pressurizer PORVs	Provide a copy of the completed post modification test and supporting calculations as soon as possible.	Schrum
LDCP 4656 Add air flow devise in the control air line to containment	Provide a copy of all supporting calculations as soon as possible.	Schrum
DCP 4595 Seal the TDAFP rooms, common hallway and MDAFP rooms	Provide a copy of all supporting calculations as soon as possible.	Schrum
DCP 4559 Install additional compressed air bottles to increase back-up air volume to pressurizer power operated relief valves	Provide a copy of all supporting calculations as soon as possible.	Schrum
DCP 4795 Modify Unit 1 auxiliary feedwater room fire damper installation	Provide a copy of all supporting calculations as soon as possible.	Schrum
LDCP 4684 Containment water level switches for the control indication	Provide a copy of the completed post modification test and all supporting calculations as soon as possible.	Scott
DCP 634 Modify CEQ fans to start on a phase A containment isolation signal	Provide a copy of all supporting calculation(s) upon arrival on site.	Scott

Modification	Information needed to complete review	Requesting Inspector
DCP 4690 Unit 1 250 VDC fuse replacement"	Provide a copy of all supporting calculation(s) upon arrival on site.	Scott
DCP 185 Replace EDG Starting air receiver pressure switches	Provide a copy of Open Item 6369	Scott
DCP 853 Relocate sensing line for RWST level transmitters	Provide a copy of the safety screening / evaluation	Scott

Requesting Inspector	Safety Evaluations	Safety Screenings		
Lougheed	2000-0457	2000-0025	2000-0584	2000-1895
	2000-0512	2000-0031	2000-0597	2000-1954
	2000-0806	2000-0041	2000-0605	2000-1970
	2000-1018	2000-0064	2000-0616	2000-1978
	2000-1069	2000-0120	2000-0618	2000-2082
	2000-1217	2000-0143	2000-0783	
	2000-1372	2000-0162	2000-0857	
	2000-1377	2000-0166	2000-0909	
	2000-1444	2000-0174	2000-0951	
	2000-1473	2000-0178	2000-0986	
	2000-1529	2000-0212	2000-1007	
	2000-1552	2000-0216	2000-1014	
	2000-1577	2000-0225	2000-1147	
	2000-1589	2000-0241	2000-1190	
	2000-1644	2000-0310	2000-1262	
	2000-1650	2000-0343	2000-1444	

Requesting Inspector	Safety Evaluations	Safety Screenings		
	2000-1681	2000-0344	2000-1842	
	2000-1745	2000-0528	2000-1809	
	2000-1790	2000-0555	2000-1832	
	1999-0339			
	2000-0262			

Provide a list of 2000 Safety Evaluations or Screenings not on Previous Lists (provide requesting title, status, and reason for cancellation, if canceled)																	
0001	0002	0003	0006	0007	0008	0009	0016	0017	0027	0028	0034	0035	0047	0050	0051	0061	0065
0069	0072	0074	0075	0076	0080	0084	0085	0086	0088	0100	0103	0104	0105	0110	0111	0115	0119
0126	0132	0137	0144	0150	0154	0158	0159	0160	0164	0169	0170	0172	0173	0179	0184	0185	0186
0187	0188	0192	0193	0195	0196	0198	0203	0209	0211	0217	0222	0223	0224	0226	0228	0232	0233
0234	0235	0240	0242	0245	0248	0249	0250	0254	0255	0256	0262	0265	0269	0274	0276	0286	0287
0288	0289	0291	0292	0294	0297	0301	0306	0311	0312	0313	0314	0317	0322	0327	0330	0333	0334
0335	0336	0337	0339	0340	0342	0345	0346	0349	0350	0352	0353	0356	0358	0360	0362	0363	0365
0367	0368	0369	0371	0372	0374	0376	0378	0379	0380	0386	0387	0388	0389	0390	0391	0392	0401
0407	0410	0411	0412	0419	0422	0426	0429	0433	0440	0442	0447	0448	0449	0452	0453	0454	0460
0461	0463	0466	0468	0469	0472	0473	0474	0475	0476	0477	0482	0483	0484	0485	0489	0491	0492
0497	0499	0500	0505	0507	0508	0509	0515	0519	0520	0521	0522	0523	0525	0527	0531	0536	0540
0543	0544	0545	0546	0547	0548	0549	0550	0552	0554	0563	0564	0566	0567	0569	0571	0572	0574
0579	0588	0589	0591	0592	0595	0596	0599	0600	0607	0608	0612	0617	0622	0625	0626	0629	0630
0639	0642	0644	0645	0646	0647	0648	0649	0655	0656	0667	0675	0685	0687	0689	0690	0691	0693
0700	0701	0702	0704	0711	0715	0717	0718	0721	0724	0725	0726	0729	0730	0732	0735	0736	0738
0746	0749	0756	0757	0763	0764	0768	0769	0770	0778	0780	0782	0786	0794	0795	0796	0800	0801
0805	0809	0811	0816	0819	0820	0822	0823	0828	0847	0850	0853	0859	0861	0862	0871	0872	0882
0883	0884	0894	0898	0900	0902	0906	0912	0913	0916	0918	0919	0926	0929	0930	0940	0945	0949

Provide a list of 2000 Safety Evaluations or Screenings not on Previous Lists (provide requesting title, status, and reason for cancellation, if canceled)																	
0963	0964	0969	0971	0972	0975	0977	0982	0985	0992	0994	0999	1000	1002	1003	1004	1011	1019
1030	1031	1032	1033	1037	1039	1047	1048	1049	1052	1056	1063	1064	1067	1070	1072	1096	1097
1099	1102	1106	1109	1115	1117	1130	1132	1152	1160	1164	1167	1171	1172	1180	1181	1183	1191
1192	1208	1209	1214	1216	1218	1221	1222	1226	1234	1237	1238	1241	1245	1263	1264	1273	1282
1283	1284	1285	1289	1292	1298	1299	1302	1303	1311	1318	1320	1322	1327	1331	1341	1344	1347
1348	1350	1355	1357	1362	1365	1367	1368	1373	1374	1376	1380	1384	1388	1389	1393	1394	1395
1398	1401	1402	1404	1406	1408	1410	1418	1436	1437	1438	1449	1453	1457	1458	1459	1460	1463
1464	1466	1472	1475	1477	1480	1483	1484	1489	1494	1496	1498	1499	1503	1504	1513	1522	1527
1535	1537	1539	1553	1557	1561	1564	1565	1569	1570	1572	1573	1575	1578	1590	1595	1599	1606
1613	1615	1629	1634	1635	1641	1644	1645	1646	1650	1651	1658	1660	1664	1665	1673	1674	1676
1680	1686	1687	1697	1699	1702	1713	1715	1716	1718	1719	1722	1723	1733	1736	1738	1741	1742
1746	1747	1751	1753	1755	1756	1759	1761	1763	1766	1768	1769	1772	1776	1779	1782	1788	1791
1793	1797	1798	1799	1807	1817	1822	1825	1827	1831	1833	1844	1848	1863	1868	1892	1896	1898
1906	1908	1918	1920	1923	1928	1938	1945	1946	1947	1949	1958	1962	1972	1973	1987	1996	1997
2011	2013	2014	2020	2021	2023	2025	2026	2028	2034	2036	2040	2043	2049	2055	2056	2057	2063
2065	2068	2074	2075	2076	2077	2078	2081	2083	2086	2087	2090	2091	2096	2097	2098	2099	2100
2101	2108	2109	2117	2120	2121	2128	2130	2131	2132	2133	2134	2136	2137	2139	2142	2143	2144

Selected CRs, ARs on Mod Process	Selected ARs, CRs on 50.59 SEs or screenings	Requesting Inspector
99-29677	99-29296 , 00-08854,	Holmberg
00-247	AR 00286075	Holmberg
99-29039	00-07067, 00-05746	Holmberg
00-01785	99-29063	Scott

Selected CRs, ARs on Mod Process	Selected ARs, CRs on 50.59 SEs or screenings	Requesting Inspector
00-11171	00-01926	Scott
00-03361	00-05856	Scott
A/R 00278056	00-01078	Scott
P-00-06731	P-00-07535, P-00-09122, P-00-09347, P-00-11103	Lougheed
P-00-11171	P-00-08008, P-00-09957, P-00-10912, P-00-11407, 00241020, 00244092, 00250025, 00273073, 00273049	Lougheed
P-00-01908	P-00-02732	Schrum
P-00-04888	P-00-03731	Schrum
P-00-05185	P-00-04081	Schrum
P-00-05525	P-00-07643	Schrum
P-00-05233	P-99-24709	Schrum