ENCLOSURE 2 TO AEP:NRC:3071

DONALD C. COOK NUCLEAR PLANT

REPORT OF CHANGES, TESTS AND EXPERIMENTS

PURSUANT TO 10 CFR 50.59(d)(2)

I. INTRODUCTION

In accordance with the requirements of 10 CFR 50.59(d)(2), the following report contains a brief description of changes made to the Donald C. Cook Nuclear Power Plant (CNP), and summary of the associated 50.59 Evaluations for the period January 1, 2001, through December 7, 2002.

II. STANDARD ACRONYMS

The following acronyms are used in this report:

COLR	Core Operating Limits Report	RCS	Reactor Coolant System
CTS	Containment Spray System	RHR	Residual Heat Removal
DBA	Design Basis Accident	SGBD	Steam Generator Blowdown
DCP	Design Change Package	SSC	System, Structure, or Component
DNBR	Departure from Nucleate Boiling Ratio	TMD	Transient Mass Distribution
EDG	Emergency Diesel Generator	UCR	UFSAR Change Request
ESW	Essential Service Water	UFSAR	Updated Final Safety Analysis Report
HELB	High Energy Line Break	VAC	Volts Alternating Current
IST	In-Service Test	VDC	Volts Direct Current
LOCA	Loss-of-Coolant Accident		

III. CHANGES, TESTS, AND EXPERIMENTS

The following report contains brief descriptions of physical changes made to the facility implemented under provisions of 10 CFR 50.59(d)(2) and summaries of the associated safety evaluations. This report is organized as follows:

- Facility Changes

- Procedure Changes

- Other Changes

FACILITY CHANGES

1. Change Document Number

1-DCP-4810, Revision 0, and 2-DCP-4785, Revision 0

Title

Install SGBD Flowmeters

Description of Change

This change installed ultrasonic clamp-on type flowmeters on the SGBD system piping outside of containment. The flow meters provide an input to the plant process computer to be used in the calorimetric calculation of reactor thermal power. Local SGBD flow indication was also provided as part of the change. The changes were made to improve calorimetric calculations by providing an actual SGBD flow input allowing these calculations to be performed without securing the SGBD system.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The systems affected by the change continue to perform their design functions as before. The use of externally mounted instruments does not create flow blockages or breach pressure boundaries, therefore, no new HELB locations are created and the installation does not over stress the piping or supports. The instruments are seismically installed to prevent becoming missiles during seismic events. The change does not change the methodology for calorimetric calculations but provides a flow input for accuracy. Technical Specification and License conditions are not impacted by this change. Reactor thermal power will continue to be calculated and compared to actual power using existing procedures. Based on the evaluation, it was determined that the change does not require prior NRC approval.

2. Change Document Number

1-DCP-5173, Revision 0, and 2-DCP-5174, Revision 0, and Related Changes

Title

Provide ESW Minimum Flow Path via CTS Heat Exchangers, including corresponding changes to Technical Specification Bases 3/4.1.2 and 3/4.5.5, and UFSAR Chapters 6, 9, and 14

Description of Change

This change included a combination of design changes, Technical Specification Bases changes, and UFSAR changes. Taken together with applicable procedure changes, these changes supported and enabled the circulation of ESW through the CTS heat exchanger during normal operation, and added this mode to the licensing basis. Prior to the change, the licensing basis did not provide for ESW flow through the CTS heat exchangers except during the containment sump recirculation phase following a postulated accident. In that licensing basis, the CTS heat exchanger path in the ESW system was isolated during normal operation and during the injection phase following a postulated accident.

The design changes modified the control connections to the outlet valves on the ESW side of the CTS heat exchanger to permit adjusting the flow through CTS heat exchangers to achieve an overall ESW system target flow. Prior to the changes, these valves operated only fully open or fully closed. Additionally, portions of the ESW piping downstream of the CTS heat exchangers were insulated to prevent condensation from adversely affecting electrical equipment.

The changes to Technical Specification Bases 3/4.1.2 and 3/4.5.5 incorporated results from a calculation that determined the pH of the containment recirculation sump contents and the CTS during a postulated accident. This calculation revision was required because of a change in the melt rate for the ice condenser in the event of a postulated accident if ESW flow was being provided to the CTS heat exchanger at the time of the postulated accident.

UCR-1609 revised UFSAR Sections 6.3.1, 6.3.2, 9.8.3.2, and 14.4.11.2.2 (Unit 2); Tables 14.3.1-2 and 14.3.1-3 (Unit 1) and Tables 14.3.1-2, 14.3.1-11, and 14.3.1-12 (Unit 2) to correctly reflect the changes identified above for this change.

The purpose of circulating ESW through the normally isolated CTS heat exchangers during normal operation is to enable ESW system flow to achieve the 2000 gpm minimum flow prescribed for ESW pumps. During cold lake water conditions, ESW flow might not reach 2000 gpm if this additional flow path was not available. Prior to the change, the added flow path made available for this purpose involved the cooling loops to the EDG. However, the flow path to the diesel generators was determined to represent common mode failure risk in the event of a failure in the ESW strainer.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. Potential impacts from this change were evaluated and it was determined that the capacity of any SSC to perform its design function was not adversely impacted. Specific evaluations were performed for the impact of the change on the accident analyses for large break LOCA, radiological consequences of a LOCA, hydrogen concentration in containment sub-compartments, containment recirculation sump level, and containment recirculation sump pH. Specific evaluations were also performed for impact of the change on the containment spray heat exchanger, the ESW outlet valves from the containment spray heat exchanger, CTS piping, and environmentally qualified equipment, coatings, sealants and seals inside containment. The pH range continues to remain within the bounds where iodine evolution is prevented, and chloride or caustic stress corrosion cracking are not of concern. Based on these evaluations, it was determined that the activity may be implemented without prior NRC approval.

3. Change Document Number

12-DCP-4252, Revision 0

Title

Provide Ammeter Indication on the 4 kilovolt (kV) Feeds of Transformer TR12-EP-1

Description of Change

This change reconfigured ammeters in each control room that monitored 69kV current supplying the emergency power transformer to indicate the individual current on the 4kV cable feeds to each unit. The design change was made to provide operators the means to monitor the ampacity limits of the 4kV feeder cables. The change replaced existing GE Type AB-40 ammeters, three per control room, with Yokogawa Type AB-40 ammeters, seismically mounted in the existing locations. The existing ammeter inputs to the ampere demand meter monitoring the total system load were reconfigured to reflect the revised installation without changing the meter function. Cable changes, as needed, and ammeter scale changes were also part of the change.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The modification does not change the function, performance requirements or interfaces of any plant system other than the interface between the ammeters and transformer TR12-EP-1. Re-wiring the ammeter inputs from indicating the current supply on each ammeter now monitoring the 69KV side of TR12-EP-1, to indicating the individual current supply on the 4 KV feeds to each unit does not impact the performance of the transformer. This design change also provides indication of the current being supplied to each unit.

The modification does not impact either the function or performance transformer TR12-EP-1. Modification to assure that a component meets its design basis requirements without introducing any new functions, performance requirements or system interfaces, other than the interface between the ammeters and the transformer for measurement purposes, does not require prior NRC approval.

4. Change Document Number

12-DCP-542, Revision 0

Title

Modification to Plant Security Gates

Description of Change

This change modified the latch mechanism of three security gates by welding additional steel members of suitable sizes to the existing steel. The affected gates were 1-GT-AUX343, 2-GT-SCN-211A, and 2-GT-AUX900, located at the entrances of the Unit 1 4kV switchgear room and the Unit 2 ESW pump room, and on the stairway to the spent fuel pool area, respectively. Also, for gate 1-GT-AUX343, some minor adjustments were made in the doorpost to allow the gate to secure properly. For the above gates, excessive gaps of approximately 1/2 inch existed between the protective cover and the face of the doorposts when closed, and the latch of the gate 1-GT-AUX343 did not secure properly. These conditions required compensatory actions, security officer posting, to ensure that the areas remained secured.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The modification adds additional angle iron to three plant security doors to prevent possible tampering with the latch mechanism. The gates are operated by security badge readers to control access to certain vital areas, and do not inhibit required routine or emergency access. The gates are not barriers to radioactive release. The change does not impact the Modified Amended Security Plan. Based on the evaluation, no prior NRC approval was required.

5. Change Document Number

12-TM-01-03, Revision 0

Title

System Upgrades Involving Blocking Six Auxiliary Building Doors and Blocking the Hatches to the Condensate Storage Tank/Refueling Water Storage Tank Pipe Tunnels

Description of Change

This change installed a temporary modification in both units to block six auxiliary building doors and four hatches to the condensate storage tank/refueling water storage tank pipe tunnels. The change was developed to assure that plant security personnel would meet specific timelines to prescribed locations in order to stop a Design Basis Threat. A 10 CFR 50.59 evaluation was performed because the temporary modification conflicted with the plant configuration shown in UFSAR Figure 1.3-8 during the period of its installation.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The evaluation determined that the change did not affect the ability of any system to perform its design function to mitigate the consequences of a DBA. The change did not result in an increase in probability or consequences of previously analyzed accidents or equipment malfunctions and did not create the possibility of any new accidents or equipment malfunctions. The change did not introduce any fire or missile hazard, cause any adverse seismic interaction with any safety-related SSC or SSC important to safety, or interfere with operator routine duties or response to plant transients or accidents. The change adhered to the requirements of the Plant Fire Protection Program. None of the doors or hatches were a fire rated assembly or relied upon to prevent the spread of fire. No new credible failure modes were created and existing failure modes were not adversely impacted by the change. Based on the evaluation, it was determined that prior NRC approval was not required.

6. Change Document Number

1-DCP-5075, Revision 0 and 2-DCP-4891, Revision 0

Title

Unit 1 Cycle 18 and Unit 2 Cycle 13 Reload Safety Evaluation

Description of Change

This change consisted of a combination of changes to the reactor core design and the COLR with corresponding changes to UFSAR Chapters 3, 9, and 14 to support the Unit 1, Cycle 18 and Unit 2 Cycle 13 Reload Safety Evaluations. The change involved the use of annular blankets, thimble plug removal, a Doppler-only power defect lower than previously analyzed (Unit 2 only), and the implementation of ZIRLO cladding including replacement of methodologies used for analyzing the use of ZIRLO (Unit 2 only).

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The reactor operational power, power distribution, reactivity, and temperature response will be maintained within acceptable limits. The evaluation concluded that there are no consequences impacted by the increased nuclide inventories. All fuel rod design criteria are met with the removal of the thimble plugs and resulting increase in bypass flow. In the transient analysis, a combination of the existing DNBR margin and 1% power margin support the thimble plug removal while the LOCA and non-LOCA analyses are either not affected or bounded by the conservative modeling of previous analyses. No accident is initiated due to the lower Doppler-only power defect since all applicable acceptance criteria continue to be met. The adoption of the PAD 4.0 fuel thermal safety model and the LOCA model in analyzing the use of ZIRLO did not constitute a departure from a method of evaluation described in the UFSAR because the NRC approved the use of these models for this application.

7. Change Document Number

Temporary Modification 12-TM-01-52, Revision 0, and Compensatory Actions for Condition Report 01242013

Title

Disable Automatic Opening Feature of ESW Supply Header to EDG Heat Exchangers Shutoff Valves (1-WMO-723, 1-WMO-727, 2-WMO-724, and 2-WMO-728) During the EDG Start Sequence/Run

Description of Change

This change disabled the automatic opening signal on the alternate ESW cooling supply valves to the EDG coolers during EDG starting as compensatory actions for Condition Report 012342013. The change was implemented to preclude a common-cause failure mode of all four EDGs due to the cross-connection of the ESW trains through the EDG coolers and the failure of a single ESW strainer during phenomenon that could promote a silt-intrusion event. These changes will ensure ESW train separation with respect to the EDG coolers.

The change also included procedure changes that allowed for the opening of the alternate EDG cooling supply valves under certain conditions that did not conflict with the design and licensing basis. The affected procedures were Operation of the ESW System, ESW System Loss/Rupture, AB Diesel Generator Operability Test (Train B), CD Diesel Generator Operability Test (Train A), East ESW System Test, West ESW System Test, Reactor Trip or Safety Injection, and Loss of all AC Power.

10 CFR 50.59 Evaluation Summary:

The evaluation of the change determined that a license amendment was not required. The evaluation determined that there was no impact on any accident initiators and that there were no new accident initiators or accidents created by the change. Separation of the ESW trains facilitated by these changes continue to ensure that analyzed ESW flows and single failure criteria assumed in the accident analyses are met. The compensatory actions do not substitute operator actions for automatic actions. There are no design or licensing basis requirements to maintain these valves open or closed. Based on the evaluation, prior NRC approval was not required for implementation.

PROCEDURE CHANGES

8. Change Document Number

02-OHP-4021-052-001, Revision 5 and UCR-1597, Revision 0

Title

Steam Dump Control System Operation and Change to UFSAR Section 10.2.2 to Revise Description of Unit 2 Steam Dump Valve Use During Plant Cooldown

Description of Change

This change revised Operating Procedure 02-OHP-4021-052-001 to allow the operators to use nine steam dump valves in Modes 4 and 5 for RCS cooldown with the RCS temperature less than 350 degrees Fahrenheit (?F) until the RHR system was placed in service at approximately 180 ?F. The change included installing (and controls for removing) a temporary alteration in the form of a jumper that allowed all nine valves to operate in the "Steam Pressure" mode and removing fuses to defeat the Lo-Lo Tavg (P-12) isolation of the steam dump valves. Other P-12 and steam line isolation functions continued to be operable during the period that this alteration was installed.

Prior to the change the UFSAR described the use of six steam dump valves in the Steam Pressure mode of operation. Steam dump operation is further limited to three valves below Lo-Lo Tavg by P-12 design. The change was made to allow a more expeditious cooldown of the RCS, and therefore, shorten outage time.

The change also included an UFSAR Change, UCR-1597, to revise the Section 10.2.2 description of the use of Unit 2 steam dump valves during cooldown. The change revised the existing description of use of six valves to nine during Unit 2 cooldown when less than 350 ?F for consistency with the procedure change.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The change does not increase the frequency of, or create any, accidents or equipment malfunctions. There are no system interfaces created or affected by the change. Other than enabling use of nine bypass valves below 350 ?F, all other automatic isolation features associated with the P-12 interlock and steam line isolation are unaffected by the proceduralized temporary alteration. The change does not constitute the substitution of operator actions for automatic actions, based on review of Nuclear Regulatory Commission Information Notice 97-78 criteria. The administrative control of cooldown rate by the Operations Department procedures is not changed. The design and licensing basis and current accident analyses continue to bound this change. A calculation confirmed that the maximum heat removal rate at 350 ?F or less (when nine turbine bypass valves were available) was bounded by the maximum rate above 350 ?F (when limited to three valves). Based on the evaluation, prior NRC approval was not required.

9. Change Document Number

PMP-2270-EVL-002, Revision 0 and Fire Protection Program Manual, Revision 3

Title

Evaluation of Fire Protection Program Changes and Fire Protection Program Manual

Description of Change

This change issued a new procedure, PMP-2270-EVL-002, and revised the Fire Protection Program Manual to address the review and evaluation of fire protection program changes in accordance with the revised 10 CFR 50.59. The procedure provided for evaluation of changes for compliance with the fire protection license condition for each unit's operating license and for applicability of 10 CFR 50.59. The fire protection plan was updated to properly reflect the new review process for fire protection program changes and to reflect recent organizational title and management line of command changes for the fire protection program.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The change addresses the change review mechanism for fire protection program changes in accordance with revised regulatory requirements. The change does not introduce any new fire ignition sources or affect other analyzed accidents or equipment malfunctions that could compromise nuclear safety or the existing defense-in-depth features of the fire protection program. The administrative changes do not eliminate any responsibilities or compromise the efficiency of the fire protection plan. There are no adverse impacts on the ability to achieve and maintain safe shutdown in the event of a fire or on the fire hazards analysis. The change does not involve any system/component interfaces or involve or affect any equipment failure modes and effects. Based on the evaluation, prior NRC approval was not required.

OTHER CHANGES

10. Change Document Number

UCR-0850, Revision 3

Title

Changes to UFSAR Section 5.2, Unit 1 UFSAR Section 14.3.4, and Related Figures and Tables to Incorporate Revised Analyses

Description of Change

This change revised UFSAR Section 5.2 and Unit 1 UFSAR Section 14.3.4 as a result of reconstitution of the TMD computer code (short-term containment pressure) analysis and the performance of associated structural calculations and analyses to reestablish the design basis capability of the containment structures. Tables and figures associated with these sections affected by these analyses were also revised or were added. The change allowed closure of several operability evaluations that had determined that containment structures were operable but degraded. The changes to the UFSAR, performed per UCR-0850, Revision 3, included changes to text, figures, and tables to reflect the TMD and structural analysis results. Specific changes included revisions to discussions of the calculation models and assumptions, peak and differential pressures, and structural capacities.

10 CFR 50.59 Evaluation Summary

The evaluation of the change determined that a license amendment was not required. The evaluation concluded that the revised TMD computer code analysis and associated structural analyses did not increase the frequency of previously analyzed accidents or create the possibility of new accidents or malfunctions. The containment structures are not accident initiators and the changes did not create any new failure modes or interactions that could lead to new accidents or equipment malfunctions. The containment structures were determined to be fully capable of performing their design functions; therefore, the consequences of previously analyzed accidents or equipment malfunctions were also not increased. The containment interior structures support the containment structure, which was a fission product barrier. The design basis limits for the containment structures, the factored load combinations/capacities of UFSAR Section 5.2.2.3, continued to be met. Therefore, no design basis limit for the containment fission product barrier was exceeded. Although some evaluation methods changed from previous containment short-term pressure analyses, the newly utilized methods are consistent with methods previously reviewed and approved by the Nuclear Regulatory Commission.

11. Change Document Number

UCR-1607, Revision 0

Title

Change to UFSAR Tables 2.9-2, 6.2-1, 9.2-1, and 11.1-1 to Incorporate American Society of Mechanical Engineers Code, Section III (ASME III), Appendix F Applicability to Certain Piping Systems

Description of Change

The change incorporated applicability of ASME III, Appendix F, for certain piping systems during and following a DBA, into the CNP Licensing and Design Bases via UCR-1607, Revision 0. Incorporation of this criteria into the design and licensing basis was reflected by the addition of information related to the use of ASME III, Appendix F, criteria for the affected piping segments to UFSAR Tables 2.9-2, 6.2-1, 9.2-1 and 11.1-1. The change resolved Generic Letter 96-06 open issues and allowed closure of Operability Determination Evaluations associated with the affected systems.

10 CFR 50.59 Evaluation Summary:

The evaluation of the change determined that a license amendment was not required. The evaluation concluded that it was appropriate to apply the criteria of ASME III, Appendix F, instead of the code of record, USAS B31.1, to certain piping systems that met the concern of thermally induced overpressurization of isolated water-filled piping sections as a permanent licensing and design condition. The NRC approved the analysis method outlined in ASME III, Appendix F, for Nine Mile Point Nuclear Station as a replacement for USAS B31.1 in a supplemental SER to the utility, dated January 31, 2001. Because the NRC has previously approved the method for the intended application, the evaluation determined that this change did not constitute a departure from a method of evaluation.