UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
801 WARRENVILLE ROAD
LISLE, ILLINOIS 60532-4351

June 14, 2000

MEMORANDUM TO: Donald C. Cook Nuclear Plant Manual Chapter 0350 Panel

FROM: John A. Grobe, Director, DRS

SUBJECT: MINUTES OF INTERNAL MEETING OF THE DONALD C. COOK NUCLEAR PLANT MANUAL CHAPTER 0350 PANEL

The Donald C. Cook Nuclear Plant Manual Chapter 0350 Panel charter was announced on April 17, 1998, and most recently revised on March 3, 2000. One of the action items of this charter was to conduct internal meetings approximately twice per month. Subsequent experience has indicated that these meetings need to be held on a more frequent basis. These internal meetings are used to discuss significant technical and performance issues, NRC regulatory approach, and inspection resources and priorities. Attached for your information are the minutes from the internal meeting of the Inspection Manual Chapter 0350 Restart Panel held on June 7 and June 8, 2000.

Docket Nos. 50-315; 50-316

Enclosures: As stated

cc w/att: M. Satorius, NRR
J. Zwolinski, NRR
S. Singh Bajwa, NRR
C. Craig, NRR
J. Stang, NRR
J. Thompson, NRR
J. Grobe, RIII
G. Shear, RIII
A. Vegel, RIII
M. Holmberg, RIII
B. Bartlett, RIII
D. Passehl, RIII
K. Coyne, RIII
J. Maynen, RIII
Docket Files

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OFFICE	RIII:DRP	E	HQ:NRR	E	RIII:DRS			
NAME	AVegel/co AV		CCraig S. Bajwa for		JGrobe			
DATE	06/14/00		06/14/00		06/14/00			

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MEETING MINUTES: Internal MC 0350 Restart Panel Meeting on the D. C. Cook Nuclear Plant

DATE: June 7 and June 8, 2000

TIME: 2:00 p.m. Central (June 7, 2000)
8:30 a.m. Central (June 8, 2000)

ATTENDEES:

S. Bajwa
B. Bartlett
S. Black
M. Case (June 7 only)
R. Gardner (June 7 only)
J. Grobe
M. Holmberg
C. Lyon
J. Maynen
D. Passehl
R. Quirk (June 7 only)
W. Reckley (June 7 only)
G. Shear
J. Stang
A. Vegel
J. Zwolinski (June 7 only)

Discussion Topics:

1. *Plant Status and Inspector Insights*

The plant is in Mode 4. The licensee is preparing to perform a walkdown of the reactor coolant system while pressure in the reactor coolant system is 1000 psi. The licensee completed calibration of reactor coolant system resistance temperature detectors. During the June 8, 2000, meeting, the plant experienced a partial loss of offsite power. Mr. Bartlett and the resident staff initiated review of this event.

2. *Other*

- Messrs. Quirk and Gardner discussed the status of a concern regarding the safeguards test cabinet and the engineered safety features actuation system. Messrs. Quirk and Bartlett took action to brief licensee management on the concern.

- NRR took the lead to provide a point by point response to a memorandum from the Regional Administrator to Mr. Grobe regarding an operable but degraded containment concrete wall.

- The Inspection Manual Chapter 0350 Meeting Minutes from May 24, 2000, discussed closure of Case Specific Checklist Item 7, "Resolution of Nonsafety-Related Cables Going to Shunt Trip Coils." The minutes stated that there was an outstanding item regarding whether to pursue a backfit analysis based on risk. The Panel determined during the May 24, 2000, meeting that the level of risk did not warrant a backfit. Enclosed with this set of meeting minutes is NRR's analysis. (See Enclosure 1)

3. *Discuss Closure of Restart Action Matrix (RAM) Items*

The Panel reviewed and approved the revised documentation for Item R.3.8, "Control Room Habitability."

The Panel approved the following RAM items for closure:

- Item R.3.12, "Tornado Missile - Related issue on Unit 2: Missile issue for the Heating, Ventilation and Air Conditioning (HVAC) intake hoods located on the roof of the Electrical Switchgear Room and Spent Fuel Building"

- Item R.3.13,"HELB USQ - Licensing Basis Change Request for 10D on Plume and SRP, MEB 3-1 exclusion areas"

  (Note: The NRC Office of Nuclear Regulatory Research (RES) performed additional reviews. RES evaluated potential accident sequence precursors during their review of the licensee event reports and condition reports for D.C. Cook Units 1 and 2. For those issues identified as potential precursors, modifications and compensatory measures have been made by the licensee to address the concerns. RES identified no restart constraints.)

- Item R.3.14, "Methodology Changes to Steam Generator Tube Rupture (SGTR) Analysis: Original 30-minute operator action time to isolate the affected Steam Generator to prevent overfill was not supported by analysis"

- Item R.3.15, "Loss of AC and Feedwater Analyses Revision: Input changes on positive Moderator Temperature Coefficient (MTC) used to meet acceptance criteria, resulting in a reduction in safety margin for Unit 2"

- Item R.3.16, "Auxiliary Building Engineered Safety Feature Ventilation System (AES or ESF) Filtration System Bypass Damper Redundancy: The previous charcoal filter bypass dampers were installed in series; because of excess leakage rates they were replaced, however, the replacement dampers were installed in parallel and are subject to single failure issues"

- Item R.3.17, "Changes in Input Assumptions and the UFSAR for Transient Mass Distribution (TMD) Analysis: Reconstitution of Sub-Compartment Blowdown Analysis and Assumptions Resulted in Differential Pressures Higher than in the UFSAR"

Enclosure 2 provides details on the Inspection Manual Chapter 0350 Panel's assessment of the above items.

4. *Discuss Emergent Safety Issues for Commission.*

The Inspection Manual Chapter 0350 Restart Panel discussed the items below for possible emergent safety issues.

- 10 CFR 50.72 Reports
- Licensee Event Reports
- Inspection Findings
- Allegations
- Third Party Issues (e.g., Union of Concerned Scientists)
- Pending Investigations

- Pending Escalated Enforcement
- New Licensing Issues

The Panel identified no new issues.

5. *Discuss Status of Meeting Minutes*

   Mr. Passehl reported no timeliness concerns with issuance of outstanding meeting minutes.

6. *Discussion of Licensing Plan.*

   Mr. Stang reported no new licensing actions.

7. *Discuss/Update Milestones and Commitments.*

   The Panel discussed important upcoming meetings and deadlines.

8. *Other.*

   The latest Restart Action Matrix is included as Enclosure 3.

Enclosure 1

May 24, 2000

MEMORANDUM TO: John A. Grobe, Director
Division of Reactor Safety
Region III

FROM: Mark F. Reinhart, Acting Chief */RA/*
Probabilistic Safety assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Regulatory Regulation

Subject: D.C. COOK - RISK IMPLICATIONS ASSOCIATED WITH THE D.C. COOK USE OF THE NON-SAFETY RELATED CONTROL CABLES IN THE EMERGENCY DIESEL GENERATOR LOAD SHEDDING CIRCUITRY THAT ARE ROUTED IN COMMON TRAYS WITHOUT PHYSICAL SEPARATION TO PERFORM SAFETY-RELATED FUNCTION (TIA 99-031)

Per your request, the Operations Support Team of the Probabilistic Safety Assessment Branch (SPSB) in NRR provides its evaluation of the risk implications associated the use of nonsafety-related control cables in the emergency diesel generator (EDG) load shedding circuitry that are routed in common trays without physical separation to perform the safety-related function at D.C. Cook. Our evaluation is attached.

Based on our expedited evaluation of the issue, we find that the risk impact of this condition is insignificant.

Should you have any questions, contact Ian Jung at 301-415-1837.

Attachment: As stated

Enclosure 1

Risk Implications: Use of nonsafety-related control cables in the EDG load shedding circuitry that are routed in common trays without physical separation to perform the safety-related function.

Load shedding of nonsafety-related loads during loss of offsite power (LOOP) is a component for determining overall emergency diesel generator (EDG) system reliability. The impact on EDG reliability due to the as-found condition is difficult to measure; however, review of this issue and industry experience have shown that it is most likely insignificant. Examination of the Accident Sequence Precursor (ASP) reports (NUREG/CR-4674, Vol. 14 - Vol. 26[1]) since 1990 indicated that there were no ASP events of similar nature for all nuclear power plants. Search of the Licensee Evaluation Reports (LERs) for all sites since 1985 identified only one similar issue (Accession No. 9208050265), but the issue was concluded to be of negligible risk significance due to the similar reasons on which the conclusion of this evaluation is based. Plant-specific operating experience also indicates that D.C. Cook did not identify any actual failures/events associated with these control cables. It is believed that a cable fault in these low-energy control cables is rare and, in addition, the cable interaction that causes multiple cable failures in the cable tray appears unlikely based on the design characteristic, i.e., protective devices, and experience. The licensee's testing results also indicate that the postulated fire scenario in these low energy cables, rendering the load shedding capability for both trains inoperable, is relatively unlikely.

In addition, relevant core damage scenarios and their frequencies were examined to gain risk insights. For design basis accidents such as a loss of coolant accident (LOCA)-induced LOOP, the overloading condition is more likely without proper load shedding since the emergency core cooling system (ECCS) pumps would be actuated. However, their initiating event frequency is typically very small (below 1E-5/yr: NUREG/CR-6538[2] and NUREG/CR-5750[3]). For a small LOCA-induced LOOP, the time available for operator recovery would be more substantial and the recovery probability of offsite power is generally high. For events such as LOOP, transient-induced LOOP, and LOOP-induced LOCA, it is either; (a) less likely to have an EDG overloading condition even without load shedding due to no ECCS actuation, or (b) the initiating event frequency is generally low. Furthermore, the potential for operator recovery, i.e., manual load shedding and EDG re-start, would also be possible in some scenarios. External events such as fire, flooding, earthquake, and missile were considered. The scenarios of concern are associated with an external event that results in a common cause cable failure with a concurrent LOOP. These scenarios are believed to be very unlikely due to the low initiating event frequency. These are in addition to the fact that the impact on the EDG reliability would likely be insignificant. More detailed information from the licensee would be required to credit these aspects; however, these insights generally strengthen the insignificant risk implications of the as-found condition.

In summary, the staff believes that the impact on the EDG reliability is insignificant, and it is further supported by probabilistic risk insights. Therefore, the staff concludes that the risk implications of the as-found condition are insignificance.

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[1]NUREG/CR-4674, Vol. 14 - 26, "Precursors to Potential Core Damage Accidents," 1991 - 1998

[2]NUREG/CR-6538, "Evaluation of LOCA with Delayed LOOP and LOOP with Delayed LOCA Accident Scenarios," July 1997

[3]NUREG/CR-5750, "Rates of events at U.S. Nuclear Power Plants: 1987 -1995," February 1999