CATEGORY 2

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9711180014 DOC.DATE: 97/10/09 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana M 05000316

AUTH.NAME AUTHOR AFFILIATION
LOCHBAUM,D.A. Union of Concerned Scientists
RECIP.NAME RECIPIENT AFFILIATION
CALLAN,L.J. Ofc of the Executive Director for Operations

SUBJECT: Submits petition per 10CFR2.206 requesting that operating licenses be modified,revoked or suspended until reasonable assurance that sys in conformance w/design & licensing requirements.

DISTRIBUTION CODE: DF01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 14
TITLE: Direct Flow Distribution: 50 Docket (PDR Avail)

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
INTERNAL:	FILE CENTER 01	1	1	NUDOCS-ABSTRACT	1	1
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

C A T E G O R Y 2 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

TOTAL NUMBER OF COPIES REQUIRED: LTTR 4 ENCL 4

UNION OF
CONCERNED
SCIENTISTS

October 9, 1997

Mr. L. Joseph Callan
Executive Director for Operations
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: **PETITION PURSUANT TO 10 CFR 2.206, DONALD C. COOK NUCLEAR PLANTS UNITS 1 AND 2, DOCKET NOS. 50-315 AND 50-316**

Dear Mr. Callan:

The Union of Concerned Scientists submits this petition pursuant to 10 CFR 2.206 requesting that the operating licenses for Donald C. Cook Units 1 and 2 be modified, revoked, or suspended until there is reasonable assurance that their systems are in conformance with design and licensing bases requirements. A process comparable to the system certifications recently used by the Salem and Millstone licensees would provide this necessary level of assurance. UCS additionally requests that a public hearing into this matter be held in the Washington, DC area prior to the first unit at D C Cook being authorized to restart. At this hearing, we will present information supporting the contentions in this petition.

Background

On October 9, 1996, the NRC requested that its power reactor licensees provide information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design bases information. The NRC's issued this request as a result of its investigations at the Millstone Power Station. The licensee for the D C Cook plant responded with a letter dated February 6, 1997, describing the administrative controls it uses to provide assurance that the Cook Nuclear Plant is operated and maintained within the established design bases.

An NRC team recently conducted an architect/engineer design inspection at D C Cook. According to the NRC's Project Manager for D C Cook, this NRC team examined two safety systems and their supporting systems. The team's findings forced the licensee to shut down both units on September 10, 1997.

The NRC issued a confirmatory action letter to the licensee dated September 19, 1997, specifying issues arising from the design inspection that must be resolved prior to restarting the units. These issues (listed in Attachment 1) include physical modifications to the plants and revisions to the plants' operating licenses. Numerous NRC Daily Event Reports (listed in Attachment 2) described the findings from design inspection as reported by the licensee. The NRC has not yet released the design inspection report and we have been told that it will not be issued until next week at the earliest.

DF01 |
cc: D. Hood

9711180014 971009
PDR ADOCK 05000315
H PDR

Washington Office: 1616 P Street NW Suite 310 • Washington, DC 20036-1495 • 202-332-0900 • FAX: 202-332-0905
Cambridge Headquarters: Two Brattle Square • Cambridge, MA 02238-9105 • 617-547-5552 • FAX: 617-864-9405
California Office: 2397 Shattuck Avenue Suite 203 • Berkeley, CA 94704-1567 • 510-843-1872 • FAX: 510-843-3785

PRINTED ON RECYCLED PAPER

Basis for Requested Action

The NRC conducted architect/engineer design inspections at only six of its nearly 70 operating power reactor licensee sites. These design inspections examined only one or two safety systems along with their supporting systems at each site. The NRC Project Manager reported that the design inspection at D C Cook examined the residual heat removal and component cooling water systems along with their supporting systems. These design inspections focused on the facilities' original design and the licensees' conformance with the safety analysis reports.

The systems examined by the NRC at D C Cook had already been covered by the licensee's design basis documentation reconstitution program. Design basis documents (DBDs) for the containment, containment structure, containment spray, emergency core cooling, component cooling water, and residual heat removal systems had been approved by the licensee prior to the NRC team's arrival. The licensee informed the NRC that its DBD program had not identified any deficiencies involving equipment operability.

The findings by the NRC design inspection team prompted the licensee to declare both trains of the emergency core cooling systems and the containment spray system inoperable. The units were shut down on September 8 and 9, 1997. The licensee reported making physical changes to the plant to correct some of the problems and indicated that additional physical changes may be required.

The licensee has proposed fixing the specific operability issues identified during the NRC design inspection and then restarting the units. Confining the scope of the restart activities in this way would be treating the symptoms rather than the cause of the problems. The NRC design inspection revealed serious deficiencies in the licensee's design control programs. These deficiencies created the specific problems that forced the plants to be shut down. These deficiencies may also be responsible for similar problems in other safety systems which were not examined by the NRC.

It is important to note that the NRC identified significant operability problems in systems that the licensee had covered in recently approved DBDs. The licensee stated in its February 6, 1997, submittal that it verifies and validates the information in its DBDs via reviews and physical plant walkdowns prior to their approval. Thus, the NRC discovered significant problems in systems which had been closely scrutinized by the licensee. Had the NRC's findings involved systems which have not yet been covered under the licensees' DBD program, it might be reasonable to assume that the licensee would have identified them at that later date. However, there is little reason to believe that these problems would have been resolved unless the NRC had identified them.

Attachment 2 lists NRC Daily Event Reports (DERs) involving issues identified by the NRC design inspection at D C Cook. DER Nos. 32740, 32806, 32822, 32839, 32843, 32875, 32890, 32904, 32914, 32915, 32921, 32948, and 32988 describe potential deficiencies that appear to have existed at D C Cook prior to the initiation of its design basis documentation reconstitution effort in 1992. That effort was therefore apparently unable to detect these potential deficiencies. DER Nos. 32823, 32824, 32903, 32939, and 32948 describe potential deficiencies that appear to have been introduced since 1992. Thus, the licensee's design control and quality assurance programs are apparently unable to ensure that the facility is maintained within its design bases.

UCS feels that the design basis documentation reconstitution and Updated Final Safety Analysis Report (UFSAR) validation programs as described in the licensee's response to the NRC's 50.54(f) letter lack the rigor and focus necessary to identify potential design-related operability issues. Our conviction is supported by the findings from the NRC design inspection. Since the corrections to the NRC's findings were not limited to mere paperwork fixes but included actual changes to the plant's physical configuration, the safety significance of these and potentially other undetected problems cannot be understated.

The flaws in the licensee's design control programs must be corrected. The systems at D C Cook, at least those with a safety function, must be certified to be capable of performing their required actions under all design conditions. Then, and only then, can the units be restarted with reasonable assurance that public safety will be adequately protected. It would be irresponsible to restart these units knowing that the programmatic failures that caused the safety problems identified by the NRC team may have produced comparable problems affecting the operability of other safety systems.

The legal precedent for our position is stated by the NRC's Atomic Safety and Licensing Appeal Board in the Matter of Vermont Yankee Nuclear Power Corporation , Memorandum and Order (ALAB-138), dated July 31, 1973:

> "As a general rule, the Commission's regulations preclude a challenge to applicable regulations in an individual licensing proceeding. 10 CFR 2.758. This rule has been frequently applied in such proceedings to preclude challenges by intervenors to Commission regulations. Generally, then, an intervenor cannot validly argue on safety grounds that a reactor which meets applicable standards should not be licensed. By the same token, neither the applicant nor the staff should be permitted to challenge applicable regulations, either directly or indirectly. Thus, those parties should not generally be permitted to seek or justify the licensing of a reactor which does not comply with applicable standards. Nor can they avoid compliance by arguing that, although an applicable regulation is not met, the public health and safety will still be protected. **For, once a regulation is adopted, the standards it embodies represent the Commission's definition of what is required to protect the public health and safety.**" [emphasis added]
>
> "In short, in order for a facility to be licensed to operate, the applicant must establish that the facility complies with all applicable regulations. **If the facility does not comply, or if there has been no showing that it does comply, it may not be licensed.**" [emphasis added]

The NRC design inspection at D C Cook identified significant issues which caused both units to be shut down. These issues were caused by programmatic deficiencies in the licensee's design control programs. A contributing factor for these issues is the failure of the licensee's quality assurance and self-assessment programs to detect these problems. Nothing in the reported findings from the design inspection supports a conclusion that these findings are isolated consequences. The NRC's design inspection invalidates any showing that this facility complies with all applicable regulations. Therefore, the design control deficiencies must be corrected to prevent future non-compliances with safety regulations. And just as importantly, a thorough review of all systems with safety functions must be completed prior to restart to detect and correct past non-compliances.

UCS is not advocating that the NRC apply a higher standard at D C Cook. Instead, we are requesting that the NRC ensure that the D C Cook facility is in accordance with the minimum safety standards which constitute the legal grounds for allowing the units to operate. Our request is consistent with the measures required by the NRC when other sampling inspections find problems. We ask the NRC to expand the inspection scope based upon the identified problems just as would be required when snubber (e.g., pipe restraint) and reactor vessel internals inspections found problems.

Requested Actions

UCS petitions the NRC to protect public health and safety by preventing the units at D C Cook from operating until such time that there is reasonable assurance that all significant non-compliances have been identified and corrected. The system certification process recently used at the Salem Generating Station and the Millstone Power Station would provide such reasonable assurance. We request a public hearing on this matter be held in the Washington, DC area before any unit at D C Cook is authorized to restart.

Sincerely,

David A. Lochbaum

David A. Lochbaum
Nuclear Safety Engineer

cc: Chairman Shirley Ann Jackson
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Mr. A. B. Beach, Regional Administrator
United States Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Honorable Fred Upton
United States House of Representatives
Washington, DC 20515-2206

Honorable Spencer Abraham
United States Senate
Washington, DC 20510-2203

Honorable Carl Levin
United States Senate
Washington, DC 20510-2202

Attachments:

1) Design Inspection Issues That Will Be Resolved Prior to D C Cook Restart
2) NRC Daily Event Reports on D C Cook Design Inspection Findings

Attachment 1
Design Inspection Issues That Will Be Resolved Prior to D C Cook Restart

The following issues, quoted verbatim, were specified on the NRC's Confirmatory Action Letter dated September 19, 1997, as requiring resolution prior to restart of any D C Cook unit:

1. **Recirculation Sump Inventory/Containment Dead Ended Compartments Issue**

   Analyses will be performed to demonstrate that the recirculation sump level is adequate to prevent vortexing, or appropriate modifications will be made. [See also Attachment 2 - Power Reactor Event Number 32890]

2. **Recirculation Sump Venting Issue**

   Venting will be re-installed in the recirculation sump cover. The design will incorporate foreign material exclusion requirements for the sump. [See also Attachment 2 – Power Reactor Event Numbers 32875 and 32903]

3. **Thirty-six Hour Cooldown, with One Train of Cooling**

   Analyses will be performed that will demonstrate the capability to cool down the units consistent with design basis requirements and necessary changes to procedures will be completed.

4. **ES-1.3 (Switchover to Recirculation Sump) Procedure**

   Changes to the emergency procedure used for switchover of the emergency core cooling and containment spray pumps to the recirculation sump will be implemented. These changes will provide assurance there will be adequate sump volume, with proper consideration of instrument bias and single failure criteria. [See also Att. 2 – Power Reactor Event Numbers 32806 and 32904]

5. **Compressed Air Overpressure Issue**

   Overpressure protection will be provided downstream of the 20 psig, 50 psig, and 85 psig control air regulators to mitigate the effects of a postulated failed regulator. [See also Attachment 2 – Power Reactor Event Numbers 32939 and 32988]

6. **Residual Heat Removal (RHR) Suction Valve Interlock Issue**

   A technical specification change to allow operation in mode 4 with the RHR suction valves open and power removed is being processed. Approval of this change by the NRC will be required prior to restart. [See also Attachment 2 – Power Reactor Event Numbers 32914 and 32921]

7. **Fibrous Material in Containment**

   Removal of fibrous material from containment that could clog the recirculation sump will be completed. [See also Attachment 2 – Power Reactor Event Number 32948]

Attachment 2
NRC Daily Event Reports on D C Cook Design Inspection Findings

The following summaries were taken from the daily event reports available on the NRC's website (www.nrc.gov). The only editing involved deletion of unnecessary detail, such as who was notified about the events, and the addition of clarification for acronyms. Otherwise, these narratives are verbatim.

POWER REACTOR EVENT NUMBER: 32890
UNUSUAL EVENT DECLARED & TECHNICAL SPECIFICAITON REQUIRED SHUTDOWN ON BOTH UNITS DUE TO INOPERABLE CONTAINMENTS

As a result of issues raised during the ongoing architect/engineer design inspection, the licensee was reviewing the design aspects of the containments (both units have similar containments). After consulting with the nuclear steam supply system supplier (Westinghouse) the licensee determined that concerns existed about whether adequate communication (flow paths) exists between the active and inactive portions of the containment sump.

During certain scenarios, the volume of water flow back to the containment recirculation sump may not be adequate to support long-term emergency core cooling (ECC) systems (RHR [residual heat removal] system, safety injection system, charging system) or containment spray pump operation during the recirculation phase of a large or small break LOCA. The containment drainage system is designed to ensure that water entering the containment from the breach in the reactor coolant system, ECC systems injection, and ice condenser melt flows back into the containment recirculation sump via drains. Licensee analysis was unable to confirm that sufficient communication existed between inactive and active volumes of the containment to ensure adequate drainage to the recirculation sump. Without adequate drainage into the sump, a low sump level will result, which jeopardizes long term operation of the ECC Systems and containment spray pumps due to vortexing and air entrainment.

As a conservative measure because of these concerns, the licensee declared both trains of the ECC Systems and the containment spray system inoperable for both units and entered Technical Specification limiting condition for operation action statement 3.0.3 to shut down both units. The licensee commenced shutting Unit 1 down from 100% power at 1655 and Unit 2 down from 100% power at 1728. At 2000, the licensee declared an unusual event on both units due to the potential loss of containment barrier on both units.

The licensee plans to perform further analysis to determine the extent of the existing communication between the portions of the sumps and whether plant modifications will be necessary.

Update @ 0311 EDT on 09/10/97 by Tilly taken by MacKinnon

The unusual event was terminated and exited at 0303 EDT when Unit 1 entered mode 5 (cold shutdown). Unit 2 entered mode 5 at 0015 EDT (cold shutdown).

POWER REACTOR EVENT NUMBER: 32875
FAILURE TO MAINTAIN THE CONTAINMENT RECIRCULATION SUMP 1/4" PARTICULATE RETENTION REQUIREMENT (HISTORICAL ISSUE).

A 1/4" particulate retention requirement for the containment recirculation sump was not properly established in 1979 following sump modifications. The containment recirculation sump requirement to retain 1/4" particles is to ensure that containment spray nozzles do not become plugged. The containment spray system takes suction from the containment recirculation sump following injection of the refueling water storage tank supply during a loss of coolant accident.

In 1979, modifications were performed on the containment recirculation sump. One of the modifications involved moving a 1/4" retention element from inside the recirculation sump to the entrance of the sump. When the retention element was moved, the 1/4" retention requirement was not fully addressed, and pathways exceeding the 1/4" requirement were inadvertently established. The inadvertent pathways established included: 3/4" vents in the roof of the recirculation sump entrance, the containment sump drain line from the recirculation sump, and small gaps around the sump entrance. These pathways have since been eliminated or the 1/4" requirement has been established.

The licensee is reporting the fact that since 1979 until the 1/4" requirement was established or the pathway was eliminated, the containment recirculation sump did not meet its design requirement.

The containment recirculation sump currently meets the 1/4" requirement. A condition report has been written to initiate investigation into this event and determine appropriate preventive actions.

This event was determined to be reportable at 0856 on September 5, 1997.

*** Update at 1905 on 09/10/97 by Randy Ptacek entered by Jolliffe ***

After further review of the above condition, the licensee concluded that the emergency core cooling (ECC) system was outside its design basis as a result of the 1/4" requirement not being met following the 1979 plant modifications. By not adequately covering the 1/4" particulate retention requirement, larger particles had the potential to enter the recirculation sump. The ECC System has not been analyzed for these larger particles nor is it within the design of the ECC System to handle these larger particles.

The licensee has concluded that this event is also reportable to the NRC in accordance with the requirements of 10CFR50.72(b)(1)(ii)(a) unanalyzed condition, and 10CF50.72(b)(2)(iii)(d)accident mitigation.

POWER REACTOR EVENT NUMBER: 32903
CONTAINMENT RECIRCULATION SUMP VENT HOLES HAVE BEEN FILLED WITH CONCRETE

As a result of questions posed by the NRC architect/engineer design inspection team, the licensee determined that the inlet venting requirement for the containment recirculation sumps was not properly maintained following modifications to the Unit 2 sump in 1996 and the Unit 1 sump in 1997 (both units have similar containments).

The containment recirculation sump venting requirement was established in 1979 as part of the original sump design to reduce the potential for air entrainment through the sump. The venting requirement was met through the addition of five 3/4-inch diameter holes drilled in the roof of the sump inlet. (The holes did not meet the 1/4-inch diameter requirement as reported in Event #32875.) When these holes were discovered during the Unit 2 1996 refueling outage and the Unit 1 1997 refueling outage, they were classified as abandoned equipment holes that exceeded the 1/4-inch particulate retention requirement for the sumps and they were filled with concrete.

POWER REACTOR EVENT NUMBER: 32806
INSTRUMENTATION INDICATIONS USED TO DETERMINE WHEN REFUELING WATER STORAGE TANK TO CONTAINMENT SWITCHOVER IS REQUIRED MAY NOT HAVE BEEN CORRECT TO PREVENT VORTEXING IN THE CONTAINMENT RECIRCULATION SUMP.

During the evaluation of a proposed procedure change that affects switchover from the refueling water storage tank (RWST) to the containment sump during a loss-of-coolant accident (LOCA), it was determined that the instrumentation indications used to determine when the switchover is required may not have been correct to prevent vortexing in the containment recirculation sump.

To address this situation, procedures associated with the switchover (on both units) have been conservatively changed to accommodate the related instrument inaccuracies. These changes assure adequate RWST water is in containment before switchover to eliminate concerns that vortexing would occur in the containment sump after switchover.

The problem is that the RWST water level indicators are connected to the suction line that goes to the residual heat removal (RHR) pumps. Due to the flow in these lines, the indicated water level at which the switchover would be initiated would be less than the actual water level of the RWST (the licensee would be putting less water into the containment than expected). Also, the licensee said that they had some inaccuracies associated with their containment sump instrumentation. The licensee adjusted the containment sump indication to assure that they have an adequate volume in the containment to prevent vortexing. The licensee relies upon two indications for switchover; RWST water level and containment water level.

POWER REACTOR EVENT NUMBER: 32904
SINGLE FAILURE DURING RECIRC SUMP SWITCHOVER COULD BE UNANALYZED CONDITION

As a result of questions posed by the NRC architect/engineer design inspection team, the licensee determined that the possibility of a single failure during an accident while performing switchover of the emergency core cooling system pumps from the refueling water storage tank (RWST) suction to the recirculation sump suction could have resulted in the plant being in an unanalyzed condition. This condition is outside the plant design basis, and it potentially could have prevented the fulfillment of a safety function of structures or systems.

The plant emergency operating procedures (EOPs) as currently written require that the west residual heat removal (RHR) pump be the first pump switched from the RWST suction to the recirc sump suction. Once this is accomplished, the centrifugal charging (CC) pumps' suctions and the safety injection (SI) pumps' suctions are then swapped from the RWST supply to the discharge of the west RHR pump. If the west RHR pump were to fail at this point when all CC and SI pumps were being supplied from its discharge, prior to the east RHR pump suction being transferred from the RWST to the recirc sump, all CC and SI pumps could also fail due to the loss of suction flow. This would result in the loss of all high and medium head injection with only the flow from the east RHR pump available for injection into the reactor coolant system. The licensee is currently reviewing the EOPs to determine an alternate switchover sequence that would eliminate the condition as described above.

POWER REACTOR EVENT NUMBER: 32939
INSTALLED PLANT MODIFICATION INTRODUCED THE POSSIBILITY OF A SINGLE FAILURE WHICH COULD RESULT IN THE LOSS OF BOTH TRAINS OF THE ESF VENTILATION SYSTEM.

At 1620 on 09/16/97, the licensee determined that a plant modification installed between December 1996 and August 1997 introduced the possibility of a single failure which could result in the loss of both trains of the engineered safety features (ESF) ventilation system if the 85-psi air header was to be lost. Prior to the installation of the plant modification, the ESF ventilation system charcoal inlet and bypass dampers both utilized a 20-psi air header and were positioned such that the charcoal bypass dampers were normally open and would fail closed; and the charcoal inlet dampers were normally closed and would fail open. The plant modification installed new bypass dampers which required higher air pressure to operate and were, therefore, transferred to the 85-psi header. If the 85-psi air header was lost, it would result in the repositioning of the normally open bypass dampers without the opening of the charcoal inlet dampers on both trains. This would result in dead heading of the filter train fans and loss of cooling to emergency core cooling system (ECCS) equipment.

POWER REACTOR EVENT NUMBER: 32988
NON-SAFETY RELATED AIR HEADERS LACK OVERPRESSURE PROTECTION

During an architectural engineering inspection a question was raised regarding the lack of overpressure protection on the 20, 50 and 85 psig control air headers. The specific concern is the potential for common mode failure of both trains of safety related equipment served by the air headers. The overpressure condition is postulated to be caused by regulator failure.

Although system reviews have found no component failure mode which would result in the devices being incapable of going to their fail-safe position, a design change package has been prepared to provide overpressure protection on the 20, 50 and 85 psig headers.

POWER REACTOR EVENT NUMBER: 32914
LICENSEE IDENTIFIED THAT BOTH UNITS HAD OPERATED THEIR RHR SYSTEM CONTRARY TO THE DESCRIPTION IN THE FSAR.

At 1615 EDT, with Units 1 and 2 shutdown in mode 5, it was determined that both units have operated contrary to the design basis for the residual heat removal (RHR) system as described in the Final Safety Analysis report (FSAR). FSAR Chapter 9, Section 9.3, describes the interlocks associated with the residual heat removal (RHR) suction valves from the reactor coolant system (RCS). The suction line valves are interlocked through separate channels of the RCS system pressure signals to provide automatic closure of both valves whenever RCS pressure exceeds RHR design pressure. The FSAR states that the interlock may be defeated when the RCS is open to atmosphere. However, for a number of years this interlock has been procedurally defeated on both units to prevent inadvertent closure and loss of RHR suction during shutdown cooling operation by opening the valves and racking out their breakers in mode 4.

The overpressure protection afforded by the automatic closure function described in the FSAR was defeated without a safety evaluation being performed. This loss of automatic closure function represents an unanalyzed condition and is, therefore, reportable.

Plans are to degas, depressurize, and open the RCS on both units to atmosphere. Degas will start on Unit 1, and when completed, the unit will proceed to depressurize while Unit 2 starts degas procedures. When the RCS is open to atmosphere on both units, the plant will be in compliance with the FSAR.

This condition was identified by the licensee during an ongoing NRC architect/engineer inspection.

*** Update at 2130 EDT on 9/13/97 from Robert Blyth to S. Sandin ***

The licensee has completed its safety evaluation for mode 5 operation and concluded that there was no unreviewed safety question or change of operation as described in the FSAR. Consequently, degas of Unit 1 has been terminated, and neither unit will be vented to atmosphere.

POWER REACTOR EVENT NUMBER: 32921
THE LICENSEE IDENTIFIED THAT BOTH RHR PUMPS HAD BEEN OPERATED WHEN THE RCS WAS DEPRESSURIZED, WHICH IS CONTRARY TO THE DESCRIPTION IN THE FSAR.

Chapter 9 of the Final Safety Analysis Report (FSAR) states: 'Only one residual heat removal (RHR) pump will be operated when the reactor coolant system is open to atmosphere to prevent damaging both pumps in the unlikely event that suction should be lost.' Operating procedures for the RHR system do not prevent operation of both RHR pumps when the reactor coolant system (RCS) is open to atmosphere, and in the past, both RHR pumps have been run when the RCS was vented to atmosphere.

Plant operating procedures are being reviewed to determine the impact. Procedure changes will be implemented as necessary to address the FSAR requirement. A condition report has been initiated to investigate and determine appropriate preventative actions.

POWER REACTOR EVENT NUMBER: 32948
IT WAS DETERMINED THAT FIBROUS MATERIAL IS PRESENT IN BOTH UNIT 1 AND UNIT 2 CONTAINMENT IN ENOUGH QUANTITY TO POTENTIALLY CAUSE EXCESSIVE BLOCKAGE OF THE CONTAINMENT RECIRCULATION SUMP SCREEN DURING THE RECIRCULATION PHASE OF A LOSS OF COOLANT ACCIDENT.

In 1985, 1986, and 1995 "Fiberfrax" refractory insulation materials in bulk, blanket or board form were used as damming material when installing fire stops in cable trays in both containments. The specification governing installation of the fire stops did not require removal of the material, only stating that it should be removed "if necessary." The material was not removed. The material is present in 12 cable trays in Unit 1 and 15 cable trays in Unit 2.

When the Fiberfrax is exposed to water or steam/water environment it could potentially break into small pieces, which could be transported to the recirculation sump by the water flow in containment during a loss of coolant accident. Once it reaches the recirculation sump it has the potential to clog the screens in excess of the design value. Excessive screen blockage could result in ECCS inoperability during the recirculation mode.

The Fiberfrax material is currently being removed from the containments, and removal will be completed prior to restart of the units. The possibility that the licensee's work control process allowed unencapsulated fibrous material to be installed in other locations inside containment is being investigated.

Attachment 2 (continued)
NRC Daily Event Reports on D C Cook Design Inspection Findings

POWER REACTOR EVENT NUMBER: 32740
UNITS 1 & 2 OPERATED OUTSIDE THE DESIGN BASIS FOR SERVICE WATER INLET TEMP

As a result of questions posed by members of the ongoing NRC design inspection team, the licensee has determined that Units 1 & 2 have operated outside the plant design basis for service water inlet temperature.

The Updated Final Safety Analysis Report (UFSAR), Table 9.5-3, lists service water inlet temperature design value as 76°F. This value is used as input to analyses such as containment peak pressure and control room habitability. Although engineering analyses were performed in 1988 raising the temperature to 87.5°F as listed in the plant Technical Specifications, a 10CFR50.59 safety evaluation was never performed, nor was the UFSAR properly revised.

Plant service water inlet temperature is the same as Lake Michigan water temperature. A review of historical data indicates that during July and August of any year, Lake Michigan water temperature is likely to exceed the 76°F value. Specific data for 1997 shows that Lake Michigan water temperature, and therefore plant service water inlet temperature, was greater than 76°F on July 17, July 18, and August 4, 1997. All plant systems which utilize service water as a cooling medium have been determined to be operable. A 10CFR50.59 safety evaluation will be performed and appropriate changes will be incorporated into the UFSAR.

This report is intended to cover any temperature exclusions above 76°F and below the 87.5°F value listed in the plant Technical Specifications that may occur prior to the completion of the 10CFR50.59 safety evaluation.

POWER REACTOR EVENT NUMBER: 32822
DISCOVERY THAT A NORMAL OPERATING PROCEDURE ALLOWED PLANT OPERATION WITH COMPONENT COOLING WATER HEAT EXCHANGER OUTLET TEMPERATURES GREATER THAN THE DESIGN LIMIT SPECIFIED IN THE FINAL SAFETY ANALYSIS REPORT

During the ongoing NRC architect/engineer design inspection, a question was asked relative to a statement used in the normal operating procedure for the component cooling water (CCW) system. The statement allows for a heat exchanger outlet temperature for CCW to reach 120°F for a period of 3 hours during normal cooldown on the residual heat removal system. Investigation revealed that this statement was in the original issue of the procedure in 1976. However, no 10 CFR 50.59 unreviewed safety evaluation determination documentation could be found to support this design parameter.

The licensee's Final Safety Analysis Report (FSAR) states that the CCW heat exchanger outlet design temperature is 95°F. Based on the FSAR requiring the 95°F outlet temperature and the lack of an unreviewed safety question determination to justify operation exceeding 95°F, the units were in a condition that allowed operation outside the design basis because the procedure allowed operation up to 120°F for a period of 3 hours during normal cooldown on the residual heat removal system. The units are not currently in a Technical Specification limiting condition for operation as a result of this issue.

Procedure changes have been made to remove the inappropriate statement. A condition report has also been written to initiate an investigation into this event and determine appropriate preventive actions.

POWER REACTOR EVENT NUMBER: 32823
FAILURE OF A SAFETY REVIEW TO ADDRESS FINAL SAFETY ANALYSIS ATTRIBUTES ON ASSOCIATED COMPONENT COOLING WATER COOLING REQUIREMENTS

During the ongoing NRC architect/engineer design inspection, a question was asked relative to dual train component cooling water (CCW) system outages. During dual train CCW outages, CCW cooling is supplied to the spent fuel pool (SFP) heat exchanger only from the opposite unit. If that unit has a loss of coolant accident (LOCA), CCW to the SFP heat exchanger will isolate. Final Safety Analysis Report (FSAR) Table 9.5-2, footnote 3, indicates that the SFP heat exchanger is assumed to be on the non-accident unit.

The licensee reported the following inspection questions:

1) Does a dual train CCW outage represent a condition outside the plant design basis?

2) Was this reviewed as part of the process of allowing a dual train CCW outage?

Based on a review of FSAR Table 9.5-2, it was concluded that footnote 3 was established to clarify why no values for SFP heat exchanger flow for the unit undergoing the LOCA are listed in the table. Footnote 3 reflects normal SFP cooling system design and operation.

A review was performed of the safety evaluation performed for the Unit 2 full core offload with one train of spent fuel cooling. This safety review covered the Unit 2 refueling outage schedule which included a dual train CCW outage.

Footnote 3 of Table 9.5-2 represents the normal design of the SFP cooling system, that is, the SFP cooling system is designed to remove the heat generated by stored spent fuel elements in the [SFP]. The system incorporates two separate trains.

The safety review for the Unit 2 full core offload with one train of spent fuel cooling addressed the FSAR section 9.4 attribute of the SFP cooling dealing with time to boil events and bulk pool temperature requirements; however, the safety review failed to address FSAR section 9.5 attributes associated CCW cooling requirements as given in Table 9.5-2.

This issue impacts both units. However, the units are not currently in a Technical Specification limiting condition for operation as a result of this issue.

POWER REACTOR EVENT NUMBER: 32824
FAILURE TO PERFORM A 10 CFR 50.59 EVALUATION FOR A PROCEDURE CHANGE INVOLVING COMPONENT COOLING WATER HEAT EXCHANGER OUTLET TEMPERATURE LIMITS

During the ongoing NRC architect/engineer design inspection, a question was asked relative to the fact that during the last Unit 2 refueling outage, an administrative limit of 90°F was placed on the component cooling water (CCW) system. The thermal analysis indicated that a maximum CCW temperature of 90°F would eliminate all margin associated with the spent fuel pool (SFP) design assuming a design flow of 3,000 gpm.

The following inspection question was asked: Since a change in CCW temperature was required to meet the Final Safety Analysis Report (FSAR) value of 160°F for the SFP, was a 10 CFR 50.59 unreviewed safety evaluation performed?

The licensee reviewed the change to the procedure to limit CCW temperature to 90°F. The licensee considered this change to be an administrative change only to lower the allowable temperature to the SFP cooling heat exchanger. A 10 CFR 50.59 evaluation was not performed because it was not recognized that the 95°F requirement was essentially being changed.

Without the completion of an unreviewed safety question determination, the plant was in a condition outside the design basis. The units are not currently in a technical specification limiting condition for operation as a result of this issue.

A condition report has been written to initiate actions to investigate this event and provide preventive actions. The 90°F limit is no longer in the operating procedures.

POWER REACTOR EVENT NUMBER: 32839
AVAILABLE WATER VOLUME IN RWST NOT ADEQUATE IN MODES 5 AND 6

During the ongoing NRC architect/engineer design inspection, NRC inspectors asked a question about the reactor coolant makeup required after a 10CFR50, Appendix R fire. To respond to the question, the licensee reviewed two associated design calculations. The more restrictive calculation was determined to be the calculation of record to meet the requirement. This calculation requires 87,000 gallons of water to be available in the refueling water storage tank (RWST). The value of 87,000 gallons was approved on 02/20/90. During modes 1 through 4, plant procedures adequately ensure that this requirement is met. During modes 5 and 6, plant procedures are not adequate to ensure that this requirement is met.

The plant has been in modes 5 and 6 many times since this requirement became effective on 02/20/90. Based on this, the plant has been in an unanalyzed condition several times since 02/20/90.

Currently both units are in mode 1. The licensee is reviewing plant operating procedures to determine impact and will implement procedure chances as needed prior to either unit entering modes 5 or 6. The licensee is continuing to evaluate the subject calculations and plans to submit a licensee event report to the NRC on this subject.

POWER REACTOR EVENT NUMBER: 32843
LAKE MICHIGAN TEMPERATURE EXCEEDED PLANT DESIGN BASIS LIMIT IN AUGUST 1988

As a result of questions posed by members of the ongoing NRC architect/engineer design inspection team, the licensee has determined that the water temperature of Lake Michigan, the plant's ultimate heat sink, exceeded the plant design basis lake temperature limit of 76°F for 22 days during August 1988.

The control room is normally cooled by an air conditioning system which utilizes non-safety related chillers. The safety related portion of the control room air conditioning system utilizes water from Lake Michigan as the cooling medium. This water would be supplied directly to the cooling coils following manual realignment. At an average lake temperature of 81°F that existed during the 22 day period in August 1988, the temperature inside the control room could have reached 110.4°F had the non-safety related chillers not functioned. At a temperature of 110.4°F, the lifetime of some instrumentation inside the control room, the solid state protection system, and the nuclear instrumentation, is estimated to be at 150 hours or 6.25 days. The impact of this shortened instrument life span on plant operation had not been evaluated.

At the time of this event, the plant Technical Specifications allowed continuous operation with control room temperatures up to 120°F. The Technical Specifications have since been revised such that continued operation with control room temperatures in excess of 95°F is not permitted.

Operation of the plant during the time period when lake temperature exceeded the design basis limit, without analysis indicating acceptable control room cooling could be maintained above this temperature limit, and without procedures to alert personnel of the situation, is considered as operation in an unanalyzed condition. The instrumentation was not adversely impacted by the high lake temperatures as the non-safety related chillers continued to function and maintain acceptable control room temperatures.

POWER REACTOR EVENT NUMBER: 32915
OVERPRESSURE PROTECTION OF THE COMPONENT COOLING WATER SYSTEM PIPING NOT IN ACCORDANCE WITH THE ANSI CODE REQUIREMENTS

Chapter 9.5 of the FSAR states: 'The relief valve on the component [cooling water] surge tank is sized to relieve the maximum flow rate of water that would enter the surge tank following a rupture of a reactor coolant thermal barrier cooling coil. The set pressure assures that the design pressure of the component cooling system is not exceeded.'

The piping design code at the Cook plant is B31.1. B31.1 states that an intercepting stop valve cannot be located between the source of pressure and the pressure relief device credited for protecting the pipe. In this instance, the pressure source is the ruptured thermal barrier; the pressure relief device is a safety relief valve on the surge tank. Contrary to the code requirement, there are manual valves maintained open between the two. These valves were not controlled in accordance with or exempted from B31.1.

An evaluation is being performed to determine the most effective method of establishing and maintaining the code requirement. A condition report has been written to initiate an investigation into this event and determine the appropriate preventative actions."

This condition was identified in response to an ongoing NRC architect/engineer design inspection.

[illegible]