CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9802120299 DOC.DATE: 98/01/30 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
FITZPATRICK,E. Indiana Michigan Power Co.
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Responds to NRC 971126 ltr re violations noted in insp on 970804-0907.Corrective actions:RWST level taps have been moved from discharge line to dedicated instrument area that is not subject to flow effects.

DISTRIBUTION CODE: IE01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 24
TITLE: General (50 Dkt)-Insp Rept/Notice of Violation Response

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-3 PD	1	1	HICKMAN,J	1	1
INTERNAL:	AEOD/SPD/RAB	1	1	~~AEOD/TTC~~	1	1
	DEDRO	1	1	FILE CENTER	1	1
	NRR/DRCH/HHFB	1	1	~~NRR/DRPM/PECB~~	1	1
	NRR/DRPM/PERB	1	1	NUDOCS-ABSTRACT	1	1
	OE DIR	1	1	OGC/HDS2	1	1
	RGN3 FILE 01	1	1			
EXTERNAL:	LITCO BRYCE,J H	1	1	NOAC	1	1
	NRC PDR	1	1	NUDOCS FULLTEXT	1	1

C A T E G O R Y 1 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 17 ENCL 17

Indiana Michigan
Power Company
500 Circle Drive
Buchanan, MI 49107 1395

INDIANA
MICHIGAN
POWER

January 30, 1998

AEP:NRC:1260G2

Docket Nos.: 50-315
50-316

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

Gentlemen:

Donald C. Cook Nuclear Plant Units 1 and 2
INTERIM RESPONSE TO DESIGN INSPECTION REPORT

From August 4, 1997, through September 12, 1997, the NRC conducted a design inspection at Cook Nuclear Plant. In the inspection report, dated November 26, 1997, we were requested to provide a schedule within sixty days, detailing our plans to complete the corrective actions for the open items that were listed in appendix A of the inspection report. The inspection report also noted that we were expected to evaluate the applicability of the results and specific findings of this inspection to other systems and components throughout the plant.

The attachment to this letter provides an interim response to the inspection report. It provides the status of the specific items listed in appendix A of the report. Specifically, it provides a listing of the open items, including a statement regarding our agreement with the characterization of the item, and a schedule for completing remaining work related to the specific item.

As a result of the AE design inspection findings, the NRC issued a confirmatory action letter (CAL) detailing specific items that must be resolved prior to restart of our units. The CAL also required us to perform a short term assessment to determine whether the types of problems identified exist in other safety systems, and if they impact operability. The CAL also noted that, in the long term, we are to evaluate our programs to ensure these types of problems do not recur.

The results of our short term program were provided to the NRC in our letter AEP:NRC:1260G3, dated December 2, 1997. Subsequently, additional information was provided to the NRC regarding reviews of calculations, the adequacy of our 10 CFR 50.59 program, and the development of our short term assessment program. This information was provided to the NRC in our letter AEP:NRC:1260G4, dated December 24, 1997. In addition to the reviews conducted as part of our short term program, we conducted additional reviews of 10 CFR 50.59 screenings and evaluations, and design changes. The results of these additional reviews were provided to the NRC in our letter AEP:NRC:1260G5, dated December 31, 1997. These reviews were discussed with the NRC during a meeting on January 8, 1998, and during that meeting, we also presented a description of our design bases reconstitution project. This project will integrate and enhance several existing programs such as the UFSAR revalidation

IE01

program, the design bases document reconstitution program, and the normal operations procedure upgrade program, and is considered to be part of the long term program the NRC discussed in the CAL. The information we presented at the January 8, 1998, meeting was docketed in our letter AEP:NRC:1260G6, dated January 8, 1998. The letters described herein are considered to be part of our response to the AE design inspection report, and are hereby incorporated by reference.

We are currently in the process of conducting additional reviews of previous 10 CFR 50.59 screenings and evaluations, as well as self-assessment of processes that had the potential to make changes to the plant outside of our 10 CFR 50.59 programs. We will supplement this letter with another letter providing our basis for considering the units ready for restart. Under separate correspondence, we will also define our long term program in accordance with the CAL, and will provide an additional submittal in connection with the NRC's 10 CFR 50.54(f) letter concerning the adequacy and availability of design bases information.

We take the NRC's overall findings and conclusions from the AE design inspection seriously, and are aggressively pursuing the specific issues as well as the programmatic issues associated with these findings.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

/vlb

Attachment

c: J. A. Abramson
A. B. Beach
MDEQ - DW & RPD
NRC Resident Inspector
J. R. Sampson

ATTACHMENT TO AEP:NRC:1260G2

INTERIM RESPONSE TO DESIGN INSPECTION REPORT

ITEM NO. URI-01

Issue

Apparent failure to recognize and evaluate all RWST level measurement errors and uncertainties. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2A).

Status

We agree with the characterization of this issue.

The condition has been corrected. RWST level taps have been moved from the discharge line to a dedicated instrument area that is not subject to flow effects. All other safety related level loops were reviewed for possible flow interaction. Three safety related systems were identified as having installations where potential flow induced errors may exist. These systems were: condensate storage tank (CST) level instruments, mid-loop level instruments, reactor vessel level indication system (RVLIS). Analysis of the CST indicated that a small flow induced bias could potentially exist for the CST instruments at peak auxiliary feedwater flow. No adverse impacts on system operability were identified related to this instrument configuration. The flow induced errors for the mid-loop level instruments were found to be negligible. All flow induced errors identified in the RVLIS had been previously accounted for in the original design. Instrument uncertainty calculations were revised to address flow-induced errors that include pipe entrance losses, pipe pressure drops and Bernoulli effects.

The instrument and control section engineering guide (EG-IC-004) has been revised to ensure the proper evaluation of level instrument loops in the future.

All actions have been completed.

ITEM NO. URI-02

Issue

Incorrect RWST level acceptance criterion specified in technical specification (T/S) surveillance procedure could have allowed the RWST level to be less than the T/S requirement. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2A2).

Status

We agree with the characterization of this issue.

The T/S surveillance procedure (1/2-OHP 4030.STP.030) has been revised to include the correct RWST level acceptance criterion. The procedures for removing the residual heat removal (RHR) loop from service, 01-OHP 4021.017.003 and 02-OHP 4021.017.003, have also been revised to ensure that the T/S limits are met. The instrument uncertainty program is being expanded as described in our letter AEP:NRC:1260G3, dated December 2, 1997, attachment 3.

All items are complete.

ITEM NO. URI-03 AND URI-04

Issues

Apparent failure to consider potential for vortexing and air entrainment when establishing the RWST Low-Low Level setpoint - 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2A3).

Apparent failure to take prompt corrective action after the 1993 SBICI finding regarding the potential for vortexing and air entrainment in the RWST, and after documented by the licensee in 1995 in CR 95-1015. 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Action) (E1.1.1.2A3).

Status

We agree with the characterization of these issues.

Calculation ENSM 970606 JJR, Revision 2, established vortex limits and provided input to ECP 1-2-I9-03, Calculation 3, which establishes the RWST low-low level setpoint. ENSM 970606 JJR, Revision 2, also determined that past operability was not compromised.

ECP 1-2-I9-03 has been revised to incorporate vortexing.

We agree that our corrective actions on this matter were not timely. Our corrective actions program was re-engineered in May 1996. One of the issues that prompted the revision of the program was an inconsistency in completing corrective actions for condition reports, self assessments, and audit findings in a timely manner. We continue to monitor the progress of the program through the corrective action program continuous improvement team (CAPCIT).

All actions are complete.

As a point of information, the site vice president is currently providing management oversight of a root cause analysis team that is determining the cause of continued poor performance in this area as documented in CR 97-3360. In addition, we will be participating in an industry project sponsored by EPRI plant support engineering subcommittee to develop guidance to optimize engineering activities in support of corrective action programs.

ITEM NO. URI-05

Issue

The uncertainty calculations for the containment and containment sump level instrumentation loops do not account for the impact on the post-accident containment water levels (ECPs 1-2-N3-01, 1-RPC-14, and 2-RPC-14), and do not consider the potential for vortexing, air entrainment, or net positive suction head requirements. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2B3).

Status

We agree with the characterization of this issue.

ECP 12-00-14 has been revised to include vortexing, air entrainment and NPSH requirements. Emergency operating procedures (EOPs) 1/2-OHP 4023.ES-1.3 have been revised to properly incorporate concerns.

Westinghouse has performed an evaluation of the peak containment pressure based on the current revision of ES-1.3. Westinghouse is currently verifying inputs to the evaluation and will complete its analysis. This will be completed prior to restart.

ITEM NO. URI-06

Issue

Apparent failure to demonstrate, using design bases documentation, that there was adequate containment recirculation sump water volume following a loss-of-coolant accident (LOCA). 10 CFR Part 50.46 (emergency core cooling system [ECCS] performance criteria) (E1.1.1.2C).

Status

We agree with the characterization of this issue.

New analyses for current operations have been completed. It has been confirmed that, for both postulated large and small break LOCAs an adequate volume of water would be resident in the containment structure, and that adequate communication exists in the containment subcompartment boundaries to ensure sufficient drainage to the containment recirculation sump. Additionally, it has been confirmed that, for current operation, for postulated LOCAs there would be no impact on the containment integrity analysis and sufficient water exists to maintain the core subcritical. These analyses included extensive reviews of loss of inventory to various containment volumes and concluded these losses could be tolerated.

In our letter AEP:NRC:0900K, dated October 8, 1997, a T/S change was submitted that increased minimum ice condenser ice weights and allowed consideration of water from ice melt in conjunction with water from the RWST, which increases sump water volume following a LOCA. T/S amendments 220 for unit 1, and 204 for unit 2, were approved by the NRC on January 2, 1998.

All actions have been completed.

ITEM NO. URI-07

Issue

Apparent failure to preclude a single active failure when performing changes to the plant, which is contrary to the assumptions in the updated final safety analysis report (UFSAR) and the design bases. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.1.1.2D).

Status

We agree with the characterization of this issue.

1/2-OHP 4023.ES-1.3 were revised to transfer both the east and west RHR pumps from the RWST to the recirculation sump. Subsequent to the transfer of the RHR pumps to the recirculation sump, RHR supply will be established to the safety injection (SI) pumps and the centrifugal charging pumps (CCPs). After the RHR supply is established to the SI pumps and CCPs, their RWST suction valves will be closed. The new switchover sequence will preclude the situation where a single active failure would cause a redundant train from being impacted. This revision was implemented on January 2, 1998.

The procedure, revision 2, which created the single failure vulnerability, occurred in 1992. Since that time, the review process for EOPs has been improved. Currently, there are multi-discipline reviews from technically qualified personnel. This process has been effective in identifying and correcting procedural problems. The nature of the incorporation of the unacceptable ECCS lineup in Revision 2 of OHP 4023.ES-1.3, "A Transfer to Cold Leg Recirculation", would suggest that this was an isolated case.

Policy 800000-POL-2300-04, "Definition and Use of Design Bases", was issued on December 8, 1997. Attachment 3 of this policy provides specific direction on the definition and use of the "single failure criteria". Within the text of attachment 3, a specific example of an "active failure" is cited as "the failure to continue to run". Training for personnel impacted by the issuance of this policy was completed in November 1997.

As described in attachment 4 to our letter AEP:NRC:1260G3, a contributing factor to the ECCS switchover procedure issue was the aspect of the design that crossties the ECCS system trains through a common recirculation suction source for the intermediate and high head injection pumps. We performed a review of other safety systems with crosstie capabilities, either between trains or between units, to provide reasonable assurance that single failure criteria have been appropriately considered and that procedures allowing the use of the crossties have been properly evaluated. Systems reviewed were auxiliary feedwater (AFW), essential service water (ESW), chemical and volume control (CVCS), component cooling water (CCW), and electrical distribution. No operability concerns were identified with the use of safety system crossties.

All actions are complete.

ITEM NO. URI-08

Issue

Apparent failure to maintain the 1/4" containment recirculation sump particulate retention requirement, which could allow the ECCS throttle valves and containment spray nozzles to become inoperable. 10 CFR Part 50.46 (ECCS performance criteria) (E1.1.1.2E2).

Status

We agree with the characterization of this issue.

In late 1996 (unit 2) and early 1997 (unit 1) vents in the sump top cover, which were not previously provided with particle retention screens, were incorrectly plugged to address the threat of foreign material. These vents have been reinstalled with appropriate particle retention screens.

All actions have been completed.

ITEM NO. URI-09

ECCS pump suction valves not leak-rate tested to confirm accident analysis assumption. 10 CFR Part 50, Appendix B, Criterion XI, (Test Control) (E1.1.1.2G4).

Status

We agree with the characterization of this issue.

Testing was performed on the valves that were not previously tested for potential leakage back to the RWST. The test results showed that the total leakage for these paths back to the RWST was well below the 10 gpm value in the UFSAR. The leakages for each of the valves tested is as follows:

Unit 1 and Unit 2	IMO-910 = 0 gpm
Unit 1 and Unit 2	IMO-911 = 0 gpm
Unit 1 and Unit 2	IMO-261 = 0 gpm
Unit 1	RH-130 = 0 gpm
Unit 2	RH-130 = 0.48 gpm.

All corrective actions are completed. These valves will be included in the in-service testing program procedures by June 1, 1998.

ITEM NO. URI-010

Issue

Apparent failure to demonstrate, using design bases documentation, that the plant could perform a T/S 3.0.3 shutdown in 36 hours to 200° F, using one CCW train and design bases assumptions. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.2.1.2B).

Status

We agree with the characterization of this issue.

The original single train 36 hour cooldown analysis allowed the component cooling water supply temperature to reach 120° F. Operating procedures acknowledge that CCW supply temperature could elevate to 120° F during some cooldown evolutions. Contrary to this, the design bases CCW supply temperature as detailed in the UFSAR was 95° F. A cooldown analysis had not been completed to show that a 36 hour single train cooldown could be achieved with the CCW supply temperature limited to 95° F. In addition, the analysis of record had incorrectly modeled the CCW heat exchanger as counterflow instead of TEMA type E. Due to these errors, we could not demonstrate the plant could perform a 36 hour shutdown using design bases assumptions.

Evaluation and analysis were completed to allow operation of the CCW system with a maximum supply temperature of 120° F. This review was completed under design change package (DCP) 12-DCP-855. Subsequent to the review, a change to the UFSAR was initiated to allow the CCW supply temperature to reach 120° F during emergency cooldown and post-accident conditions. The heat exchanger model was corrected.

A new cooldown analysis has been completed that demonstrates the capability to perform a 36 hour cooldown using one CCW train and design bases assumptions.

All actions are complete.

<u>ITEM NO. URI-011</u>

<u>Issue</u>

Apparent failure to correctly translate the as-built design of the CCW heat exchanger into safety related calculations and analyses. 10 CFR 50, Appendix B, Criterion III (Design Control) (E1.2.1.2E2).

<u>Status</u>

We agree with the characterization of this issue.

Prior to the design inspection, the CCW heat exchanger was incorrectly modeled by both us and Westinghouse as a counterflow heat exchanger. Contrary to this, the CCW heat exchanger is a TEMA type E heat exchanger. This condition resulted from an error in the modeling of the heat exchanger during original plant design, which was carried through to more recent calculations.

The unit 1 30% steam generator tube plugging analysis has been corrected. In our letter AEP:NRC:1223M, dated September 10, 1997, we requested that the NRC suspend review of the unit 2 uprate.

We sent a team to the Westinghouse offices to review the analyses of record for both units. The team determined that Westinghouse had modeled other safety related heat exchangers properly. It was also confirmed that Holtec International, who performed the analysis of record for the spent fuel pool (SFP) system, correctly modeled the SFP heat exchanger. We then reviewed internal calculations to determine if safety related heat exchangers had been modeled properly. This review concluded that three heat exchangers were incorrectly modeled in our analyses. Specifically, the CCW heat exchanger, the diesel generator jacket water cooler, and diesel generator lube oil cooler were also modeled as counterflow heat exchangers, when in reality they are TEMA-E design. Reviews were performed to ensure that these heat exchanger modeling errors did not change the conclusions of these calculations. These internal calculations will be revised to ensure that the heat exchangers are modeled properly by March 31, 1998.

<u>ITEM NO. URI-012</u>

<u>Issue</u>

Apparent lack of documentation to demonstrate that the control room equipment was qualified at worst case operating temperatures in the

control room. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.2.1.2E2).

Status

We agree with the characterization of this issue.

This issue addresses the operation of the plant with lake water temperatures above 76° F.

Restrictions have been placed on plant operation such that the plant will not be operated with lake temperatures above 76° F. These restrictions will remain in place until all analyses, 10 CFR 50.59 safety evaluations, and the necessary approvals are complete to change the temperature.

Procedure OHI-4013 has been revised to direct the use of procedure 12-OHP 4021.019.001 if the circulating water inlet temperature approaches 74.5° F.

Procedure 12-OHP-4021.019.001 has been revised to require that if the floating one hour average circulating water temperature is greater than 75° F, unit shutdown shall be initiated in both units. This procedure requires that if the circulating water inlet temperature is greater than 75.5° F, as indicated by instrumentation, the ESW system is to be considered inoperable.

All actions are complete.

ITEM NO. URI-013

Issue

Apparent failure to analyze all potential failure modes of the instrument air system that could render redundant trains of safety related equipment inoperable. 10 CFR Part 50, Appendix B, Criterion III (Design Control) (E1.2.1.2F).

Status

We agree with the characterization of this issue.

DCP 854 added safety valves to provide overpressure protection for the control air headers in the turbine building, auxiliary building, and containment. Although this modification was completed, we became concerned that pressure accumulation in the piping to the containment air header safety valves could be excessive and would not ensure that all components served by the header would be adequately protected from overpressure.

An additional modification to correct this concern is in progress and will be completed prior to entering mode 4.

ITEM NO. URI-014

Issue

Operation of the plant with CCW-supplied flows to safety related and important to safety components contrary to the values stated in the UFSAR. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.2.1.2G).

Status

We agree with the characterization of this issue.

A flow range was established for each component cooled by the CCW system. Specifically, a minimum flow rate was established (or verified to be correct) to ensure that each component received sufficient cooling and a maximum flow rate was established to protect each component from damaging flow rates. Current flow rates have been confirmed within the new limits and these limits will be incorporated into the flow balance procedure prior to its next use.

With respect to the specific issue of the CCW flow to the RCP thermal barrier heat exchanger, the minimum flow requirement was decreased from 35 gpm to 24 gpm. A 10 CFR 50.59 evaluation has been completed and the new flow rates will be included in the next UFSAR update.

ITEM NO. URI-015

Issue

Apparent failure to establish controls to prevent potential operation of the CCW system with the CCW heat exchangers above the maximum fouling factor value established by the GL 89-13 testing program. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.2.1.2H).

Status

We agree with the characterization of this issue.

Maximum allowable CCW heat exchanger fouling within the generic letter (GL) 89-13 service water program has been set at the design maximum fouling value. This acceptance limit does not provide margin to ensure that heat exchanger fouling stays below the maximum limit during the next operating cycle. As a point of information, since the GL 89-13 program has been in place, the CCW heat exchangers have been visually inspected during all except one refueling outage, where inspection was not necessary for one heat exchanger based on test results indicating zero fouling. When heat exchangers were opened for inspection, they were cleaned to ensure they entered the successive operating cycle with minimum fouling.

The procedure used to collect data and measure CCW heat exchanger fouling factor will be revised prior to its next use, unit 1 refueling outage in 1999, to include a margin between the acceptance limit and the maximum fouling limit. This provides assurance that the maximum fouling will not be exceeded during the next operating cycle. Instrumentation uncertainties will be considered for inclusion in determination of the test acceptance criteria.

ITEM NO. URI-016

Issue

Performance testing of the emergency diesel generator (EDG) heat exchangers was not able to detect degradation, as required by the

licensee's GL 89-13 testing program. 10 CFR Part 50, Appendix B, Criterion XI, (Test Control) (E1.2.1.2H).

Status

We agree with the characterization of this issue.

The method currently used to trend EDG heat exchanger degradation is ineffective because heat exchanger outlet temperature is controlled by automatic valves that regulate temperature. Past internal inspection of these heat exchangers has demonstrated that these heat exchangers are in good condition, and provides assurance of current operability. Specifically, the EDG aftercoolers, which are the first of three EDG heat exchangers arranged in series, are inspected every other refueling outage, and have been observed to be clean, with the exception of some minor sediment that collects in the bottom of the heat exchangers when isolated. Inspection of the other two heat exchangers, though not on a predetermined periodicity, has produced similar results. The method to trend EDG heat exchanger degradation is under review to determine if a methodology can be devised that provides more meaningful results.

The GL 89-13 program will be revised by June 30, 1998, to either incorporate a more meaningful test method, or include a better defined schedule for inspecting the internals of these heat exchangers.

ITEM NO. URI-017

Issue

Inadequate justification to return the unit 2 250 VDC battery train CD to an operable status (E1.3.1.2).

Status

We agree with the characterization of this issue.

The two issues identified with this finding were the operability determination made on cell 34 of the unit 2 train CD battery, and the length of time the cell was individually charged. The operability determination was based on T/S surveillance 4.8.2.3.2.a, which does not specifically require the battery cell to be on float charge when reading the voltage. The 51 day equalize charge on cell 34 was excessive.

These issues are being addressed by revising T/S surveillances for the 250 VDC battery to the Westinghouse standard T/S surveillance for batteries. In concert with this effort temporary procedure 12-IHP 5021.EMP.009 will be revised to limit the amount of time an individual cell can be on equalize charge when making operability decisions, and when performing corrective maintenance activities. The procedure revision will be complete by March 2, 1998.

As a point of clarification, cell 34 was carefully inspected and no mineral crystallization was identified.

In support of our initial operability determination, cell 34 was removed from service and capacity tested. The results showed capacity of 116.9% of rated capacity. This test was performed

several months after the cell was removed from service and placed on float charge.

ITEM NO. URI-018

Issue

Apparent failure to maintain adequate design and procedural controls that allowed the plant to operate in modes 5 and 6 without an adequate volume of borated water in the other unit's RWST in order to meet appendix R requirements. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.2A).

Status

We agree with the characterization of this issue.

Instances were identified where the RWST level fell below the 87,000 gal required for appendix R. The T/Ss require that a fire watch be posted in applicable locations if the minimum water level is not restored within 7 days. There were five instances where the 7 days was exceeded. In each of the five instances there are records that indicate a fire watch was posted in the applicable places. The firewatch is allowed as an alternative for shutdown capability per T/Ss. Therefore, the T/Ss were not violated.

During the AE inspection two calculations were identified on the subject of the RWST minimum required water level to support Appendix R (ECP 1-2-I9-03 and TH-90-02). Although the results of TH-90-02 were acceptable, subsequent to the AE inspection, a new calculation ENSM 971001 CV was developed that better documents the methodology and incorporates additional conservatisms. This new calculation requires 131,921 gallons in the RWST.

Calculation TH 90-02 has been superseded. Calculation 1-2-I9-03 has been revised to remove the reference to the old calculation and reference ENSM 971001 CV.

The following procedures have been revised to incorporate the more conservative results of ENSM 971001 CV:

1/2-OHP 4030.STP.030
1/2-OHP 4021.018.005
1/2-OHP 4021.018.008
PMP-4100

The only actions that remain are to revise the descriptions in the Appendix R design bases document and the fire protection program manual. These revisions will be complete by February 15, 1998.

ITEM NO. URI-019

Issue

Apparent failure to perform instrument uncertainty calculation for the CCW heat exchanger outlet temperature loop uncertainty. 10 CFR Part 50 Appendix B, Criterion III, (Design Control) (E1.3.2.2B5).

Status

We agree with the characterization of this issue.

Instrument uncertainty calculations have been completed for the CCW heat exchanger outlet temperature and documented in ECPs 1-WSI-04 and 2-WSI-04. Operating procedures have been modified to provide a margin to CCW system temperature limits based on the calculated instrument uncertainty.

All actions have been completed.

ITEM NO. URI-020

Issue

Apparent failure to perform instrument uncertainty calculation for the ESW intake temperature loop uncertainty. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.2B6).

Status

We agree with the characterization of this issue.

ECP-12-WO-01 was complete and determined the uncertainty of the circulating water inlet temperature instrumentation. This analysis showed the uncertainties associated with this instrumentation can affect operation at elevated lake temperatures.

Therefore, interim measures (Engineering Technical Direction Memo 97-095), have been put in place to benchmark diverse installed instrumentation when the indicated circulating water temperature reaches the uncertainty limit. The benchmarking entails cross calibrating the plant process computer temperature indication with traceable M&TE. This allows the use of this installed instrumentation with a greatly reduced error margin for the circulating water inlet temperature limit.

12-OHP-4021.019.001 has been revised to perform the benchmarking prior to 72° F, and to require that if the floating one hour average circulating water temperature is greater than 75° F, unit shutdown shall be initiated. Further, at 75.5° F, the ESW system is to be declared inoperable.

A DCP 174 for increasing allowable lake temperature has been initiated. The calculations for indication uncertainty will be reviewed and new direction provided, pending final implementation of this design change.

ITEM NO. URI-021

Issue

Apparent failure to perform instrument uncertainty calculation for the control room temperature loop uncertainty. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.2B7).

Status

We agree with the characterization of this issue.

ECP-1-B1-07 and ECP-2-B1-07 have been revised and incorporated into 1/2-OHP 4030.STP.030.

All actions were completed October 24, 1997.

ITEM NO. URI-022

Issue

Apparent programmatic deficiency with the Setpoint Control Program concerning the ability to perform and account for instrument uncertainties. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.3.2.3).

Status

We agree with the characterization of this issue.

The results of uncertainty calculations were not effectively incorporated into procedures. The instrument uncertainty program is being expanded as described in AEP:NRC:1260G3, attachment 3.

All actions are to be completed by the end of December 1998.

ITEM NO. URI-023

Issue

Performing changes to safety related procedures without apparent proper review and/or approval, contrary to the provisions of T/S 6.5.3.1 and 10 CFR Part 50.59 requirements. (E1.5.2A)

Status

We agree with the characterization of this issue.

There were three specific procedures discussed.

- 01-OHP.4023-ES1.3. A safety evaluation dated October 29, 1997, was performed for 01-OHP 4023-ES1.3 Revision 5 to address the containment water level permissive and operator reliance on containment sump water level instrumentation. The safety screening and safety evaluation were performed prior to procedure approval.

- 12-OHP.4021.019.001. A 10 CFR 50.59 screening dated August 21, 1997, was performed on the procedure revision to specify that the circulating water inlet temperature of 76° F is the design bases temperature limit. The safety evaluation screening was completed prior to procedure approval.

Procedure 12-OHP-4021.019.001 has also been revised to require that if the floating one hour average circulating water temperature is greater than 75° F unit shutdown shall be initiated in both units. This procedure also requires that if the circulating water inlet temperature is greater than 75.5° F, as indicated by instrumentation, the ESW system is to be considered inoperable.

Procedure OHI-4013 has been revised to direct the use of procedure 12-OHP 4021.019.001 if the circulating water inlet temperature approaches 74.5° F.

- 2-OHP.4021.016.003. A 10 CFR 50.59 safety evaluation dated October 17, 1997, was performed to establish the appropriate CCW operating temperature limit for rapid RCS cooldown. This safety evaluation was completed prior to procedure approval.

To prevent temporary non-intent changes from being made to procedures without proper review, PMSO.179 has been issued and PMI-2010 has been revised to prohibit the revision of procedures without an approved 10 CFR 50.59 screening being completed.

ITEM NO. URI-024

Issue

Apparent failure to maintain proper design control regarding industry standards and codes. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.5.2C).

Status

We agree with the characterization of this issue.

There were four instances discussed.

- **Conflict in piping specification and classification of CCW piping inside containment.** Piping specification ES-PIPE-1013-QCN Class M-12 has been revised to reflect the correct design pressure of 2485 psig and design temperature of 550° F. A thermal analysis has been performed for the pipe supports associated with the sections of piping in question.

- **Deviation from B31.1 code requirement for overpressure protection in the CCW system.** Four CCW valves in each unit were identified as intervening stop valves. These valves have been locked open and are now included in procedure OHP 4030.STP.035, "Controlled Valve Position Logging". This will ensure that the position of these valves is checked on a monthly basis. These valves are no longer considered intervening stop valves.

- **RHR low pressure interlock during mode 4.** In our letter AEP:NRC:1278, dated September 19, 1997, a T/S change to delete this interlock was requested. Amendments 219 for unit 1 and 203 for unit 2 were approved on December 10, 1997.

- **Overpressure protection for the CCW heat exchangers.** Overpressure protection is provided for the tube side. No overpressure protection is provided for the shell side. An evaluation was performed that concluded that there are no CCW shell side pressure increases resulting from increased CCW and ESW temperatures or external sources that could cause the CCW internal pressure to exceed the design pressure of 150 psi. A 10 CFR 50.59 evaluation was performed for revising the UFSAR to include a footnote to Table 9.5-1, clarifying that the CCW heat exchangers were manufactured to ASME B&PV Code, Section VIII, 1968 edition, but installed in accordance with USAS B31.1, 1967 edition.

ITEM NO. URI-025

Issue

Apparent failure to maintain adequate design control and follow established procedures for equipment abandoned in place. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.5.2D).

Status

We agree with the characterization of this issue.

A review of each plant system will be led by the system engineer for the purpose of identifying equipment that is no longer used or useful. Equipment that falls into this category and has not been abandoned in place through the design change process shall be processed through the project acceptance/design change process. This review will also confirm that equipment previously abandoned in place has been appropriately tagged as abandoned in place.

These reviews will be completed by May 1, 1998.

ITEM NO. URI-026

Issue

Apparent failure to maintain adequate drawing control that has the potential to impact plant operating procedures, and maintenance activities that use drawings. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (E1.5.2E).

Status

- Item 4(a): CCW flow rates are shown inconsistently on drawing 1 OP-1-5135, revision 33. This drawing was corrected by removing flow rate information for all items other than pumps. This was done because the flow rates are defined in other documents.

We do not have a requirement to show branch line flow rates on our flow diagrams. Design guideline no. 0202-1(n) has been updated to ensure that flow rates will be consistently eliminated from flow diagrams as sections of the flow diagrams are revised.

- Item 4(b): A horizontal perforated plate (grating) was shown on two section views on masonry drawing 1-2-3178-14.

The grating was removed via the 1979 RFC DC-12-2361. At that time, the primary steel drawing 12-3902, revision 4, was marked up to show that the grating should be removed and the removal was noted in the revision box. Because masonry drawing 1-2-3178-14 referenced back to the primary steel drawing 12-3902 concerning this grating removal, the grating would not have been reinstalled based on the masonry drawing.

The primary steel drawing 1-2-3178 was corrected with information provided through CR 97-2344.

- Item 4(c): A note concerning the temporary wiring of a pressure switch associated with the spare CCW pump had some incorrect wire numbers.

In 1983 and 1984 the subject drawings were enhanced to clarify some confusion with wire identification. The enhancement caused the subject note to be misleading.

Drawings 1-93011, 1-93048, 2-93011 and 2-93048 were corrected with information provided through CR 97-2304.

All items are complete.

<u>ITEM NO. URI-027</u>

<u>Issue</u>

Apparent failure to adequately translate design bases assumptions into Plant Procedure OHP 4021.001.004, Plant Cooldown from Hot Standby to Cold Shutdown. 10 CFR Part 50, Appendix B, Criterion III, (Design Control) (1.5.2F).

<u>Status</u>

We agree with the characterization of this issue.

1/2-OHP 4021.001.004 and 1/2-OHP 4021.017.002 have been revised to provide the operator with the guidance necessary for safe operation of the CCW system. 1/2-OHP 4021.001.004 now have a precaution to verify that the CCW heat exchanger flow is in the range of 8,000-9,000 gpm. 1/2-OHP 4021.017.002 now have a precaution to maintain CCW flow through the RHR heat exchanger between 5,000 and 5,500 gpm.

All items are complete.

<u>ITEM NO. URI-028</u>

<u>Issue</u>

The team determined that the licensee had operated the plant above the maximum UHS temperature limit without performing a 10 CFR 50.59 evaluation, which also potentially created a unreviewed safety question (USQ) regarding a reduction of safety margin as defined in the T/S bases for the control room emergency ventilation temperature limit of 95° F (E.1.2.2E[1]).

<u>Status</u>

We agree with the characterization of this issue.

Procedure OHI-4013 has been revised to direct the use of procedure 12-OHP 4021.019.001 if the circulating water inlet temperature approaches 74.5° F.

Procedure 12-OHP-4021.019.001 has been revised to require that if the floating one hour average circulating water temperature is greater than 75° F unit shutdown shall be initiated in both units. This procedure requires that if the circulating water inlet temperature is greater than 75.5° F, as indicated by instrumentation, the ESW system is to be considered inoperable. These restrictions will remain in place until all analyses and 10 CFR 50.59 safety evaluations and necessary approvals are complete to change the temperature.

All actions are complete.

ITEM NO. URI-029

Issue

The licensee's 10 CFR 50.59 evaluation, dated March 11, 1996, and March 20, 1996, respectively, that was performed to evaluate the consequences of the 1996 unit 2 full-core offload, failed to recognize the significance that the CCW heat exchanger could not perform its function under the design bases assumptions that were stated in the UFSAR and other licensing documentation. In addition, the licensee also failed to address UFSAR Section 9.4, regarding the criteria for spent fuel pool (SFP) cooling time-to-boil events and subsequently failed to identify that the conclusions reached in the evaluation would have potentially reduced the time-to-boiling in the SFP, given the assumptions stated in the SFP loading calculation and in the UFSAR. Reduction in the time to boiling criteria potentially impacts the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the safety analysis report and is an apparent USQ (E1.2.1.2C 1).

Additionally, this safety evaluation failed to identify that a dual CCW/ESW train outage with one unit in refueling and the other unit at power, was contrary to the UFSAR assumptions and placed the plant in an unanalyzed condition and outside of the design bases. This condition also potentially increased the probability of occurrence or the consequences of an accident or malfunction previously evaluated in the safety analysis report, creating the potential for a USQ (E.1.2.1.2D).

Status

We partially agree with the characterization of this issue.

The March 1996 cycle specific analysis determined the peak SFP bulk transient temperature to be in compliance with the design bases limit discussed in section 9.4 of the UFSAR, which fulfilled our commitment made to the NRC in our letter AEP:NRC:1202A, dated February 1, 1996. The analysis was conservative because it assumed fuel movement began 100 hours after shutdown, even though fuel movement was prohibited for the first 168 hours by T/Ss. This overconservatism caused the analysis to incorrectly demonstrate a need for an administrative CCW temperature limit below the allowable design bases CCW temperature. However, the 1997 cycle specific analyses demonstrate, by bounding previous offloads, that the peak SFP bulk transient temperature design bases would not have been violated during the 1996 refueling outage, even without the administrative CCW temperature limits conservatively imposed. Therefore, the CCW heat exchanger was capable of performing its design function.

The SFP analyses of record (approved by SER dated January 14, 1993) indicated that the time to boil basis would be violated for a bounding (full SFP) single SFP cooling train full core discharge proceeding at the minimum T/S subcriticality hold time. Therefore, this parameter should have been evaluated at the partial SFP loading prior to the unit 2 full core discharge in 1996. Failure to evaluate the time to boil could have potentially allowed the SFP to be placed in a condition outside of its design bases. However,

the 1997 cycle specific analyses demonstrate, by bounding previous offloads, that the time to boil bases was not violated during the 1996 unit 2 refueling outage.

Considering the deficiencies described below in the review of the dual train CCW outage for unit 2 in 1996, this was a condition outside the design bases of the plant. The contingency actions for recovering SFP cooling during the unit 2 1996 refueling outage were such that the plant was capable of restoring unit 2 CCW within 1.5 hours, assuming no errors or environmental effects. Greater than three hours would have been available before reaching the peak SFP bulk transient temperature following an unexpected loss of CCW from unit 1. The required time to implement the contingency actions was not adequately reviewed to fully credit manual restoration of CCW flow from unit 2 following an unexpected loss of CCW flow from unit 1.

Dual train CCW outages will not be scheduled during refueling outages until analyses are performed to ensure the SFP design bases will not be violated and adequate reviews are performed to ensure that an USQ does not exist.

As a point of information, Holtec International is currently performing a SFP storage capacity bounding analysis to calculate the peak SFP bulk transient temperature and time to boil for a full core discharge to the SFP, assuming operation of a single train of SFP cooling at UFSAR design bases minimum flows and maximum CCW temperatures. This analysis, when complete, is expected to eliminate the need for cycle specific peak SFP temperature (as committed in our letter AEP:NRC:1202A, dated February 1, 1996) and time to boil evaluations.

<u>ITEM NO. URI-030</u>

<u>Issue</u>

The 10 CFR 50.59 safety evaluation that was performed by the licensee for revision 2, dated June 1992, to EOP OHP 4023.ES-1.3 was not effective in identifying that the revision was creating a single failure vulnerability that could render one RHR pump and both safety related trains of SI and CC pumps inoperable. Subsequent procedural revisions (revisions 3 and 4, dated January 1996, and January 1997, respectively) failed to identify the single failure vulnerability (E1.1.1.2D).

<u>Status</u>

We agree with the characterization of this issue.

1/2-OHP 4023.ES-1.3 were revised to transfer both the east and west RHR pumps from the RWST to the recirculation sump. Subsequent to the transfer of the RHR pumps to the recirculation sump, RHR supply will be established to the SI pumps and the CCPs. After the RHR supply is established to the SI pumps and CCPs, their RWST suction valves will be closed. The new switchover sequence will preclude the situation where a single active failure would cause a redundant train from being impacted. This revision was implemented on January 2, 1998.

The procedure revision (revision 2), which created the single failure vulnerability, occurred in 1992. Since that time, the

review process for EOPs has been improved. Currently, there are multi-discipline reviews from technically qualified personnel. This process has been effective in identifying and correcting procedural problems. The nature of the incorporation of the unacceptable ECCS lineup in revision 2 of OHP 4023.ES-1.3 would suggest that this was an isolated case. As this event appears to be an isolated event and there have been improvements implemented in the EOP review process, no additional preventative actions were warranted.

As described in attachment 4 to our letter AEP:NRC:1260G3, a contributing factor to the ECCS switchover procedure issue was the aspect of the design that crossties the ECCS system trains through a common recirculation suction source for the intermediate and high head injection pumps. We performed a review of other safety systems with crosstie capabilities, either between trains or between units, to provide reasonable assurance that single failure criteria have been appropriately considered and that procedures allowing the use of the crossties have been properly evaluated. Systems reviewed were AFW, ESW, CVCS, CCW and electrical distribution. No operability concerns were identified with the use of safety system crossties.

Policy 800000-POL-2300-04, "Definition and Use of Design Bases", was issued on December 8, 1997. Attachment 3 of this policy provides specific direction regarding the definition and use of the "single failure criteria", which is defined in the body of the policy. Within the text of attachment 3 of this policy, a specific example of an "active failure" is cited as "the failure to continue to run". Training regarding this policy was initiated on November 4, 1997. The initial training for personnel impacted by the issuance of this policy was completed in November 1997.

All actions are complete.

ITEM NO. URI-031

Issue

In 1996 (unit 2) and 1997 (unit 1), the licensee filled in the containment recirculation sump roof vent holes without performing a 10 CFR 50.59 evaluation. The licensee stated that the holes were sealed because they were not indicated on any plant design drawings and because they could not locate any design requirement for consideration of the vent holes (E.1.1.1.2E 1).

Status

We agree with the characterization of this issue.

The holes were reopened and FME protection provided.

To address the concern for changes being made to design bases information without a 10 CFR 50.59 evaluation, Policy 800000-POL-2300-04, "Definition and Use of Design Bases", was issued on December 8, 1997. This policy defines design bases as information that identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.

Training regarding this policy was completed in November 1997.

We acknowledge that the NRC has expressed concerns in the CAL followup inspection regarding the process followed in reinstalling the vent holes.

ITEM NO. URI-032

Issue

Procedure 01/02-OHP 4021.016.003, "Operation of the Component Cooling Water System During Reactor Startup and Normal Operation", was revised to remove the statement, "...allowing three hours at CCW temperatures of 120° F." The licensee implemented the revision under the auspices of a non-intent procedural change, as allowed by T/S 6.5.3.1. However, the team determined that this revision constituted a change to the intent of the procedure, contrary to the conclusion reached by the licensee (E1.2.1.2C).

Status

We agree with the characterization of this issue.

A 10 CFR 50.59 safety evaluation dated October 17, 1997, was performed to establish the appropriate CCW temperature limit.

PMSO.179 was issued and PMI-2010 has been revised to prohibit the revision of procedures without an approved 10 CFR 50.59 screening being completed.

ITEM NO. URI-033

Issue

The licensee consistently operated the plant with less than the UFSAR-specified CCW flows through the RCP thermal barriers and without performing a 10 CFR 50.59 evaluation. This issue is a concern because the CCW system has operated above the maximum design bases CCW temperature limit of 95° F, and was allowed to operate at temperatures up to 120° F, without evaluating the impact on the RCP thermal barriers with the reduced CCW flows (E1.2.1.2G).

Status

We agree with the characterization of this issue.

A flow range was established for each component cooled by the CCW system. Specifically, a minimum flow rate was established (or verified to be correct) to ensure that each component received sufficient cooling and a maximum flow rate was established to protect each component from damaging flow rates. A 10 CFR 50.59 evaluation was completed and the minimum flow requirement for the thermal barriers was decreased from 35 gpm to 24 gpm. The UFSAR will be revised in the next scheduled update.

Policy 800000-POL-2300-04 "Definition and Use of Design Bases", was issued on December 8, 1997.

Training regarding this policy was completed in November 1977.

ITEM NO. URI-034

Issue

The licensee identified that they have operated the plant without overpressure protection to the RHR system, contrary to the assumptions stated in UFSAR Chapter 9.3. This change to the design bases feature that provides over pressure protection to the RHR system was to defeat the interlocks associated with ICM-129 and IMO-128, RHR hot leg inlet isolation valves, when operating in mode 4. However, the change was performed without performing a 10 CFR 50.59 evaluation (E1.5.2C 3).

Status

We partially agree with the characterization of this issue.

The RHR system suction valves from the RCS are interlocked through separate channels of RCS pressure signals to provide automatic closure in the event RCS pressure exceeds RHR system design pressure. During shutdown conditions, these interlocks are effectively defeated by removing power to the valves to prevent a loss of RHR cooling due to inadvertent valve closure. The interlocks are unnecessary in this configuration as overpressure protection system (LTOP). While this configuration improved the reliability of the RHR system during shutdown conditions, and the surveillances of the interlocks were performed in accordance with T/Ss, the removal of power to the valves was not in compliance with T/S requirements.

In our letter AEP:NRC:1278, dated September 19, 1997, we submitted a T/S change request to delete the RHR interlock requirement. Amendment 219 for unit 1, and 203 for unit 2, was approved on December 10, 1997.

Policy 800000-POL-2300-04, "Definition and Use of Design Bases", was issued on December 8, 1997. Training regarding this policy was completed in November 1997.

All actions are complete.

ITEM IFI-001

Issue

UFSAR and T/S inconsistencies with RWST volume (E1.4.2B).

Status

We agree with the characterization of this issue.

The necessary analyses to redefine the required RWST water volume as well as address instrumentation issues of the instruments used have been done to support the revision of the emergency operating procedure, which controls the transition from the injection phase to the recirculation phase. The revised analyses were an integral part of the safety review done for the procedure revision.

The UFSAR will be changed to reflect this redefinition during the next regular update.

ITEM IFI-002

Issue

The RWST and the containment water level instrumentation allowable outage times (AOTs) should appropriately be governed by consistent AOT requirements (E1.4.2D).

Status

We partially agree with the characterization of this issue.

The RWST level instrumentation (ILS) and containment water level instrumentation (NLI) instruments provide no auto start function, which was recognized in the inspection report. The ILS instrumentation provides indication only, which is used to initiate the operator's actions that manually switch the unit from injection to recirculation. The NLI instruments provide indication only, which would be used in the emergency to initiate actions in response to events beyond the design bases.

The containment water level instrumentation is used to indicate an unmonitored diversion of water from containment during injection that is outside the design bases. Therefore, the injection containment water level instrumentation is not used to satisfy an engineered safety function as stated in the inspection report. These instruments are currently covered by the post-accident instrumentation T/Ss. This is considered appropriate and no change is anticipated.

The RWST instrumentation used in conjunction with the analysis provides the necessary indication to control changeover from injection to recirculation. There are two channels of instruments providing the indication the operator would use to initiate the required actions. If only one channel was available, the operator would still initiate the required actions. Because these instruments have no auto start function and the operator would initiate actions from a single channel, if one was out of service, the coverage of these instruments by the post-accident instrumentation T/S is considered appropriate.

The loss of both channels of RWST level instruments, resulting in the loss of all approved indications, would be covered by T/S 3.0.3 requiring prompt actions to shut down the unit. A reworded post-accident instrumentation T/S 3.3.3.8 will be submitted to clarify that the loss of both RWST level instruments places the unit in T/S 3.0.3.

The T/S change submital will be provided by April 30, 1998.