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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 25 1981

MEMORANDUM FOR: Stephen Hanauer, Director
Division of Human Factors Safety (DHFS)

FROM: Thomas E. Murley, Director
Division of Safety Technology (DST)

SUBJECT: DIESEL GENERATOR LOADING PROBLEMS RELATED TO SIS RESET ON LOSS OF OFFSITE POWER

REFERENCE: Memorandum dated 11/5/80: S. Hanauer to F. Schroeder, Acting Director, DST; Subject: SAFETY INJECTION SIGNAL RESET

The Division of Safety Technology described in a September 30, 1980, memorandum to you a potential problem at the Point Beach reactors whereby a Safety Injection System (SIS) reset could result in loss of capability to actuate containment spray automatically. The memo also suggested that in other plants other safety functions could be adversely affected by SIS reset and recommended specific action.

Later, the Operating Experience Evaluation Branch (OEEB) began a review of SIS reset from a different aspect due to a recent Preliminary Notification (PNO) identifying a problem with diesel generator loading after SIS reset and loss of offsite power. This problem is the same as that raised as Technical Issue #4 in NUREG-0138 and was considered by the staff to be resolved on all operating plants. Therefore, as a result of this PNO and another relatively recent similar report, OEEB performed a review of the history of the problem and the agency's actions to resolve it. In spite of these actions the problem has persisted at least at some plants, indicating a deficiency in the process. Though it is difficult to identify at what point the process broke down, it is clear that our success was incomplete.

During our review we determined that the DHFS is presently reviewing the problem on new plants and has plans to perform such a review on operating plants in the future. Inasmuch as DHFS is addressing this problem (see reference memo) further action on the part of DST is not planned. However, for your information, we have enclosed our compilation of the history of the problem and the numerous actions taken in attempted resolution. The enclosure also identifies the concerns and the actions we believe could bring a final resolution.

All

In view of the history of this problem and seeming lack of a final resolution, we urge the NIFS in its review to consider the need to reach and document a satisfactory completion of the agency's commitments related to Technical Issue No. 4, NUREG-0138.

Original signed by
Thomas F. Murley

Thomas F. Murley, Director
Division of Safety Technology

Enclosures:
1. SIS Reset Action as it Relates to Diesel Generator Problems
2. List of Related Documents

cc: H. Denton
V. Stello
T. Murley
D. Eisenhut
H. Thornburg
N. Moseley
E. Jordan
W. Gammill
D. Crutchfield
D. Ziemann
F. Nolan
S. Bryan
D. Beckham
M. Ernst
E. Adensam
C. Michelson
H. Faulkner
D. Ross
F. Schroeder
R. Baer
A. Thadani
F. Rosa

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DATE	2/ /81 psm	2/ /81	2/ /81	2/ /81			

SIS RESET ACTION AS IT RELATES TO DIESEL GENERATOR PROBLEMS

In a recent Preliminary Notification, PNO-V: 80-68A dated September 23, 1980, on San Onofre Unit No. 1, it was reported that the licensee had, during testing, identified a problem with the design of the diesel generator sequencing circuitry. The report indicated the problem would occur when a safety injection signal is blocked in accordance with the LOCA procedure following safety injection initiation. Under these circumstances a subsequent loss of off-site power would not produce automatic resequencing of safety injection loads onto the diesel generator supplied buses.

This same concern was presented November 3, 1976, as Technical Issue No. 4 "Loss of Offsite Power Subsequent to Manual Safety Injection Reset Following a LOCA" in NUREG-0138 and is quoted as follows:

> "The ECCS designs are such that beginning about two minutes after occurrence of a LOCA the operator is required to reset the Safety Injection System (SIS) signal. If the operator does reset the SIS signal (a few minutes after LOCA) and if a loss of offsite power should then occur, prompt operator action would be required to restart the LOCA loads. The logic for startup of the emergency diesel generators would cause automatic sequencing to pick up the normal shutdown cooling loads in some designs and in others no loads would be sequenced (since there would be no accident signal present) rather than the LOCA loads, which would be the case if SIS had not been reset. The staff has accepted this design, in some plants provided that procedures for use of the SIS reset, and for actions required in the event of loss of offsite power after SIS reset to sequence on the proper loads without overloading the diesels, are carefully reviewed for adequacy.
>
> "The staff also should consider the effects on the core of loss of power to engineered safety features following a LOCA and after safety injection signal reset.
>
> "The resolution of this issue should be implemented on all plants."

The staff, in response to these concerns, committed to reexamine the plant operating and emergency procedures, which may direct or permit operator action concerning SIS reset to be taken as early as 10 minutes after a LOCA signal, and to require such procedures be revised to prohibit SIS reset by operator action earlier than 10 minutes after a LOCA signal unless it can be shown such action is required in the interest of safety. This commitment was to be implemented by the Office of Inspection and Enforcement (OIE). In addition, for all operating PWRs, the OIE was to verify that all safe shutdown loads are automatically loaded following an operator action to reset SIS including those necessary to assure continued cooling of the diesel generators.

Subsequently, OIE, at the request of NRR, issued Temporary Instruction TI 2515/5 January 14, 1977, for the purpose of surveying operating PWRs regarding diesel generator load sequencing in event of loss of offsite power following a LOCA and after SIS reset. Having reviewed the survey results, NRR concluded that at least 14 operating PWRs did not have written emergency procedures covering necessary corrective operator actions after reset. Also, NRR was unable to determine from the survey reports which PWRs had control system designs where operator action would be required prior to SIS reset to prevent emergency equipment from changing its operating position.

As a result of these findings, NRR sent a memo to OIE June 28, 1978, (Stello to Thornburg) requesting as follows:

> "Therefore, we request that the Office of Inspection and Enforcement initiate a program to assure that the necessary procedures are prepared where they do not already exist. (The regulatory authority for requiring such procedures is contained in 10 CFR 50.36, Technical Specifications, and 10 CFR 50 Appendix B. Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.) Specifically, the procedures should set forth:
>
> 1. The specific operator actions necessary to manually restart the required engineered safety features if a loss of offsite power or an accident occurs after the SIS has been reset and before the engineered safety feature equipment is returned to the automatic starting sequence mode by clearing the signals which initiated the SIS or resetting the RPS trip breakers.
>
> 2. The specific operator actions required prior to SIS reset to prevent any equipment from moving out of its emergency mode when the SIS is reset."

In response to the NRR request, OIE issued TI 2515/14 instructing that an inspection to verify items 1 and 2 above be performed prior to October 16, 1978.

Upon completion of these inspections, a memorandum was issued from OIE, March 19, 1979, (Moseley to Stello) summarizing the results as follows:

> "During these inspections, a total of 30 facilities were found to have either design features which cannot be blocked by the reset action or adequate procedures available. Procedures were also found at the remaining 16 facilities; however, procedure revision was required based on information discussed in the TI. All deficient procedures were corrected by December 31, 1978."

This action was thought to have completed the resolution of this concern until Licensee Event Report (LER) 79-141/01T-0 North Anna Units 1 and 2, dated November 20, 1979, which indicated that the situation was not adequately resolved.

A memorandum was then issued from Division of Operating Reactors to OIE, December 20, 1979, (Gammill to Jordan). This memorandum referred to the March 19, 1979, OIE (Moseley) memorandum which stated that as of December 31, 1978, all of the deficient procedures had been corrected. The (Gammill) memorandum also contained the following:

> "Since that time we have received a preliminary notification (PNO-II-79-25) and subsequent licensee event report (LER 79-141/CIT-0) describing design deficiencies that have recently been discovered at the North Anna Power Station Units 1 and 2, which could cause equipment important to safety to operate less conservatively than assumed in the safety analysis. Specifically,

it has been found that certain equipment would return to its non-safety mode following the reset of an ESF signal; thus, protective actions of the affected systems could be compromised once the associated actuation signal is reset.

"The drawing review during the previous IE investigation into this matter was carried out at the logic diagram level. As evidenced above, this review did not, in all cases, adequately reflect the actual system design. When the schematic diagrams were subsequently examined, the abnormalities were found. A review at the full detail level is needed to determine whether or not the above problem exists at other operating facilities.

"In light of the above, we believe the problem of equipment prematurely moving out of its emergency mode upon an SIS reset could still exist at some operating facilities. Since this problem, which we once thought to be corrected, has recurred and since we are not convinced that the licensees themselves know whether or not this problem exists at their facilities, we suggest the issuance of the enclosed proposed IE Bulletin."

Following this, OIE issued Bulletin 80-06 which covered the concern of equipment changing position after an SIS reset. The responses to this bulletin are to be reviewed by the I&C Branch. However, the bulletin does not cover the original concern of NUREG-0138 regarding the loading of the diesel generators. The North Anna event occurred after the OIE review, indicating that there were still problems with SIS reset. There is, therefore, reason to suspect that the diesel generator loading after SIS reset may still have problems as indicated by the recent PNO-V: 80-68A on San Onofre-1.

We recognize that if the San Onofre plant were designed to have no SIS reset, OIE may not have found it necessary to review the procedures covering manual actions (the response from OIE implies this). However, in view of the problems identified on North Anna and San Onofre it is obvious that design problems may exist which were neither discovered by the OIE review nor by preoperational or normal ongoing testing of these systems. In addition, some licensee's may not know how their design was implemented as indicated by the San Onofre-1 report.

The following actions describe the means by which this recurring problem could be brought to a final resolution.

Request licensees of operating reactors to:

1. submit a detailed description of the time sequence loading (manual and/or automatic) of the diesel generators under the following conditions:

   a. Loss of station power (LOSP) subsequent to a LOCA signal.

   b. LOSP subsequent to an SIS reset after a LOCA signal.

2. describe the tests which were performed to verify (1) above.

3. describe the procedural limitations on the SIS reset (in what time frame after a LOCA signal it is allowed and under what circumstances).

4. if reset is not prohibited for at least 10 minutes after a LOCA signal, as stated by the response to item 4, NUREG-0138, justify such prior action.

5. describe the manual actions necessary to reestablish the operation of equipment necessary to maintain a safe plant condition for items (1)a and (1)b above and the time frame within which they must be performed. Identify the plant procedures covering such situations.

6. describe any electrically operated equipment required for the continued operation of the emergency power sources which are not designed to be automatically loaded onto the emergency buses.

A review of this information should be performed on each operating PWR to assure that:

a. diesel generators are not overloaded because of auto loading without proper sequencing.

b. diesel generators are not being blocked from loading except as accomplished by a sequencer for auto loading.

c. the design accommodates automatic loading of equipment necessary for continued operation of the diesel generators.

d. if manual actions are required to reload the diesel generators sufficient time is available to perform such actions and that the plant has operating procedures covering such actions.

Safety evaluation reports to document the findings of these reviews showing resolution of these concerns would provide a documented basis of the staff commitment in NUREG-0138.

Attachment:
Related Documents List

Related Documents:

1. NUREG-0138
2. Memorandum: V. Stello, Director, DOR/NRR to H. Thornburg, Director, DRO/IE, dated June 28, 1978, Subject: Emergency Operating Procedures Governing SIS Reset At Operating PWRs.
3. Inspection and Enforcement Temporary Instruction TI 2515/5, January 14, 1977, (Survey of Operating PWRs Regarding D/G Load Sequencing in Event of Loss of Offsite Power Following a LOCA and Subsequent to SIS Reset).
4. Inspection and Enforcement Temporary Instruction TI 2515/14, July 14, 1978, (Survey of Operating PWRs to Assure that Facilities Having an SIS Reset Feature Have Written Procedures to Cover All Necessary Operator Actions Prior to and After SIS Reset).
5. Memorandum: N. Moseley, Director, DRO/IE to V. Stello, Director, DOR/NRR dated March 19, 1979, Subject: Emergency Operating Procedures Governing SIS Reset At Operating PWRs - Memorandum dated June 28, 1978.
6. North Anna Unit 1 Licensee Event Report (LER) 79-141/01T-0 dated November 20, 1979.
7. Memorandum: December 20, 1979, W. Gammill, Acting Director DORP/DOR to E. Jordan, Assistant Director, DROI/IE, Subject: Engineered Safety Feature Reset Design Deficiences.
8. Memorandum: January 14, 1980, D. Crutchfield, Chief, SEPB/DOR, to D. Ziemann, Chief, ORB2/DOR, Subject: Resolution of SEP Topic VII-2, ESF System Control and Design.
9. Memorandum: D. Ziemann, Chief, ORB-2/DOR to E.L. Jordan, Assistant Director, DROI/IE, Subject: Request for Reinspection of SIS Reset Procedures.
10. Memorandum: February 20, 1980, W. Gammill, Acting Director, DORP/DOR to E. Jordan, Assistant Director, DROI/IE, Subject: Comments on IE Draft Bulletin, Engineered Safety Feature Reset Design Deficiencies.
11. IE Bulletin No. 80-06 March 13, 1980, Engineered Safety Feature (ESF) Reset Controls.
12. Region Report, Page 203 of Volume II, Part 2.
13. Memorandum: May 23, 1980, Frank Nolan to S.E. Bryan, Subject: Comments - Rogovin Report Allegation RE: IE Error in Concluding That Adequate Procedures Were In Place in All Operating Reactors Including TMI-2, to Cover All Necessary Operator Actions Before and After SIS Reset.
14. Memorandum: June 2, 1980, S.E. Bryan, Assistant Director for Field Coordination, ROI/IE to N.C. Moseley, Director, ROI/IE, Subject: Rogovin Report Statement on Inadequacy of SIS Reset.
15. San Onofre Unit 1 Preliminary Notification PNO-V-80-68A September 23, 1980, Subject: Potential Failure of Emergency Diesel Generator Load Sequences to Perform as Described in FSAR.
16. Memorandum: September 30, 1980, Frank Schroeder to Robert Bernero, Subject: Safety Injection Signal Reset.
17. Memorandum: September 30, 1980, Malcolm Ernst to Gus C. Lainas, Subject: Containment Spray Pump Actuation Logic.
18. Memorandum: September 30, 1980, Frank Schroeder to Stephen S. Hanauer, Subject: Safety Injection Reset.

INADVERTENT BORON DILUTION

(22)

DRAFT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MEMORANDUM FOR: D. G. Eisenhut, Director
Division of Licensing

FROM: R. J. Mattson, Director
Division of Systems Integration

SUBJECT: PROTECTION AGAINST INADVERTENT BORON DILUTION

During our review of the accident analyses submitted for recent PWR fuel reload applications, we have concluded that many of these PWRs are not adequately protected against inadvertent boron dilution during shutdown and refueling. A detailed description of the inadequacies and the basis for the concern are provided in Enclosure 1. We have required upgrading of the protection system for boron dilution events on a case-by-case basis during our reload reviews. However, there are plants operating cycle after cycle without a special reason to submit a revised safety analysis report. Thus, inadequacies in protection against boron dilution events remain uncorrected. In order to make sure that all operating PWRs are properly protected against inadvertent boron dilution, we have concluded that a generic review of inadvertent boron dilution events and related protective measures be carried out by each licensee.

Enclosure 2 is a draft of a letter we request you issue to all licensees of operating PWR nuclear power plants. It requests they review their protection systems for inadvertent boron dilution accidents against the staff's acceptbility criteria (provided) and commit to upgrade their protection system as necessary. This requirement has been reviewed by DST and they have concurred in this action.

Roger J. Mattson, Director
Division of Systems Integtation
Office of Reactor Regulatory

Enclosures:
As stated

cc: T. Murley
P. Check
T. Speis
B. Sheron

Contact: J. Laaksonen, X28507
B. Sheron, X29453

ENCLOSURE 1

SAFETY ASSESSMENT OF BORON DILUTION EVENTS

I. BACKGROUND

Reactivity control in PWRs is accomplished using control rods and boron. During power operation, the control rods are usually removed from the core, and the boron concentration of the primary coolant is adjusted to control reactivity.

In the event that the boron concentration is inadvertently diluted during power operation, the resulting reactivity insertion will increase the reactor power and automatic safety systems will act to shut down the reactor and maintain the plant within safety limits.

When the plant is shutdown and the control rods are inserted, boron is still required to be dissolved in the primary coolant in order to maintain the necessary shutdown margin. In the event of an inadvertent dilution of the primary coolant boron concentration while at shutdown, there are no automatic safety systems which will automatically mitigate the event. Mitigation is accomplished by the operator, who must be alerted to the event, diagnose the cause, and take the proper corrective action before loss of shutdown margin occurs.

The present criteria for assessing the acceptability of plants to accomodate boron dilution events is stated in section 15.4.6, Part II of the Standard Review Plan. The key elements of these criteria are that:

(1) in the absence of a single failure, the consequences of the event should be within the limits prescribed for anticipated operational occurrences.

(2) the time available for the operator to take corrective action between actuation of alarm and loss of shutdown margin should be 30 minutes during refueling and 15 minutes during all other modes.

Moreover, Section III of 15.4.6 of the SRP states that ". . . the redundancy of alarms that alert the operator to be unplanned dilution is confirmed."

II. STATEMENT OF PROBLEM

a. Recent Problems

Recent reviews of reload applications and operating license applications have shown that significant deficiencies exist in protection against boron dilution events.

- o Case review of San Onofre II and III, Waterford, St. Lucie II, and Palo Verde have indicated the detection and alarms available during shutdown and refueling did not meet the single failure criteria.
- o Calvert Cliffs reload application did not have redundant detection and alarm systems. A boronometer which was claimed to be available during shutdown was found not be be capable of measuring boron concentration during certain modes of operation (measurement range needed not within instrument capability). The plant computer alarm cited was not audible and would require a technical specification on the operator to check the alarm readout every 8.8 minutes.
- o V. C. Summer's initial submittal measured operator action time from the time of the dilution event starting rather than the time of the alarm.

- o Detection and Alarm circuits are generally not safety grade but control grade.

- o No technical specifications generally exist which prohibit detection and alarm systems from being taken out of service during operational modes in which they would be required.

- o ANO-2 had no audible alarms for the operator to respond to. The audible count-rate meter was the proposed protection.

b. Operational History

Licensee Event Reports between 1969 and the present were examined to determine all events in which the boron concentration of the primary coolant was inadvertently changed (both increased and decreased). These are provided in Table ?.

This review indicated that 25 such events have occurred. Using the total number of PWR reactor years as approximately 300, the ~~probability~~ frequency of an event in which the boron concentration is inadvertently changed is estimated to be, 25/300 = .083 ≅ 0.1 per reactor year, or the frequency is less than 0.15 per reactor year with 90% confidence.

But what is the frequency of a radioactive release, and what release, i.e., how many Curies?

Of significance also is that a boron dilution path presently not considered in the safety analysis reports has been identified. This is the back leakage of secondary coolant into the primary through steam generator tube leaks when the system is shutdown and the primary pressure is below the secondary pressure.

For all of the events reported, mitigating action was taken by the operator before the shutdown margin was lost. No information was found on the instrumentation that alerted the operator to the event, or the rate of boron concentration change (it is reasonable to assume that the rate was less than that assumed in FSAR analyses).

Safety Significance

The consequences of unmitigated boron dilution event are typically not analyzed in Safety Analysis Reports. Rather, the analyses focus on demonstrating that sufficient time (per SRP 15.4.6) exists for the operator to identify the event and take the proper corrective action prior to loss of shutdown margin.

A boron dilution event which occurs during either shutdown or refueling is not ~~protected~~ mitigated by any automatic protection system. The only means of protecting against a loss of shutdown margin and return to criticality is the operator being alerted to the event and acting properly and timely.

criticality by itself does not lead to a radioactive release. what about temperature feedback?

While an argument could be made that the detection and alarm system needs to be safety grade, the present control grade systems are considered sufficient. However, because the event is mitigated solely by operator action, it is imperative that the detection and alarm systems be sufficiently reliable and available during all times in which they would be needed.

An unmitigated boron dilution event would probably result in the most severe consequences during refueling with the reactor head off and the primary system partially drained.

Even after criticality significant boron dilution would be required to raise temp. to bulk boiling and fuel damage.

Radiation associated with the power burst would have a direct leakage path to the containment with associated contamination of any workers who were present in containment at the time of the event.

With the reactor vessel head in place, a system overpressure and fuel damage potential exist. Consideration of this event from a thermal shock standpoint may also be warranted.

Based on the present SRP catagorization of a boron dilution event as an anticipated operational occurrence (AOO), which is supported by the operational data of Table 1, it is expected that an unmitigated boron dilution would exceed the criteria for AOO's (No fuel failure, pressure below 110% of design pressure). If an operator error rate is 0.1 to 0.01 per demand, the probability of an unmitigated boron dilution event is on the order of 10^{-2} to 10^{-3}.

Proposed Criteria

Based on the above assessment, it is concluded that all operating PWRs should meet the acceptance criteria set forth in SRP section 15.4.6 in order to assure that they are adequately protected against boron dilution events.

The following guidance is provided for demonstrating that the SRP criteria are met:

1. When analyzing boron dilution during cold shutdown and refueling modes, the fluid in the upper head of the reactor vessel and in the primary loop should be considered in poor communication with the fluid circulating in the lower part of the reactor vessel and in the RHR loop. This relatively stagnant fluid should not be credited when defining the reactor coolant volume susceptible to dilution, unless experimental evidence is presented justifying the inclusion of this volume.

2. The flow of unborated water to the reactor coolant system should be assumed to be the flow corresponding to operation of makeup and letdown systems at their full capacity if administrative measures to block part of the system during shutdown are not required in the Technical Specifications.

3. At least two independent means for detecting and alerting the operator to a boron dilution event should be available during all modes of operation. Technical Specifications for each operating mode where operator actions are the only protective measures should require that operable alarm channels cannot be taken out ~~or~~ of service deliberately ~~on purpose~~ ~~if~~ if this removal from service results in less than two independent alarm channels being available to alert the operator to boron dilution events. If all alarms are lost while in one of those modes, the plant conditions shall be monitored using a method and a frequency clearly specified in Technical Specifications.

4. Alarms making the operator aware of an unplanned moderator dilution should be distinct and audible.

5. During cold shutdown and refueling modes, postulated sources of water for boron dilution events should include backleakage from the secondary to primary side of the steam generator. Leakage rates corresponding to the technical specification allowable rates during power operation ~~operator~~ should be assumed. Thirty (30) minutes for operator response time should be available for this method of dilution.

Expected Impact

Our experience with NTOL plants to date has shown that the SRP criteria can be met primarily through the revision of the required shutdown margin and the detection alarm setpoints. For example, increasing the shutdown margin from 1 to 2 percent $\Delta k/k$ and reducing the alarm setpoint from 5 times to about 2 to 3 times background have been shown to be effective.

Cost of these changes to industry?

Most plants have mutiple source range detectors which could be alarmed. This is considered sufficient to meet the single failure criteria (redundant, independent detectors and alarms).

cost of staff time?

A reevaluation of boron dilution events will be required, as well as possible revision to the technical specifications.

Based on the serious consequences of an unmitigated boron dilution event, the fact that during shutdown and refueling, operator action is the only means of protection against this event, the present deficiencies in the detection and alarm systems as well in the adminstrat~~ive~~ re controls at operating plants, we have concluded that the proposed upgrade is necessary and justified from a cost-benefit standpoint to public health and safety.

?

BORON DILUTION EVENTS
1969 – 6/81
TABLE 1

1

RKANSAS-1 05000313 080879
RESIDUAL HEAT REMOV SYS & CONT 79-013/03X-1 112180
COMPONENT CODE NOT APPLICABLE 026695 OTHER
SUBCOMPONENT NOT APPLICABLE B&W
PERSONNEL ERROR
OTHER
ITEM NOT APPLICABLE

DURING NORMAL COLD SHUTDOWN OPERATION, THE DECAY HEAT BORON CONCENTRATION WAS NOT MONITORED DURING THE PRESCRIBED INTERVAL AS REQUIRED BY [illegible] SPEC. 4.1.B, TABLE 4.1-3, ITEM 1F. THE INSERVICE DH LOOP WAS SWITCHED WITHOUT SWITCHING THE DH SAMPLE LINEUP. SUBSEQUENT SAMPLES TAKEN [illegible] FOR BORON CONCENTRATION WERE FROM THE "OUT-OF SERVICE" LOOP. THERE [illegible] BEEN NO SIMILAR OCCURRENCES. REPORTABLE PER TECH. SPEC. 6.12.3.2 B.

THE RADIOCHEMISTRY DEPARTMENT WAS NOT AWARE OF THE SYSTEM OPERATION [illegible] OF. THE RADIOCHEMISTS WERE INSTRUCTED TO VERIFY SAMPLE LINEUP [illegible] OPERATIONS. A HARDWARE DESIGN CHANGE WAS COMPLETED, ADDING CHECK VALVES IN THE SAMPLE LINE TO PREVENT INADVERTANT SAMPLING OF THE DH LOOP OUT OF SERVICE.

2

RKANSAS-1 05000313 122179
CHEM, VOL CONT & LIQ POISN SYS 79-023/03L-0 011880
PUMPS 027964 30-DAY
CENTRIFUGAL B&W
PERSONNEL ERROR
LICENSED & SENIOR OPERATORS
ITEM NOT APPLICABLE

DURING A NORMAL MAKEUP OPERATION TO THE RCS, THE BORIC ACID PUMPS [illegible] INADVERTANTLY LEFT OPERATING. THIS RESULTED IN OVER BORATION OF THE [illegible] TOR COOLANT AND AUTOMATIC WITHDRAWAL OF RODS ABOVE THE LIMITS PER T.S. 3.5.2.5.3. THE RCS WAS IMMEDIATELY DILUTED AND THE ROD INDEX WAS RETURNED TO THE NORMAL RANGE WITHIN THE ALLOTTED 4 HOURS. LER 50-313/78-[illegible] A SIMILAR OCCURRENCE OF THE ROD INDEX OUT OF LIMITS. REPORTABLE PER T.S. 6.12.3.2.B.
DURING A NORMAL MAKEUP OPERATION TO THE RCS, A REACTOR OPERATOR INADVERTANTLY FAILED TO SECURE THE BORIC ACID PUMPS. THE OPERATOR WAS COUNSELED AND COPIES OF THE FAILURE REPORT WERE DISTRIBUTED TO ALL LICENSED OPERATORS.

3

EAVER VALLEY-1 05000334 020976
OTHER COOLANT SUBSYS & CONTROL 76 03L 030376
VALVES 018078 30-DAY
NO SUBCOMPONENT PROVIDED WEST
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
ITEM NOT APPLICABLE

(76-3/3L) DURING INITIAL FUELING, MODE 6 VESSEL HEAD IN PLACE NOT [illegible] ,GUIDE STUDS IN PLACE, WHILE TESTING S.I. VALVES, A FLOW PATH WAS INADVERTENTLY OPENED, ALLOWING THE RWST TO GRAVITY FLOW INTO THE RCS, OVERFILLING THE VESSEL. RWST BORON CONCENTRATION WAS HIGHER THAN THAT IN THE RCS. ADMIN. PROC. WILL BE MODIFIED.

4

EAVER VALLEY-1 05000334 092280
REACTIVITY CONTROL SYSTEMS 80-074/01T-0 093080
ACCUMULATORS 032536 2-WEEK
SUBCOMPONENT NOT APPLICABLE WEST
DEFECTIVE PROCEDURES
NOT APPLICABLE
OTHER

UNPLANNED DILUTION OF THE REACTOR COOLANT SYSTEM DUE TO PROCEDURAL ERROR. THE BORIC ACID BLENDER PROCEDURE DID NOT ACCOUNT FOR THE MANUAL EMERGENCY BORATION FLOW PATH BEING IN SERVICE. THE SHUTDOWN MARGIN DID NOT GO UNDER THE ADMINISTRATIVE LIMIT OF 5% DELTA K/K. THEREFORE, THERE WERE NO CONSEQUENCES TO THIS EVENT.

THE BORIC ACID BLENDER PROCEDURE DID NOT ACCOUNT FOR A CONDITION WHEN THE MANUAL EMERGENCY BORATION FLOW PATH IS IN SERVICE. THE AUTOMATIC CONTROLS OF THIS SYSTEM RAISED PRIMARY GRADE WATER FLOW TO AN AMOUNT EQUAL TO THE MISSING BORIC ACID FLOW. THE PROCEDURE HAS BEEN CHANGED TO ACCOUNT FOR THIS CONDITION.

5

CALVERT CLIFFS-1 05000317 012878
SYSTEM CODE NOT APPLICABLE 78 03L 020278
COMPONENT CODE NOT APPLICABLE 020386 30-DAY
SUBCOMPONENT NOT APPLICABLE COMB
DEFECTIVE PROCEDURES
NOT APPLICABLE
ITEM NOT APPLICABLE

RCS HAD BEEN DRAINED DOWN TO ABOUT 12" ABOVE BOTTOM OF HOT LEG IN [illegible] INSPECTION OF PRIMARY SIDE OF 11 STEAM GENERATOR. IN THIS CONDITION [illegible] S WAS BORATED TO GREATER THAN 1720 PPM. BORON SAMPLES [illegible] B. REVEALED THAT BORON CONCENTRATION HAD DECREASED TO APPROX. [illegible] BORON CONC. RAISED BUT AGAIN FELL TO LESS THAN 1720 PPM. [illegible] (78-9).

TABLE I

CRYSTAL RIVER-3
REAC COOL CLEANUP SYS & CONT
DEMINERALIZERS
SUBCOMPONENT NOT APPLICABLE
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
BABCOCK & WILCOX COMPANY

05000302 | 012677
77 011 | 020977
017161 | 2-WEEK
B&W

(77-8) WHILE IN MODE 3 (HOT STANDBY) DURING START-UP, [illegible] WAS FOUND TO BE APPROXIMATELY 50 PPM DILUTION CAUSING [illegible] INSERTION. RECOVERING FROM TRIP OF 4160V UNIT BUS [illegible] FICATION DEMINERALIZER HAD INADVERTENTLY BEEN PLACED IN [illegible] N MARGIN REMAINED 4.34% DELTA K/K. DEMINERALIZER WAS IMMEDIATELY [illegible] PLANT WAS IN MODE 3(HOT STANDBY),SHUTDOWN MARGIN REMAINED GREATER [illegible] % DELTA K/K,SAFETY ROD GROUPS WERE WITHDRAWN AND CHECKED.
OPERATOR ERRED IN PLACING UNBORATED MIXED BED DEMIN. ON LINE. [illegible] N EMPLOYED TO ALERT OPERATORS WHICH UNIT IS TO BE OUT OF SERVICE.

6

CRYSTAL RIVER-3
REAC COOL CLEANUP SYS & CONT
DEMINERALIZERS
SUBCOMPONENT NOT APPLICABLE
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
BABCOCK & WILCOX COMPANY

05000302 | 021077
77 011 | 021677
017171 | 2-WEEK
B&W

(77-12) DURING CLEANUP OF REACTOR COOLANT SYSTEM AFTER INJECTION [illegible] ,BORON CONCENTRATION WAS DILUTED BY 230 PPM. MIXED BED DEMIN. PLACED [illegible] SVC INSTEAD OF CATION BED AS INTENDED. REDUNDANT SYSTEMS WERE AVAIL[illegible] . SECOND INADVERTENT DEBORATION. FIRST FROM THIS CAUSE. MIXED BED [illegible] ATELY BYPASSED. ADDED BORIC ACID IMMED. ACT SHUTDOWN MARG. [illegible] DELTA [illegible]

PERSONNEL ERRED IN LOADING MIXED BED CATION TO DEMIN. OPERATORS CAUTIONE D TO CAREFULLY CHECK RESINS BEFORE INTRODUCTION TO SYSTEMS. ALSO, ALL [illegible] UNIS CLEARLY MARKED TO PRECLUDE RECURRENCE.

7

CRYSTAL RIVER-3
COOLANT RECIRC SYS & CONTROLS
OTHER COMPONENTS
SUBCOMPONENT NOT APPLICABLE
DESIGN/FABRICATION ERROR
CAUSE SUBCODE NOT PROVIDED
ITEM NOT APPLICABLE

05000302 | 061477
77 017 | 062877
017939 | 2-WEEK
B&W

NOT A SPECIFIC EVENT

(77-52) SUBSEQUENT TO INTRODUCTION OF NAOH INTO RCS (SEE 77-17), B&W EVA LUATION OF EVENT INDICATED POSSIBLE UNREVIEWED SAFETY QUESTION AS DEFINE D BY 10 CFR50.59(B). SINCE UNTERMINATED INJECTION OF NAOH TANK CONTENTS INTO THE RCS COULD RESULT IN CORE CRITICALITY WITH ALL RODS INSERTED. RE DUNDANCY N/A. THIS EVENT HAS NOT OCCURRED.
THIS OCCURRENCE HAVE BEEN PROVIDED.

POSSIBLE MODERATOR DILUTION BY UNTERMINATED INJECTION OF ENTIRE CONTENTS OF NAOH TANK INTO REACTOR COOLANT WHILE IN DECAY HEAT REMOVAL CYCLE

CRYSTAL RIVER-3
REACTIVITY CONTROL SYSTEMS
COMPONENT CODE NOT APPLICABLE
SUBCOMPONENT NOT APPLICABLE
OTHER
NOT APPLICABLE
ITEM NOT APPLICABLE

05000302 | 031078
78-016/01T-0 | 031778
020758 | 2-WEEK
B&W

IN MODE 5 AN UNPLANNED REACTIVITY INSERTION OF 4.3% DELTA K/K DUE TO [illegible] DILUTION OCCURRED CONTRARY TO TECH. SPEC. 6.9.1.8.B. NO SAFETY [illegible] . SHUTDOWN MARGIN GREATER THAN 10% DELTA K/K AT ALL TIMES. [illegible] NA. SECOND OCCURRENCE OF AN EVENT OF THIS TYPE. [illegible] ILUTED RCS RETURNED TO 2024 PPMB AT 1030 ON 10 MARCH 1978.

DILUTION OF THE RCS UPON DRAINING. LESS CONCENTRATED COOLANT IN THE [illegible] TSO DILUTED REACTOR VESSEL COOLANT APPROXIMATELY 350 PPMB. RCS [illegible] RVEILLANCE HAS BEEN INCREASED, AND CONCERNED PERSONNEL HAVE BEEN [illegible] ARE OF THE CONSEQUENCES OF THIS EVENT.

8

DAVIS-BESSE-1
COOLANT RECIRC SYS & CONTROLS
VALVES
NO SUBCOMPONENT PROVIDED
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
COPES-VULCAN, INC.

05000346 | 050777
77 03L | 060777
013365 | 30-DAY
B&W

(NP-33-77-2) BORATED WATER ADDED TO RAISE RX VESSEL WATER LEVEL [illegible] E DETECTORS COULD BE INSTALLED. BORATED WATER ADDED WAS [illegible] 850 PPMB. AFTER WATER LEVEL REACHED, ANALYSIS REVEALED [illegible] ON IN VESSEL 1743 PPMB. K EFF BELOW 0.92.

LEAKING DEMIN WATER SHUTOFF VALVE DILUTED BORIC ACID SO [illegible] WAS BEING ADDED THAN TOTALIZER INDICATED. OPERATORS [illegible] CID WAS BEING ADDED.

9

TABLE 1

(10)

DAVIS-BESSE-1
COOLANT RECIRC SYS & CONTROLS
VALVES
GLOBE
COMPONENT FAILURE
MECHANICAL
COPES-VULCAN, INC.

05000346 042379
79-050/011-0 050479
025942 2-WEEK
BIW

DURING RCS FILL [illegible] REPORTED [illegible] M WHICH WAS [illegible] THAN ANTICIPATED FINAL VALUE OF [illegible] PPM [illegible] LANT FLOW VERIFIED >/= 2800 GPM DURING DEBORATION IN ACCORDANCE WITH [illegible] . 3.1.1.2. EXCESS DEBORATION RESULTED IN [illegible] OF 0.59% DELTA K/K WHICH EXCEEDS 0.50% LIMIT OF [illegible] ABLE TO [illegible] WITHIN 24 HR. SHUTDOWN MARGIN WAS MAINTAINED SUBSTANTIALLY ABOVE 1.00% DELTA K/K.
COMBINATION OF MECHANICAL, PERSONNEL, AND PROCEDURAL ERROR. [illegible] XPERIENCED PREVIOUS LEAKAGE. SCHEDULING PROBLEMS PREVENTED VALVE [illegible] IN A TIMELY MANNER AND VALVE IS NOW BEING REPAIRED. CHANGES IN [illegible] D VENT PROCEDURE ARE BEING MADE. INCIDENT WILL BE REVIEWED WITH [illegible] RATIONS PERSONNEL.

(11)

MILLSTONE-2
REACTIVITY CONTROL SYSTEMS
VALVES
GLOBE
PERSONNEL ERROR
LICENSED OPERATIONS PERSONNEL
VELAN VALVE CORP.

05000336 041478
78-005/011-0 050478
021332 2-WEEK
COMB

DURING PREPARATIONS FOR STARTUP FOLLOWING THE UNIT'S FIRST [illegible] AGE, AN UNPLANNED REACTIVITY INSERTION OF MORE THAN 0.5 PERCENT [illegible] K OCCURRED. THE REACTOR COOLANT SYSTEM BORON CONCENTRATION WAS [illegible] FROM 1634 PPM TO 1473 PPM, WHICH IS EQUIVALENT TO A REACTIVITY [illegible] OF 1.9 PERCENT DELTA K/K. THE SHUTDOWN MARGIN OF [illegible] PERCENT [illegible] REQUIRED BY TECHNICAL SPECIFICATION 3.[illegible], WAS MAINTAINED [illegible] DILUTION.
THE CAUSE OF THE REACTIVITY INSERTION WAS THE ADDITION OF PRIMARY [illegible] WATER BY THE CHARGING PUMPS TO THE REACTOR COOLANT SYSTEM. THE [illegible] OF THE DILUTION WAS THE BYPASS VALVE AROUND THE PRIMARY MAKEUP [illegible] N CONTROL VALVE TO THE CHARGING SYSTEM. THE VALVE WAS [illegible] TO BE THREE QUARTERS OF A TURN OPEN. THE VALVE WAS CLOSED AND [illegible]

(12)

OCONEE-2
CHEM, VOL CONT & LIQ POISN SYS
DEMINERALIZERS
SUBCOMPONENT NOT APPLICABLE
DEFECTIVE PROCEDURES
NOT APPLICABLE
ITEM NOT APPLICABLE

05000270 050780
80-003/011-0 060580
031185 2-WEEK
BIW

A DEBORATING DEMINERALIZER WAS PLACED IN SERVICE WHILE OCONEE 2 WAS [illegible] OLD SHUTDOWN & RCS WAS DRAINED DOWN. CHEMISTRY RECORDS & THE FIRST [illegible] LE INDICATED A BORON CONCENTRATION CLOSE TO THAT OF RCS. HOWEVER, [illegible] BORON CONCENTRATION WAS REDUCED FROM 1895 PPM TO 1539 PPM. A SUBSTANTIAL CONCENTRATION ABOVE THAT NECESSARY FOR A 1% DELTA K/K SHUTDOWN [illegible] AS MAINTAINED. INCIDENT NOT CONSIDERED TO BE SIGNIFICANT WITH [illegible] D SAFE OPERATION, & HEALTH & SAFETY OF THE PUBLIC WERE NOT AFFECTED. APPARENTLY THE CHEMISTRY RECORDS WERE NOT UP-TO-DATE. IT IS POSSIBLE AT AN EQUILIBRIUM DEMINERALIZER OUTLET CONCENTRATION WAS NOT REACHED [illegible] OR TO TAKING THE FIRST SAMPLE. ADMINISTRATIVE CONTROLS WILL BE [illegible] D CONCERNING USE OF THE DEMINERALIZERS, INCLUDING PRECLUDING USE [illegible] HE RCS IS DRAINED DOWN.

(13)

ROBERT E. GINNA-1
CHEM, VOL CONT & LIQ POISN SYS
COMPONENT CODE NOT APPLICABLE
SUBCOMPONENT NOT APPLICABLE
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
ITEM NOT APPLICABLE

05000244 070570
70 01T 071570
004081 2-WEEK
WEST

OPERATOR ATTEMPTED TO DILUTE BORON WHILE REACTOR WAS CRITICAL AND THERE WAS NO COOLANT PUMP OR RESIDUAL HEAT REMOVAL PUMP IN OPERATION, [illegible] TO REQUIREMENTS OF TECHNICAL SPECIFICATIONS.

ADDITIONAL INFORMATION ON CORRECTIVE MEASURES IS INCLUDED IN [illegible] 2-17-70
OPERATOR DID NOT FOLLOW PROCEDURES.

GINNA-1 05000244 011874
COOLANT RECIRC SYS & CONTROLS 74 011 012874
OTHER COMPONENTS 000/08 2-WEEK
SUBCOMPONENT NOT APPLICABLE WEST
DEFECTIVE PROCEDURES
CAUSE SUBCODE NOT PROVIDED
NO MANUFACTURER SPECIFIED

AN UNCONTROLLED CHANGE IN REACTIVITY OF APPROXIMATELY [illegible] RED WHICH REDUCED SHUTDOWN MARGIN TO 13%. PRIMARY SYSTEM WATER WAS [illegible] SED FROM A STEAM GENERATOR. (50-244/74-1)

PERSONNEL EXPOSURE WAS NEGLIGIBLE.

14

DURING S.G. MAINTENANCE A MANWAY WAS OPENED AND WATER IN THE S.G. [illegible] OF 913 P.P.M. BORON EMPTIED INTO THE COOLANT LOOPS AND DILUTED THE [illegible] P.P.M BORON CORE COOLANT.

ROBERT E. GINNA-1 05000244 041978
CHEM, VOL CONT & LIQ POISN SYS 78-005/01T-0 050278
VALVES 021334 2-WEEK
DIAPHRAGM WEST
PERSONNEL ERROR
NONLIC. OPERATIONS PERSONNEL
ITEM NOT APPLICABLE

DURING RESIN REMOVAL FROM THE A DEBORATING DI, OPERATOR ERRONEOUSLY [illegible] ED BOTH A AND B DEBORATING DI RESIN OUTLETS THEREBY CREATING PATH [illegible] ACTOR MAKEUP WATER TO THE REACTOR COOLANT SYSTEM WHICH WAS IN REFUELING MODE. BEFORE ERROR WAS CORRECTED, PRIMARY LOOP BORON CONCEN. DECREASED FROM 2140 PPM TO 2072 PPM. THIS RESULTED IN AN UNPLANNED REACTIVITY [illegible] RTION OF > 0.5% DELTA-K/K (T.S. 6.9.2.A(4)) HOWEVER PRIMARY LOOP [illegible] WAS NOT REDUCED BELOW THE 2000 PPM REQUIRED FOR REFUELING MODE OPERATION. UNLICENSED OPERATOR OPENED THE WRONG RESIN OUTLET VALVE AND THEN OPENED THE CORRECT VALVE PRIOR TO CLOSING THE FIRST ONE. OPERATOR INSTRUCTED TO FOLLOW PROCEDURE. PROCEDURE CHANGE INSTITUTED TO INCREASE CLARITY.

15

SAN ONOFRE-1 05000206 100177
COOLANT RECIRC SYS & CONTROLS 77-014/01T-0 100977
HEAT EXCHANGERS 019293 2-WEEK
STEAM GENERATOR WEST
PERSONNEL ERROR
NONLIC. OPERATIONS PERSONNEL
WESTINGHOUSE ELECTRIC CORP.

AN INADVERTENT DILUTION OF REACTOR COOLANT SYSTEM OCCURRED WHEN SECONDARY SIDE OF "B" STEAM GEN WAS FILLED FOLLOWING TUBE PLUGGING OPERATIONS. THE DILUTION WAS CAUSED BY PRIMARY TO SECONDARY LEAKAGE. BORON DILUTION OF APPROX. 75 PPM RESULTED IN POSITIVE REACTIVITY ADDITION INSERTION OF [illegible] 1.16(T.S. 6.9.2.A(4)). SHUTDOWN MARGIN REMAINED >6% CONTAINMENT INTEGRITY WAS INTACT.

16

EDDY CURRENT TESTING INDICATED DEFECTIVE TUBE WAS RESTRICTED AT 4TH SUPPORT PLATE. IN FUTURE, TUBE SHEETS WILL BE VISUALLY INSPECTED FOR [illegible] E DURING FILLING OF SECONDARY SIDE.

SAN ONOFRE-1 05000206 070580
REACTIVITY CONTROL SYSTEMS 80-029/01T-0 072180
HEAT EXCHANGERS 031699 2-WEEK
STEAM GENERATOR WEST
COMPONENT FAILURE
MECHANICAL
WESTINGHOUSE ELECTRIC CORP.

INADVERTENT DILUTION OF REACTOR COOLANT SYSTEM OCCURRED DURING A TIME WHEN CONTAINMENT INTEGRITY WAS NOT ESTABLISHED. AT NO TIME DID BORON CONCENTRATION DECREASE BELOW 2400 PPM WHICH REPRESENTS A KEFF LESS THAN OR EQUAL TO 0.90. THERE WAS NO EFFECT UPON PUBLIC HEALTH OR SAFETY.

17

INFLATABLE PLUGS USED TO ISOLATE STEAM GENERATORS DURING CHANNEL HEAD DECONTAMINATION LEAKED. SEVERAL ATTEMPTS TO STOP LEAKAGE FAILED. DECONTAMINATION OF STEAM GENERATORS CHANNEL HEADS WAS DISCONTINUED.

SAN ONOFRE-1 05000206 090180
REACTIVITY CONTROL SYSTEMS 80-034/03L-0 091780
COMPONENT CODE NOT APPLICABLE 032520 30-DAY
SUBCOMPONENT NOT APPLICABLE WEST
DEFECTIVE PROCEDURES
NOT APPLICABLE
ITEM NOT APPLICABLE

WHILE PERFORMING STEAM GENERATOR TUBE REMOVAL, UNEXPECTED WATER IN THE SECONDARY SIDE INTRUDED INTO THE REACTOR COOLANT SYSTEM. A POSITIVE REACTIVITY INSERTION OF $0.44 OCCURRED AS A RESULT OF A 35PPM BORON DILUTION. THIS EVENT OCCURRED WITHOUT CONTAINMENT INTEGRITY AS REQUIRED BY TECHNICAL SPECIFICATION 3.6.B(3). INVESTIGATION REVEALED AN ON-GOING GRADUAL RCS DILUTION. THERE WAS NO DEGRADATION TO PUBLIC SAFETY.

18

35PPM DILUTION OCCURRED DUE TO UNEXPECTED WATER IN PREVIOUSLY DRAINED SECONDARY SIDE. A DRAIN HAS BEEN PROVIDED BETWEEN WATER SOURCE AND GENERATORS. GRADUAL DILUTION OCCURRED DUE TO INADEQUATE MAKEUP BORON CONCENTRATION AT A VERY CONSERVATIVE SHUTDOWN MARGIN. MAKEUP BORON CONCENTRATION HAS BEEN ADJUSTED.

Table 1 5 of 6

SAN ONOFRE-1 05000206 092280
REACTIVITY CONTROL SYSTEMS 80-036/03L-0 100380
OTHER COMPONENTS 032814 30-DAY
SUBCOMPONENT NOT APPLICABLE WEST
DEFECTIVE PROCEDURES
NOT APPLICABLE
ITEM NOT APPLICABLE

(19)

INADVERTENT DILUTION OF REACTOR COOLANT SYSTEM OCCURRED [illegible] EN CONTAINMENT INTEGRITY WAS NOT ESTABLISHED. AT NO TIME DID [illegible] ENTRATION DECREASE BELOW 2500 PPM WHICH REPRESENTS [illegible] EQUAL TO 0.90. THERE WAS NO ADVERSE EFFECT TO PUBLIC HEALTH OR SAFETY.

DILUTION RESULTED FROM A RCS LOOP SEAL FAILURE DURING STEAM GENERATOR [illegible] ANNEL HEAD CONTAMINATION. DECONTAMINATION OF STEAM GENERATOR CHANNEL HE AD WAS DISCONTINUED UNTIL THE LOOP SEAL MATERIAL WAS REPLACED AND [illegible] AL REQUALIFIED.

ST. LUCIE-1 05000335 123180
CHEM. VOL CONT + LIQ POISN SYS 80-071/03L-0 012981
VALVES 033747 30-DAY
OTHER COMB
COMPONENT FAILURE
MECHANICAL
CROSBY VALVE & GAGE CO.

(20)

DURING NORMAL OPERATION, AT 100% POWER, THE VOLUME CONTROL TANK (VCT) WA S INADVERTENTLY DILUTED, LEADING TO REACTOR COOLANT SYSTEM DILUTION. IN DICATED REACTOR POWER INCREASED OVER ABOUT 15 MINUTES, PEAKING AT 102%. TC ALSO INCREASED DURING THE TRANSIENT REACHING 542.5 DEG. F. (THE STEA DY STATE LIMIT IS 542 DEG. F.). ACTION PER TECH. SPEC. 3.2.5 WAS TAKEN AND POWER AND TC WERE REDUCED. THIS IS THE FIRST EVENT OF THIS TYPE.

WHILE ADDING MAKEUP AND ADJUSTING BORON CONCENTRATION IN THE VCT, A RELI EF VALVE (V2173, CROSBY, STYLE JR-HR-S) ON THE ACID PUMP DISCHARGE [illegible] D, RETURNING BORIC ACID TO THE SOURCE TANK. FLOW INDICATION, UPSTREAM F ROM THE RELIEF, WAS NORMAL. THE RELIEF VALVE WAS EXERCISED AND [illegible] PAIRED.

SURRY-1 05000280 011875
REACTIVITY CONTROL SYSTEMS 75 01T 022475
INSTRUMENTATION + CONTROLS 012264 2-WEEK
NO SUBCOMPONENT PROVIDED WEST
DESIGN/FABRICATION ERROR
CAUSE SUBCODE NOT PROVIDED
ITEM NOT APPLICABLE

(21)

THE PRIMARY SYSTEM BORON CONCENTRATION HAS DILUTED TO 1842 PPM DURING A COLD SHUTDOWN CONDITION FOR A REFUELING OUTAGE. [illegible] PERIOD TEST CONDITIO HAD RESULTED IN A TEMPORARY LOSS OF POWER TO THE [illegible] CONTROLL ER FC-1114A. (AO-75-1) THE BORON CONCENTRATION DID NOT GO BELOW [illegible] C. REQUIREMENTS.

DESIGNS UPON LOSS OF POWER TO FC-1114A, THE CONTROLLER REVERTS TO THE MA NUAL MODE OF OPERATION, AND REMAINS POSITIONED AT POSITION AT LOSS OF PO WER. PANEL INDICAT. LIGHTS FOR "MANUAL [illegible]" WERE BURNED OUT.

SURRY-2 05000281 052176
MAIN STEAM SUPPLY SYS + CONT 76 01X 072776
HEAT EXCHANGERS 014883 OTHER
NO SUBCOMPONENT PROVIDED WEST
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
WESTINGHOUSE ELECTRIC CORP.

(22)

(76-3) WITH UNIT 2 AT COLD SHUTDOWN FOLLOWING A REFUELING, ROUTINE CHEMI STRY ANALYSIS OF THE PRIMARY COOLANT BORON CONCENTRATION INDICATED THAT AN UNPLANNED DILUTION HAD OCCURRED. (REPORTABLE PER T.S.6.6.2.A(4)). TER MINATE THE SOURCE OF THE DILUTION, LEAKAGE FROM THE SECONDARY SIDE OF 2A STEAM GENERATOR, BY SECURING THE AUXILIARY STEAM GENERATOR FEED

PERSONNEL ERROR: STEAM GENERATOR "2A" HAD THREE TUBES WHICH HAD APPARENT LY BEEN CUT DURING THE REMOVAL OF A SECTION OF THE SEVENTH TUBE SUPPORT PLATE. WHEN THE S/G LEVEL WAS RAISED ABOVE THE LEVEL OF THE CUTS, [illegible]

SURRY-2
CHEM. VOL CONT + LIQ POISH SYS
INSTRUMENTATION + CONTROLS
CONTROLLER
COMPONENT FAILURE
ELECTRONIC
HAGAN CONTROLS

05000281 [illegible]78
78 012/011-0 040478
020897 2 WEEK
WEST

AT ABOUT 1 [illegible] 3-2[illegible]-78, WITH THE [illegible] [illegible]
HED, THE REACTOR BORON CONCENTRATION WAS [illegible]
PPM FROM THE PREVIOUS DAY'S SAMPLE OF 1372 PPM. THIS [illegible]
UNPLANNED REACTIVITY ADDITION OF MORE THAN 0.5[illegible]. SINCE CONTAINMENT [illegible]
GRITY WAS NOT MET, THIS EVENT IS CONTRARY TO T.S. 3.[illegible].A.5. AND [illegible]
EPORTABLE PER T.S. 6.6.2A(4). THE HEALTH AND SAFETY OF THE PUBLIC
NOT AFFECTED SINCE REACTOR WAS AT ALL TIMES GREATER THAN [illegible]
A FAILED CAPACITOR IN FLOW CONTROLLER FC-2-11[illegible] CAUSED THE PRIMARY
(PG) WATER VALVE FCV-114A TO OVER-FEED DURING BLEND OPERATION [illegible]
URE ALSO INTERFERED WITH THE PG WATER FLOW DEVIATION ALARM [illegible]
TERY SYSTEMS WERE RETURNED TO NORMAL. PROCEDURES WILL BE REVISED [illegible]
BSEQUENT OPERATION WAS NORMAL

23

ZION-1
REAC CORE ISOL COOL SYS + CONT
VALVES
NO SUBCOMPONENT PROVIDED
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
ITEM NOT APPLICABLE

05000295 100476
76 03L 101576
016043 30-DAY
WEST

(RO 76-62). CHEMICAL ANALYSIS AND VISUAL OBSERVATIONS OF THE PRI[illegible]
LEVEL INDICATED AN UNPLANNED DILUTION OF THE REACTOR COOLANT [illegible]
D CONCENTRATION DROPPED TO 964 PPM. INJECTION RATE WAS ABOUT [illegible]
ACTOR WAS IN SHUTDOWN CONDITION AND BORON CONCENTRATION LIMIT [illegible]
WAS NOT VIOLATED.

24

VALVE 1IW0153, NORMALLY CLOSED, WAS IN OPEN POSITION. VALVE WAS CLOSED [illegible]
PERSONNEL REINSTRUCTED AND PROCEDURES CHANGED TO BETTER FOLLOW CHANGE [illegible]
BORON CONCENTRATION.

ZION-2
CNTMNT ISOLATION SYS + CONT
COMPONENT CODE NOT APPLICABLE
SUBCOMPONENT NOT APPLICABLE
PERSONNEL ERROR
CAUSE SUBCODE NOT PROVIDED
ITEM NOT APPLICABLE

05000304 031077
77 01T 032477
017407 2-WEEK
WEST

(RO 77-9)WHILE IN COLD SHUTDOWN, THE UNIT 2 RC LEVEL WAS FOUND TO BE INC
REASING FROM THE 594 FT LEVEL AT 2000 HOURS TO THE 597 FT LEVEL AT [illegible]
OURS (ON 3-10-77). THIS 3 FT INCREASE IN RC LEVEL HAD OCCURRED WITH NO C
HANGES ON THE CONTROL ROOM BOARD LINEUP.

25

THE UNPLANNED DILUTION OF THE RC SYSTEM OCCURRED BECAUSE VALVES 2IW0[illegible]
AND 2IW0155 WERE INCORRECTLY IN THE OPEN POSITION. PROCEDURES WERE [illegible]
ED.

To: All Licensees of Operating Pressurized Water Nuclear Power Reactors

Gentlemen:

SUBJECT: PROTECTION AGAINST INADVERTENT BORON DILUTION

During our review of accident analyses submitted in support of recent PWR fuel reload applications, we have concluded that many operating ~~of these~~ PWR's are not adequately protected against inadvertent boron dilution during shutdown and refueling.

The experience gained from operating plants has shown that inadvertent changes in the boron concentration are expected to occur with a frequency of approximately once every 10 reactor years.

In all of the cases reviewed, the operator was able to stop the dilution before the shutdown margin was lost.

Notwithstanding the successful operating experience to date, we believe that the consequences of an unmitigated boron dilution event are serious enough to require that all operating PWRs meet tha present Standard Review Plan (SRP) criteria for protecting against boron dilution events. The following guidance is provided for demonstrating that SRP 15.4.6 Acceptance Criteria are met:

1. When analyzing boron dilution during cold shutdown and refueling modes, the fluid in the upper head of the reactor vessel and in the primary loop should be considered in poor communication with the fluid circulating in the lower part of the reactor vessel and in the RHR loop. This relatively stagnant fluid volume should not be included ~~credited~~ when defining the reactor coolant volume susceptible to dilution, unless experimental evidence is presented justifying the inclusion of this volume.

2. The flow of unborated water to the reactor coolant system should be assumed to be the flow corresponding to operation of makeup and letdown systems at their full capacity if administrative measures to block part of the system during shutdown are not required in the Technical Specifications.

3. At least two independent means for detecting and alerting the operator to a boron dilution event should be available during all modes of operation. Technical Specifications for each operating mode where operator actions are the only protective measures should require that operable alarm channels cannot be taken out of service on purpose if this removal from service results in less than two independent alarm channels being available to alert the operator to boron dilution events. If all alarms are lost while in one of those modes, the plant conditions shall be monitored using a method and a frequency clearly specified in Technical Specifications.

4. Alarms making the operator aware of an unplanned moderator dilution should be distinct and audible.

5. During cold shutdown and refueling modes, postulated sources of water for boron dilution events should include backleakage from the secondary to primary side of the steam generator. Leakage rates corresponding to the technical specification allowable rates during power operator should be assumed. Thirty (30) minutes for operator response time should be available for this method of dilution.

These numbers assume a diesel failure rate of 0.03 failures per start attempt for each of two diesels (no common mode failures). Field experience indicates an average failure rate of roughly 0.05 failures per start attempt, with about 5% of these being common mode. (The numbers are also based one one plant design. Other plants may differ.)

It is estimated that this rate can be improved to .02 failures/attempted start. It is further estimated that about 22 plants will need to install the hardware modifications cited in Reference 9 to achieve this target rate.

Removing the assumption of 0.03 failures/attempt from the WASH-1400 figures, and inserting the new probabilities with a 5% common mode assumption, the results are:

Release Category	PWR-1	PWR-2	PWR-6
Present Probability	1.6×10^{-7}	1.4×10^{-5}	3.2×10^{-6}
Improved Probability	4.5×10^{-8}	4.1×10^{-6}	9.2×10^{-7}
Curies Released	1.2×10^{9}	9.3×10^{8}	1.0×10^{8}
$\Delta F_n R_n^{1.2}$	9.0×10^{3}	5.7×10^{5}	9.1×10^{3}

$$\text{Total } \Delta F_n R_n^{1.2} = 5.9 \times 10^{5}$$

It is estimated that the cost to NRC will be $400,000 and the cost to a licensee will be $830,000. The priority score is then:

$$S = \frac{N\Delta[FR^{1.2}]}{C + NI}$$

$$S = \frac{(22)(5.9 \times 10^{5})}{(.4) + (22)(.83)}$$

$$S = 7 \times 10^{5}$$

To calculate the range, we assume a factor of two range in both the initial and final failure rates. The transient frequencies and radioactive releases are assumed to be within a factor of 5, and costs to be within +5 or -2. The result is a range of :

$$7 \times 10^{4} \text{ to } 6 \times 10^{6}$$

8. Boron Dilution.

Many PWRs have no positive means of detecting boron dilution during cold shutdown.[10] Some operations carried out during outages (e.g., steam generator decontamination) reduce the RCS volume, thus speeding up dilution. Boron dilution has taken[11] place during such operations, although thus far criticality has not occurred.

The frequency is difficult to estimate since no actual radioactivity-releasing events have occurred in the 280 PWR-years of current experience. Moreover, the event under consideration occurs during shutdown conditions, for which no appropriate event tree studies exist.

An upper limit on frequency can be calculated based on statistical considerations. If we assume a Poisson process, having zero occurrences in the 280 reactor-years implies that, to 95% probability.

$F \leq 1.1 \times 10^{-2}$ events/R-Y

To get a lower bound, we make the analogous pragmatic assumption (based also on our judgement) that the probability of one event in the same period is at least 5%. This implies that:

$F \geq 1.9 \times 10^{-4}$ events/R-Y

A "best estimate" value is needed for the priority score formula. The statistical methods above cannot give this, so we take the pragmatic approach of using the geometric mean of the two limits:

$F = 1.4 \times 10^{-3}$ events/R-Y.

A criticality caused by boron dilution during shutdown (with all rods in) will take time to reverse. We make the reasonable but purely judgmental assumption that 1% of the fuel pins release gap activity. If 180 assemblies are present, 1% of the total gap activity would be 4.3×10^{5} Ci. Since the reactor head is removed, about 10% of this activity would escape from the containment ventilating system. (See dropped fuel assembly accident analysis, Surry FSAR.)

$R = 4.3 \times 10^{4}$ Ci

All 43 operating PWRs are affected by this condition. The fix is to install instrumentation to detect the event and stop the dilution, either automatically or, if the detection is sufficiently early, by alerting the operator. Cost to the licensee is estimated to be $200,000. NRC costs are estimated to be about one month of generic effort plus one week per reactor, or $73,000. The priority score is:

$$S = \frac{N\Delta[FR^{1.2}]}{C + NI}$$

$$S = \frac{(43)(1.4 \times 10^{-3})(4.3 \times 10^{4})^{1.2}}{(.073) + (43)(.2)}$$

$$S = 3 \times 10^{3}$$

To estimate the frequency range, we use the 95% statistical bounds given above. The release is also quite uncertain, and a factor of 10 is used here. Costs are assumed to be within -2 to +5. The resulting range is:

2×10^{2} to 5×10^{4}