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March 5, 1990

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT UNIT 1 - INSPECTION REPORT NOS. 50-327/90-01 AND 50-328/90-01 - ENFORCEMENT CONFERENCE FOLLOWUP

TVA met with the NRC staff on February 14, 1990, to discuss potential enforcement actions resulting from the subject inspection. The discussions at the enforcement conference were related to TVA's action relative to deadheading under certain plant conditions of a residual heat removal (RHR) pump during two pump operation.

TVA believes that the meeting was beneficial and has provided a better understanding of the problems addressed in the inspection report, the enforcement issues, and the status of our corrective actions. The enclosure to this letter provides a summary of the information presented by TVA in the meeting regarding the findings in the inspection report related to the enforcement issues. In several cases TVA is providing information that differs from the inspection findings. A revision to Licensee Event Report (LER) 50-327/89031 was submitted separately to reflect the current status of our review findings and corrective actions. We hope that the information in this letter and the LER revision will assist you in your determination of any appropriate enforcement action.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

M. O. Medford

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Enclosure
cc: See page 2

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ENCLOSURE

TVA met with the NRC staff on February 14, 1990, to discuss potential enforcement actions resulting from the subject inspection. The discussions at the enforcement conference were related to TVA's action relative to potential deadheading under certain plant conditions of an RHR pump during two pump operation. In addition, an error in the preparation of a safety assessment for an emergency operating procedure change on RHR pump operation to address the RHR pump interaction was discussed. This enclosure provides a summary of the information presented by TVA in the meeting regarding the findings in the inspection report related to the enforcement issues. In several cases TVA is providing information that differs from the inspection findings. A revision to Licensee Event Report (LER) 50-327/89031 was submitted separately to reflect the current status of our review findings and corrective actions. We hope that the information in this letter and the LER revision will assist you in your determination of any appropriate enforcement action.

1. Nuclear Experience Review Program

The inspection report contains three findings (at pages 15, 18, and 23) relative to the nuclear experience review program. One finding was the determination by the inspector that TVA had been informed of the RHR pump-to-pump interaction by NRC Information Notice 87-59 and two Westinghouse letters issued in October and November, 1987. The second finding was that the nuclear experience review (NER) program relies on the CAQR program to make operability determinations only after distribution of incoming information to line organizations. The third finding was the lack of a formal policy for time sensitive determination. A recommendation was also made to notify Operations of immediate operability concerns to obtain formal operability determination.

TVA conducted a review of the previous NER items (NRC Information Notice 87-59 and Westinghouse letters issued on October and November 1987). These items discussed two potential problems with the RHR system: deadheading of pumps with common miniflow lines and inadequate miniflow capacity for single pump operation. Later Westinghouse correspondence (November 1987) concluded that the earlier information was not applicable to SQN because of separate RHR miniflow lines and the hydraulic isolation of other emergency core cooling system (ECCS) pumps with common miniflow lines through the use of separate flow restricting orifices for each pump. The TVA evaluation of these items led to the same conclusion regarding the pump-to-pump interaction and also confirmed the adequacy of miniflow capacity for single RHR pump operation. The pump-to-pump interaction discussed in Bulletin 88-04 was not sufficiently defined until receipt of another Westinghouse letter in late May 1988, which described the specific SQN interaction and noted that their previous conclusions regarding pump deadheading potential made in November 1987 were no longer correct. The RHR pumps were included in the evaluation of Bulletin 88-04 at this time.

While the NER program was not directly involved in this problem, several changes to the NER program had been made in June 1989. A dedicated and expanded staff is now onsite at TVA plants. The experience and qualifications of managers in the NER program have been upgraded. Weekly reviews are conducted by conference call between the sites and Corporate NER groups to identify significant safety issues. The new organization and staffing provide improved capability to recognize and act on CAQRs and potential significant safety issues. In particular, items that are identified as potentially safety significant are designated as immediate attention items and handcarried to the appropriate principal managers for evaluation. The line managers complete the evaluation and make operability and immediate reportability determinations. Items that meet the criteria are written as CAQRs and are reviewed by the management review committee. TVA agrees that this policy was not explicitly defined in the TVA Standard STD-1.3.1, "Managing the Nuclear Experience Review Program." This standard has since been revised to directly incorporate this policy.

TVA has considered the recommendations regarding the use of Operations to assess potentially significant safety issues to make formal operability determinations. TVA also assessed the types of items that have been recently processed as immediate attention items. As a result we have concluded that the types of items handled in this manner typically require technical review prior to making the operability determination. Consequently, TVA believes the NER program established in June 1989 is adequate.

2. Emergency Instruction E-0, Revision 7

After verifying that the RHR pump deadheading problem existed for SQN Unit 1, TVA issued a procedure change that required the operators to shut off both RHR pumps based solely on RCS pressure being above 180 psig. In addition the new step was placed before steps that examine certain parameters that diagnose whether a loss of coolant accident (LOCA) occurred. During the night of December 6, 1989, Operations prepared Emergency Procedure E-0, Revision 7, to stop both RHR pumps if reactor coolant system (RCS) pressure was greater than 180 psid. It was believed that this was a minor change as a similar step already existed in the procedure but later in the sequence. The need for this change was based on an NE calculation that determined that operation in the deadheaded configuration for longer than 11 minutes would lead to pump damage. The addition of a step early in the procedure was intended to terminate deadheaded operation before pump damage could occur.

The following day, discussion commenced with NRC and Westinghouse to address concerns raised regarding the stoppage of both RHR pumps (as opposed to only one) early in the postaccident sequence. TVA prepared E-0, Revision 8, on December 8, 1989, to require stopping one RHR pump in response to potential deadheaded conditions. This revision was made with Westinghouse concurrence and is consistent with their generic recommendations for addressing Bulletin 88-04.

Revision 7 to E-0, which required stopping of both RHR pumps, was determined to be technically deficient. The technical evaluation was done solely on the basis of the accident analyses presented in the Final Safety Analysis Report (FSAR). During this evaluation, the reviewers did not adequately review the potential impact of the change in that they only considered the specific break sizes analyzed in the FSAR. FSAR, Chapter 15, addresses a specific set of bounding 10 CFR 50, Appendix K, breaks. They considered these breaks to be bounding for all cases when, in fact, the procedure change had an impact on other break sizes. The FSAR did not explicitly define the key assumptions regarding RHR operation for small break LOCAs; therefore, the reviewers assumed that no credit was taken for RHR injection for these breaks. The Operations personnel responsible for implementation of the emergency procedure program had assumed that additional responsibility during a recent reorganization of the Operations staff. Special emphasis had not been provided during indoctrination in this function and program requirements were not fully understood by the personnel involved with the subject revision. Consequently, Revision 7 to Emergency Procedure E-0 was considered to be a minor change by the preparers, Westinghouse was not consulted regarding the change, and the emergency procedure change evaluation was not well documented.

The administrative controls have been strengthened for emergency operating procedure changes. Plant Operations Review Committee review and Plant Manager approval is now required for all emergency procedure changes. The step deviation document process will be enhanced by detailing specific evaluation criteria in the procedure generation program (PGP). Specific criteria will be established and added to the PGP to define when Westinghouse consultation is required. Additional clarification on verification and validation requirements will be added to the PGP, with particular emphasis on simulator validation whenever possible. In the interim, Administrative Instruction (AI) 2, "Guidelines for Preparing, Verifying and Validating Operating Instructions," has been revised to require Westinghouse concurrence with any change to the emergency operating procedures that deviates from the Westinghouse Owner's Group/Emergency Response Guidelines and to strengthen verification and validation requirements. Training on these changes will be conducted for involved personnel. Westinghouse had previously confirmed the technical adequacy of the emergency procedures in September 1989. Revisions 7 and 8 to E-0 were the only emergency procedure changes made since the Westinghouse evaluation.

3. Safety Assessments for Emergency Instruction E-0 Changes

The inspection report findings (pages 6 and 7) conclude that a safety evaluation was not performed because E-0 is not described in the FSAR. Secondly, the inspection findings conclude that the preparers and reviewers did not consider "the question of whether the procedure is described in the SAR is meant to encompass the function that the procedure controls." The safety assessment form contains two questions that are relevant to this issue. The first question asks whether the change will "affect system operation characteristics from that described in the SAR?" The second question asks whether the change will "affect a process or procedure outlined, summarized, or described in the SAR?" (Emphasis added in both questions) The intent of these questions is to determine whether the procedure is described directly in the FSAR (second part of the second question) or whether the function that the procedure controls is described (first question and the first part of the second question).

During the preparation of the safety assessments for both Revisions 7 and 8 to E-0, the individuals involved made an incorrect determination that, because the change had no consequences (based on their review of the accident analyses discussed in the FSAR), it had no effect on RHR system operation or information provided in the FSAR. Accordingly, the safety assessment questions were checked "no" indicating that a safety evaluation was not required. In both Revisions 7 and 8, it was clear that the reviewers were concerned with whether E-0 itself was described in the FSAR as well as whether the RHR system operation, as controlled by E-0 was described in the FSAR. As such, we believe the reviewers were addressing the questions of how the procedures were described in the FSAR in the correct manner.

In addition, TVA does not believe that the description of emergency instructions in Chapter 13 of the FSAR contains sufficient detail to describe any system operating characteristic or process for operator response to reactor trips or safety injections. Rather, the intent of the Chapter 13 discussion on emergency instructions is to establish that a category of procedures and administrative controls have been established for accident response.

The inspection findings (at page 7) conclude that neither safety assessment for the changes to E-0 addressed the accident scenario involving the recirculation mode of RHR. The safety assessment for Revision 7 concluded in the evaluation of the potential safety impact that both RHR trains remain operable and only one is required for short and long term accident mitigation (emphasis added). The reference to long-term accident mitigation encompasses long term recirculation mode of RHR operation. The assessment was made with the knowledge that procedures already exist to establish and control the recirculation mode of operation. The safety assessment for Revision 8 concluded in the evaluation of potential safety impact that both trains of RHR remain operable and that "each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period (emphasis added)."

As a result of TVA's review of the safety assessment problems, two areas for correction were identified during a review of the safety assessment problems. A clarification to the FSAR regarding RHR operation for small break LOCAs is scheduled for the April 1990 update. A training letter was sent to all Level I and Level II 50.59 reviewers describing the RHR event, the subsequent procedure changes, safety evaluations, and the lessons learned. Particular emphasis was placed on the fact that a change without consequence could still result in having an effect on the system that would require a safety evaluation. This training is considered an enhancement to the major 50.59 program changes made in November 1989. The inadequate safety evaluation is a single implementation problem identified under the revised 50.59 process, as noted in NRC Inspection Report 90-01.

The inspection report concludes (at page 7) that an unreviewed safety question (USQ) was not involved with the Revision 7 change to E-0. This conclusion is apparently based on information provided by Westinghouse and given to NRC during the inspection. This information was provided to TVA by Westinghouse in a January 11, 1990, letter and constitutes the Westinghouse input for TVA's justification for continued operation regarding the Revision 7 procedure change. That information is not considered an unreviewed safety question determination (USQD) and TVA has not performed a separate USQD for Revision 7.

There was also a potential impact on peak clad temperature under certain conditions during the short period of time that E-0, Revision 7, was in effect. The specific conditions include a limited spectrum of RCS breaks coupled with an assumed operator error in failing to restart at least one RHR pump when conditions require it. Mitigating factors include the fact that the particular break size of concern (1-3 square foot range) is nonmechanistic (larger than all branch connections yet smaller than the double-ended cold leg guillotine break), leak-before-break methodology that indicates that increased, detectable RCS leakage will occur before any break occurs, generic best-estimate accident analyses that would predict more time for operator reinitiation of RHR flow, and good operator training on fundamentals that reduce the chance for error. In fact, operators taking simulator examinations during this period (using E-0, Revision 7) correctly executed the activities to stop and restart the RHR pumps in response to LOCA conditions.

The operator actions and required timeframes are consistent with the reactor coolant pump trip criteria that received generic NRC approval. The operator actions are less significant than those required to mitigate a steam generator tube rupture. However, the timeframe for the E-0, Revision 7 operator action is less than the typical ten minute operator action time allotted in the FSAR. As such, E-0 Revision 7 may have been constituted a potential USQ.

4. Measuring and Test Equipment (M&TE) Problems

The inspection report (at page 14) contained several findings in the area of M&TE problems in the American Society of Mechanical Engineers (ASME) Section XI program. The findings involved specific questions regarding the impact of pulsations in the pressure measurements on differential pressure determinations and more general questions regarding the validity of previous ASME Section XI test results.

TVA acknowledges that the pressure pulsations did call into question the validity of the pressure data for use in determining pump-to-pump differential pressures. Two basic problems were noted with the pump data. First, when data on a single pump was compared from one run to the next, there was significant data scatter. RHR Pump 1A-A showed up to 14 psid between performances with an average of approximately 7 psid. RHR Pump 1B-B showed up to 11 psid with an average of 4 psid. Second, when comparing single-point data over the time period following the Unit 1 Cycle 3 refueling outage, seven instances were seen when the pump-to-pump data would pass the 11.1 psid criteria and ten instances where it would not. When all data points since the last refueling were averaged in accordance with SI-754.1, the value computed was between 12 and 13 psid. When just the data since restart were averaged, the pump-to-pump differential became more pronounced as indicated in the memorandum to NE from the system engineer dated November 27, 1989. By the end of Friday, December 1, Technical Support was not able to finalize its evaluation because of doubts about both the accuracy of the data and the methods established in SI-754.1 to evaluate deadheading. Technical Support proposed revising SI-754.1 to run a field test to validate operability and to establish the actual conditions under which a pump would deadhead. The duty plant manager was contacted by the System Engineering supervisor and the acting Technical Support Manager to review the status and to discuss whether the issue needed to be pursued over the weekend. The duty plant manager was advised that the data was confusing and that Technical Support did not believe it warranted calling the pumps inoperable.

TVA also reviewed the ASME, Section XI, program to determine the cause for recent problems in ASME, Section XI, testing. TVA concluded that the program is in overall compliance with requirements. However, several recommendations were made regarding the use of snubbers on test instruments to reduce data scatter, use of dedicated test equipment for ASME, Section XI, testing, and potential upgrade of installed instrumentation as an alternative to test equipment. The NRC senior resident inspector reviewed this report during routine monthly activities in January 1990. No specific safety or technical issues were identified during the review, as reported at the exit meeting.

5. Date of NRC Inspector Involvement

The inspection report (at pages 15 and 17) notes that the resident inspector identified the RHR pump deadheading issue on November 29, 1989.

Several activities occurred prior to confirming a Unit 1 RHR deadheading problem. During the week of November 20, 1989, the RHR system engineer reviewed the Surveillance Instruction (SI) 754 series surveillance in preparation for the Unit 1 Cycle 4 refueling outage. SI-754 had been prepared in response to Bulletin 88-04 and is discussed in detail below. The system engineer determined that the acceptance criteria would not be met. Failure to meet the acceptance criteria of 8 psid required notification to and evaluation by Nuclear Engineering (NE). This finding was documented and discussed with his supervisor on November 22, 1989. As required by SI-754, NE was notified of the data and the fact that the acceptance criteria would not be met by the correspondence dated November 27, 1989. At this point, the system engineer was not familiar with the problems identified in Bulletin 88-04 and did not recognize the significance of not meeting the acceptance criteria.

On November 28, 1989, during the performance of SI-128.4, "Residual Heat Removal Pump 2A-A Quarterly Operability Test," RHR pump 2A-A exceeded its developed head criteria as specified in the SI. Additional testing was performed, and the pump was still exceeding the differential pressure requirements. The pumps were determined to be acceptable/operable in accordance with ASME Section XI, pump testing requirements on November 29, 1989. Additionally, NE was concerned with the potential of a pump-to-pump interaction because of the increased head on the 2A-A pump. An engineering evaluation was performed that demonstrated the differential pressure between A and B train pumps did not exceed the 11.1 psid limit established for the NRC bulletin response. As a result of the disposition of the Unit 2 pump problem, the NRC resident inspector questioned the System Engineering supervisor on November 30, 1989, about the Unit 1 RHR pump-to-pump differential head data. The NRC inspector was informed that NE was evaluating the Unit 1 data based on the November 27 notification by the system engineer. Also, Technical Support began formulation of an action plan for further evaluation of the Unit 1 RHR data in which significant scatter had been observed. The data scatter, while not significant for evaluating differential pressure across the pump (approximately 200 psid), did cause the system engineer to question the validity of the data for use in determining pump-to-pump differential pressures (approximately 10 psid).

6. Bulletin 88-04 Response

The Inspection Report (at pages 15, 16, and 17) contains a number of findings related to the adequacy of the response for SQN for Bulletin 88-04. These findings discuss the adequacy and validity of the calculation prepared by NE to support the response, the review of RHR pump test data for pump-to-pump differential pressures, and the misuse of the calculation and vendor information is developing corrective actions.

TVA agrees that the response to the Bulletin for SQN was inadequate.

The root cause of this event was an inadequate program review of the data in preparing the response to NRC Bulletin 88-04. Evaluation of and response to this bulletin were handled by Knoxville NE. These individuals were not sensitive to the implications of test data and therefore limited their evaluation of acceptability to the most recent data point. Coordination and communications with the knowledgeable site organizations were not effectively employed.

To support evaluation of the bulletin concern, a calculation had been prepared in Knoxville addressing the permissible head differential between RHR pumps and temperature rises for various flow rates. The results of the analysis were:

1. Eleven minutes to exceed design temperature at deadhead (no flow) conditions.

2. Flows as low as 50-100 gpm prevent excessive temperature rise in the pump.

3. A relationship between weak pump flow and differential heads (e.g., 0 gpm for greater than 12.6 psid, 100 gpm for 11.1 psid, and 200 gpm for 8.9 psid).

The vendor was also contacted regarding operation with lower recirculation flows. The vendor established operations at 100 gpm acceptable for no more than 20 minutes and continuous operation at 500 gpm. As a result, 11.1 psid and 20 minutes were selected as key criteria (with corresponding weak pump flow at 100 gpm).

The most recent RHR pump data (second quarter 1988) was checked for head differential and determined acceptable (pump-to-pump differential head was 10.5 psid). Accordingly, actions at that time were taken to address a potentially low flow condition (100 gpm), not a deadheading condition. The emergency procedures were reviewed and judged adequate to ensure parallel pump operation on miniflow would be terminated within 20 minutes. SOIs were revised to limit parallel pump operation to 20 minutes. A surveillance program was developed to monitor and assess the pump-to-pump interaction.

The monitoring program used 8 psid as a trigger to initiate an evaluation by NE. The program averaged pump data collected between refueling outages to eliminate statistical scatter. The procedure was scheduled to be performed at the Cycle 4 refueling outage.

TVA has reviewed the response to Bulletin 88-04 and the calculation prepared for the response. The calculation was determined to be valid and adequate. However, there was a lack of sensitivity to test data when only the most recent set of data was used to determine the condition of the pumps.

As NRC is aware, TVA has previously instituted major programmatic changes that improved the methods used to manage significant licensing issues (e.g., NRC bulletins) and used in the preparation of licensing correspondence. The Licensing project management system was instituted in January 1989. In particular, Licensing has lead responsibility for NRC Bulletins. A licensing project manager (LPM) is assigned to each issue and has the responsibility to develop detailed action plans defining tasks, scope, schedules, and responsibilities. Information provided by the various organizations is handled through the formal licensing information request process that has the necessary controls to ensure completeness and accuracy. The LPM initiates and drives bulletin investigations and evaluations. Specific condition adverse to quality reports (CAQRs) are written when deficiencies are identified.

TVA conducted a review of the 1987-1989 SQN Bulletin responses and records and concluded that the responses are appropriate and valid with the single exception, Bulletin 88-04. The review also established that CAQRs were written when deficiencies were identified, with the exception of a single prerestart item that was tracked as a specific nuclear performance plan restart item, rather than a CAQR.

Several other programmatic changes had been made at SQN that will have a positive impact on the resolution of licensing issues. The SQN NE Project Engineer has responsibility for and control of all engineering related to SQN. In addition, 90 percent of all engineering is now done at the SQN site. Technical Support is now the focal point at the plant for all system-related problems. The close proximity and direct involvement of these organizations contribute to a more systematic and thorough evaluation of plant problems.

7. Condition Adverse to Quality Reporting (CAQR) Program

The inspection report (at pages 14, 16, 17, and 23) contained findings related to the CAQR program at SQN. In particular, there were questions regarding the lack of timely initiation of CAQRs for the potential M&TE problems. NRC also noted that a draft CAQR regarding the potential RHR deadheading problem was not processed and that if it had it would have solved interorganizational communication problems that appeared to affect the initial corrective actions for the bulletin. NRC questioned the lack of a CAQR for potential problem identified by the system engineering late November 1989. General concerns were identified regarding the apparent reluctance of the SQN staff to promptly initiate and process CAQRs.

TVA acknowledges that potential CAQRs were not initiated as part of the investigation of ASME Section XI M&TE problems. Potential CAQRs can be written but are not required to be written by the CAQR program requirements. TVA contends that the initiation of a potential CAQR would not have changed the approach to investigating the M&TE questions. The correct organization (Technical Support) was promptly investigating the questions with an emphasis on resolving any potential operability concerns.

The draft CAQRs prepared when the bulletin was received were not processed when it was determined that SQN did not have the RHR deadheading problem. The error was in the technical evaluation that dispositioned the problem not in the failure to process the draft CAQR. It should be noted that TVA did make an operability determination when it concluded that the deadheading problem was not present. While the evaluation was flawed, operability was assessed, contrary to the conclusions on the inspection report.

The inspection report also discusses the fact that the draft CAQR addressed the time to damage a deadheaded pump and recommended that the pump vendor be contacted. NRC implies that if the draft CAQR had been issued these actions would have lead to a correct evaluation. In fact, the TVA calculation prepared for the bulletin response calculates the time to pump damage and TVA did contact the vendor and use the information supplied regarding low flow rates and operating times. In addition, TVa believes the organizational changes made in Licensing, Nuclear Engineering, and Technical Support are the correct solution to the interorganizational communication problems associated with Bulletin 88-04.

A potential CAQR was not initiated by the system engineer in late November 1989. Instead, he chose to follow the direction of Surveillance Instruction (SI)-754.1 and requested NE to evaluate the potential problem with pump-to-pump differential pressure data. A CAQR was written immediately after the RHR parallel operation test confirmed that deadheading did exist. NRC staff questioned why a CAQR was not written earlier. The CAQR process is not a substitute for operability consciousness. Rather, it is a program designed to document known conditions adverse to quality to ensure tracking of corrective actions and trend analysis. The current TVA CAQR process is not user friendly for identifying potential problems. TVA revised the process in the spring of 1989 to include a problem report document (PRD) designed to promote potential problem identification. TVA recognizes that further improvements are needed in raising potential issues, but we do not believe this is the cause of this problem.

TVA is not satisfied with the performance of the Technical Support organization, particularly, in assessing the operability of the RHR pumps with respect to the deadheading problem. The evaluation focused on the RHR pump data scatter rather than pump-to-pump differential pressure, and development of the test procedure was slow. However, the overall performance problems were previously recognized and, as NRC is aware, major changes were already being made.

Major reorganization and management changes were made during the June to October 1989 timeframe. A new Technical Support manager with experience in managing a successful system engineering program was hired from outside TVA. The Institute of Nuclear Power Operations was requested to perform a peer evaluation of the Technical Support area during the November plant evaluation. A complete assessment of Technical Support was recently presented to senior management outlining planned improvements.

TVA is actively recruiting experienced system engineers and has established a lead engineer concept to best utilize existing strengths. A formal reevaluation of existing supervisors and engineers is being conducted. In addition, the key elements of System Engineering (ownership of problems, leadership in problem resolution, sensitivity to regulatory and operational aspects, and focus on operability and problem solving) will continue to be reinforced through direct involvement of Technical Support management. These changes were intended to further strengthen the system engineering capabilities at SQN.

As NRC is aware, recent management actions had been taken to change personnel attitudes about problem reporting and to encourage problem identification to supervisors. Two site dispatches have been issued from the Vice President, Nuclear Power Production, promoting these expectations. This topic was also emphasized in the recent Site Director's quarterly meetings held with over 1,300 SQN employees (16 meetings). The CAQR management review committee has been restructured to include senior site management to assess the extent to which the message regarding problem reporting is being understood, to ensure prompt corrective actions to identified problems, and to ensure quick resolution of potential problems. A multisite task force was formed in early November 1989 to evaluate the implementation and structure of the problem identification process and recommend improvements. The Vice President, Nuclear Power Production, was briefed on the team recommendations in January 1990. The planned changes to this program include utilization of a single problem reporting document, a lower threshold for incident investigations, and the establishment of criteria to ensure the appropriate resolution of potential problems. The change to utilize a single problem reporting document will be completed by June 1, 1990.

SUMMARY

It is clear from the above discussion that TVA's performance in identifying the RHR pumps deadhead condition was inadequate. However, the underlying causes of this problem had been previously identified, thoroughly evaluated, and corrective actions were ongoing at the time of this event. Specific efforts towards improvement in these areas included emphasizing timely identification and resolution of problems, upgrading the Technical Support organization, 10 CFR 50.59 program improvements and Licensing submittal improvements (including bulletin responses). Prompt corrective action was also taken by TVA in two additional areas: the emergency procedure change process was strengthened and specific training regarding cause and effect on FSAR assumptions was provided to all 50.59 preparers and reviewers.