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May 6, 1981

2CAN058102

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert A. Clark, Chief
Operating Reactors Branch 3
Division of Licensing
U.S. Nuclear Regulatory Comm.
Washington, D.C. 20555

SUBJECT: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Responses to NRC Question on the ANO-2,
Cycle 2 Reload Report
(File: 2-1510)

Gentlemen:

In response to your letter of April 10, 1981, which contained NRC staff questions on the ANO-2, Cycle 2 reload report, the information in Attachment A is provided. Advance copies of all but two of the responses in Attachment A were informally provided to Mr. Bob Martin of your staff on April 21 and April 23.

In response to your letter of April 23, 1981, which contained additional questions on the ANO-2 reload report, the information in Attachment B is provided. Parts of this information were discussed with PNL (Pacific Northwest Laboratories) and NRC personnel at a meeting on April 14 and 15. The remaining parts of this information were informally provided to PNL on May 1, 1981. The responses have been recorded in a report entitled "Responses to Second Round Questions...(EN-139-(A)-P and (EN-124-(B)-P Part 2". Portions of this report are proprietary and it is requested that they be classified as proprietary information in accordance with the provisions of 10 CFR 2.790. The appropriate affidavit from Combustion Engineering is attached. For your convenience eight (8) copies of the proprietary version and four (4) copies of the non-proprietary version of this report are provided.

Very truly yours,

Donald A. Rueter
for David C. Trimble,
Manager, Licensing

DCT:kb

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Attachments

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ATTACHMENT A

PART 1 - RELOAD REPORT

Q-1 (4.1) Discuss whether or not the internal fuel rod pressures are predicted to be equal to or greater than coolant pressure throughout Cycle 2.

Answer The internal fuel rod pressure is one of the performance parameters which is always calculated for all the fuel batches in a reload core. Internal pressure will increase with burnup due to accumulation of fission gases and a decrease in internal void volume the internal gas pressure is predicted by FATES using the NRC fission gas release enhancement factor to be significantly less than the nominal RCS pressure throughout ANO-2 Cycle 2.

Q-2 (4.2) Discuss your plans and procedures for the submission of CEA guide tube surveillance results pursuant to license condition number 2.C.3.1. Note that this is required prior to startup of Cycle 2.

Answer AP&L's ANO-2 CEA Guide Tube Surveillance Program was described in a March 30, 1981 letter from D. C. Trimble to R. A. Clark. It is anticipated that the preliminary results of this inspection will be available after the fuel shuffling sequence and will be transmitted verbally to the NRC at that time. Fuel shuffling is expected to be completed by May 1, 1981. A written report will be submitted before Cycle 2 criticality.

Q-3 (7.0) Discuss your plans for the submission of the CESEC code verification information pursuant to license condition 2.C.3.g. This information should be provided to support the use of CESEC - II code.

Answer The CESEC information was submitted by AP&L to the NRC on March 27, 1981.

Q-4 (7.1.4) Discuss the value of initial steam generator pressure in Table 7.1.4-1 for Cycle 2 relative to the expected values for operation.

Answer The initial steam generator pressure shown in Table 7.1.4-1 is based on the use of the initial core inlet temperature of 540 F, also shown in Table 7.1.4-1. This value is taken from Table 7-3 (Allowable Initial Conditions for Safety Analysis). The combination of RCS inlet temperature, pressure, and initial core power level determines the initial steam generator pressure. The lower initial RCS pressure, inlet temperature and steam generator pressure maximize the peak RCS pressure and secondary pressure after trip. Expected steam generator pressure during Cycle 2 operation is approximately 923 psia.

Q-5 (7.1.8) Discuss the degree of similarity of the methods used, computer codes used, etc., in the Cycle 1 analysis relative to the Cycle 2 analysis, i.e., define "consistent". Are the Cycle 2 methods and codes identical to the Cycle 1 methods and codes, except for CETOP?

Answer The methodology used in Cycle 2 to analyze the Loss of Flow Event is the same (consistent) as the methodology used in Cycle 1. However, in Cycle 2, as stated in Section 7.1.8, the CETOP code was used to calculate DNBR and, as stated in Section 7.0, the CESEC II code was used to calculate the NSSS transient response.

Q-6 (7.2.3) The time required for the automatic initiation of emergency feedwater pump operation had also been the subject of AP&L Co., letters dated August 1, 1980 and July 4, 1979. Please clarify the value of the parameter specified in Technical Specification Table 3.3-5 Items 7a and 8a which were used in all safety analysis. Identify and discuss the intervals of time which when added together result in the value specified in Table 3.3-5. Address the 65 second and 118 second values in your August 1, 1980 letter relative to the values currently specified in the Technical Specifications.

Answer ANO-2 Technical Specification Table 3.3-5 Items 7a and 8a show only the Response Time of the system. This is not the value used in the safety analyses. The values used in the safety analyses are equal to or greater than (if conservative) the values listed in Table 3.3-5 Items 7b and 8b. No credit is taken in any safety analysis for Response Times shorter than 7b and 8b.

Upon initiation of an ESFAS, all loads are shed from the safety busses and sequenced back on as required by the ANO-2 Millstone modifications. In the case of the motor driven EFW pump (2P7B) the breaker is closed in at 90 seconds from the initiation of the ESFAS with 7.4 seconds required to deliver flow resulting in a 97.4 second response time. If offsite AC power is not available, an additional 15 seconds is required for the diesel generator to start and close in on the buss before the 90 second clock begins resulting in a total response time of 112.4 seconds.

The original ANO-2 analyses used an EFW response time of 65 seconds for transients with offsite AC and 118 seconds for transmittal without offsite power. Our letter of August 1, 1980, explained that the "with offsite power" analyses had been reanalyzed to accomodate a response time of 97.4 seconds. This change was made to better accomodate diesel generator loading as a better loading "window" existed at 90 seconds vice 65 seconds. For transients without offsite AC power, all analyses were originally performed using a response time of 118 seconds which is conservative with respect to the 112.4 seconds currently in the Technical Specifications.

Q-7 (7.2.5) Identify the number of pins expected to experience DNB due to the Seized RCP Shaft event for the Cycle 1 analysis and the Cycle 2 analysis.

<u>Answer</u> For ANO-2 Cycle 1 the number of pins calculated to experience DNB due to the Seized RCP Shaft event was 789 fuel pins. For ANO-2 Cycle 2 the number of pins calculated to experience DNB due to the Seized RCP Shaft event is 512 fuel pins.

Q-8 (8.0) Reference the letter of approval for the latest version of each of the C-E topical reports and codes listed in Sections 8.0 through 8.5.

<u>Answer</u> Table E-1 identifies the NRC letters of approval for each of the C-E topical reports and codes listed in Section 8.0 through 8.5 (ECCS Analysis) of the ANO-2 Cycle 2 Reload Report.

TABLE E.1

C-E TOPICAL REPORTS LISTED IN SECTIONS 8.0 THROUGH 8.5

	NRC APPROVAL LETTER
CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model", August, 1974 (Proprietary).	1
CENPD-132, Supplement 1, "Calculational Methods for the CE Large Break LOCA Evaluation Model", February, 1975 (Proprietary).	1
CENPD-132, Supplement 2, "Calculational Methods for the CE Large Break LOCA Evaluation Model", July, 1975 (Proprietary).	2
CENPD-135, "STRIKIN-11, A Cylindrical Geometry Fuel Rod Heat Transfer Program", August, 1974 (Proprietary).	1
CENPD-135, Supplement 2, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modifications)", February 1975 (Proprietary).	1
CENPD-135, Supplement 4, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program", August, 1976 (Proprietary).	4
CENPD-135, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program", April, 1977 (Proprietary).	6
CENPD-139, "CE Fuel Evaluation Model", July, 1974 (Proprietary).	3
CENPD-138, "PARCH - A FORTRAN-IV Digital Program to Evaluation Pool Boiling, Axial Rod and Coolant Heatup", August, 1974 (Proprietary).	1
CENPD-138, Supplement 1, "PARCH - A FORTPAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup" (Modifications), February, 1975 (Proprietary).	1
CENPD-138, Supplement 2, "PARCH - A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup", January, 1977 (Proprietary).	2

NRC APPROVAL LETTERS

1. Letter, O. D. Parr (NRC) to F. M. Stern (CE), June 13, 1975.
2. Letter, O. D. Parr (NRC) to A. E. Scherer (CE), December 9, 1975.
3. Letter, O. D. Parr (NRC) to F. M. Stern (CE), December 4, 1974.
4. Letter, K. Kniel (NRC) to A. E. Scherer (CE), November 12, 1976.
5. Letter, K. Kniel (NRC) to A. E. Scherer (CE), April 10, 1978.
6. Letter, R. L. Baer (NRC) to A. E. Scherer (CE), September 6, 1978.

Q-9 (8.3) Provide a figure or table showing the values of the steam cooling that transfer coefficient versus axial location for the Cycle 1 analysis and for the Cycle 2 analysis.

Answer The steam cooling heat transfer coefficients (HTC) versus axial location for the Cycle 1 and Cycle 2 analyses are provided in Figure E.1. Steam cooling HTC are applied to the hot rod at and above the rupture location after the core reflood rate first falls below one inch per second (1"/sec). For both Cycles 1 and 2, cladding rupture was predicted to occur at axial mode 15 and the core reflood rate feil below 1"/sec at 71 seconds.

As described in the ANO-2 Cycle 2 Reload Report (Reference 1), the Cycle 2 steam cooling HTC were calculated using the approved PARCH code (Reference 2). For Cycle 1, the steam cooling HTC were calculated using the PARCH code only at the rupture location. Above the rupture location, minimum steam cooling HTC, as described in Reference 3, were conservatively used.

In the Cycle 2 analysis, conservatively generated PARCH steam cooling HTC were used. Otherwise, the PARCH steam cooling HTC for axial mode 15 (rupture location) would have been essentially equivalent in both analyses.

REFERENCES

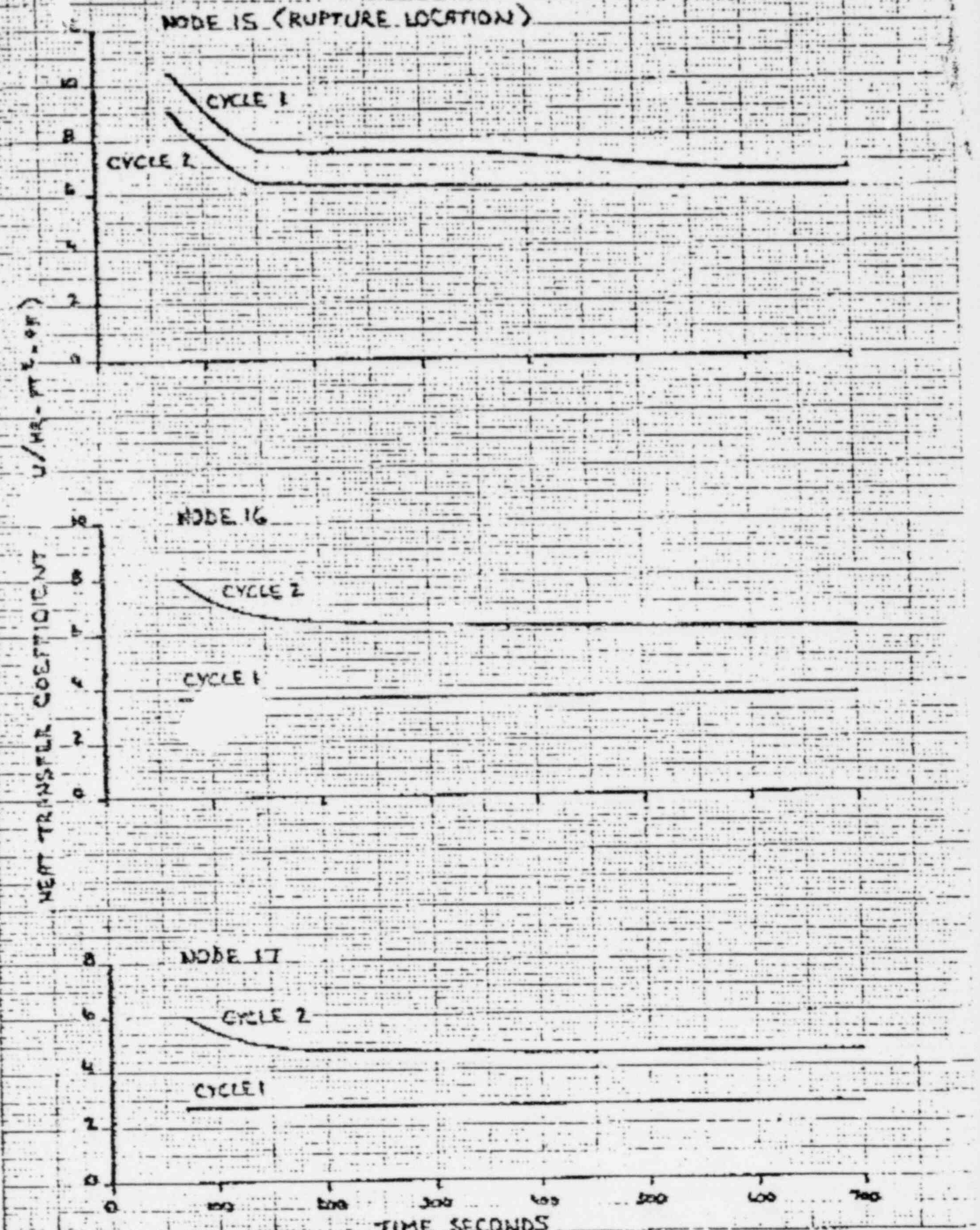
1. Letter, D. C. Trimble (AP&L) to R. A. Clar (NRC), Februray 20, 1981.

2. CENPD-138, PARCH-A FORTRAN-IV Digital Program to Evaluate Pool Poiling. Axial Rod and Coolant Heatup, August, 1974; Supplement 1, February, 1975; Supplment 2, January, 1977 (Proprietary).

3. CENPD-132, Supplement 1, Calculational Methods for the C-E Large Break LOCA Evaluation Model, February, 1975 (Proprietary).

STEAM COOLING

HEAT TRANSFER COEFFICIENTS



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Q-10 (11.0) Comments on proposed Technical Specification changes are identified below by their change number as TS page number as they appear in Section II of the Reload Report.

<u>Answer</u> The following are item by item responses to your Question 10 items.

9 (2-4) Address the effects of the set point change on those events which are calculated to reach the Linear Power Level - High trip setpoint.

Answer The Linear Power Level-High is the leading trip for the ejected CEA accident. In this case reducing the trip setpoint mitigates the consequences of this postulated accident.

9 (2-4) Discuss the uncertainties associated with the Fisher Porter versus the Rosemount transmitters which necessitated these setpoint changes. List the trip value required by the safety analyses followed by the various uncertainty contributions necessary to arrive at the instrument trip setpoint.

Answer The Fisher Porter setpoints were based on separate effects type environmental qualification testing vice sequential testing for the Rosemounts. The Rosemount transmitters were qualified and placed into service based on an AP&L specific environmental qualification test performed in 1978. Setpoints calculated for the Rosemount transmitters based on the AP&L specific test revealed the existing Technical Specification setpoints for the Fisher Porter transmitters were conservative for use with the Rosemounts and thus the setpoints were not changed.

Subsequent to the Fisher Porter/Rosemount change out, a second set of setpoints were calculated for the Rosemounts based not only on the AP&L specific test but also a more concervative generic sequential test as well. These new setpoints including the generic data were calculated to further insure reliability of the transmitters and the margin of safety. By using the data from both tests, a data base from several transmitters was used vice a data base from the one transmitter used in the AP&L specific test. Use of the conservative data naturally resulted in more conservative setpoints.

Our letter of February 28, 1979 transmitted to you CEN-98(A)-P Revision 01-P which documented the methodology used for calculation of PPS setpoints for ANO-2. The new Rosemount setpoints were calculated in accordance with this methodology which was previously approved by NRC.

9 (2-4) The pressure-high trip setpoint was also the subject of your letter dated November 27, 1979 wherein you proposed an increase from 2345 to 2368 to eliminate a dynamic allowance imposed prior to operation. Does the now proposed value of 2362 reflect deletion of that dynamic allowance or is it solely due to the Fisher Porter/Rosemount transmitter changeout?

If the value of 2362 relfects deletion of the dynamic allowance please describe the physical phenomena associated with this allowance and provide justification for its deletion.

Answer The proposed pressure-high trip setpoint of 2362 psia refects deletion of the dynamic allowance and a change in instrument uncertainties, both due to the Fisher Porter/Rosemount transmitter changeout.

The 2345 psia setpoint incorporated a dynamic allowance to compensate for the fact that the actual Fisher Porter equipment response time was longer than the time used in the Safety Analysis. To assure that protective action occurs at or before the time assumed in the Safety Analysis, the setpoint was altered in a conservative direction to compensate for the sensor response characteristics.

The Rosemount transmitter response time is short enough to be within the time assumed in the Safety Analysis, thus the dynamic allowance has been deleted from the setpoint calculation. The setpoint of 2362 psia reflects this deletion and also reflects the changes in equipment uncertainties due to the change to the Rosemount transmitter.

25, 26 (3/4 2-1 3/4 2-2 3/4 2-3) It is understood that COLSS displays a power operating limit value in percent (or percent over power) which could be achieved without violating the steady state limit of 14.5 KW/ft. and that COLSS does not display values of KW/ft. Therefore, with COLSS in service the new proposed Figure 3.2-1 is governing. With COLSS in service the CPCS may display a KW/ft. value in excess of 14.5 based on the greater uncertainties in the CPCS than in COLSS. With COLSS out of service, operation is governed by the CPC KW/ft. output which must be implemented per the proposed Figure 3.2-2.

It is understood that Technical Specification change #25 does not change in any way the manner in which the plant is operated or the safety margins which are maintained but does provide clarification. Confirm that the above understanding is correct in each of its elements or provide needed clarifications.

Answer There do not appear to be any flaws in the above description.

29 (3/4 2-7) As has been discussed in previous correspondence to AP&L Co., the Cycle 1 rod bow penalty defined in Technical Specification 4.2.2.2 shall continue in effect until an alternate penalty is justified.

<u>Answer</u> See answer to question 492.31 submitted to NRC on 4/1/81.

36 (3/4 3-2, 3,4,5) The request for extended bypass times for one RPS channel will be addressed by the staff on a schedule independent of the Reload Report Schedule.

<u>Answer</u> No response required.

40 (3/4 3-18) Justify the change in the Steam Generator differential pressure trip setpoint from 39 to 90 psi. Discuss the safety analyses in the FSAR and the reload report which are based on this trip setpoint. Provide the value of the trip setpoint utilized in the safety analysis and an explanation of the derivation of the proposed new value.

Answer The Steam Generator differential pressure setpoint shown in Table 3.3-4 of the Technical Specifications is used by the Engineered Safety Feature Actuation System (ESFAS) logic to identify the intact steam generator during a Steam Line Break or Loss of Main Feedwater events. It is not a reactor trip setpoint. The logic ensures that during a Steam Line Break or Loss of Main Feedwater event an Emergency Feedwater Actuation Signal is initiated on the intact steam generator. The results of the analysis are not sensitive to this setpoint value.

The Safety Analysis originally analyzed an analysis setpoint of 100 psid. Application of equipment uncertainties resulted in a trip setpoint of 39 psid. Subsequently, the Fisher-Porter pressure transmitters were replaced with Rosemount transmitters. For this particular application and for the environmental conditions applicable, the Rosemount transmitters have larger equipment uncertainties than Fisher-Porter. As a result, the Safety Analysis was supplemented to verify that acceptance criteria would also be met with an analysis setpoint of 180 psid. When the instrument uncertainties (including the Rosemount transmitter) are now applied, the resulting trip setpoint is 90 psid to ensure a trip before 180 psid.

41 (3/4 4-1) The Reload Report, by stating that "part loop operation has not been approved by the NRC", may imply that application was made for part loop operation. Therefore it should be clarified that the application and bases for part loop operation has not been submitted on the ANO-2 docket.

Answer An application for part loop operation has not been submitted on the ANO-2 docket.

44 (3/4 42) Discuss how the application of the proposed MODE 3 Technical Specification supports the Steam Line Break Analysis.

Answer The one-loop hot standby (zero power) Steam Line Break Analysis was not reanalyzed for Cycle 2. The proposed MODE 3 Technical Specification supports the Steam Line Rupture analyses by preventing one-loop operation at hot standby.

47 (3/4 7-5) Reference and/or provide the sections of ASME Section XI and the ANO-2 Inservice Inspection and Testing Plan which govern testing of the EFW pump. Does the plan require the monthly testing to verify a specified flow rate at a specified discharge pressure?

Answer ASME Section XI Inservice Testing (IST), subsection IWP-1000, applies to class 1, 2 and 3 pumps which are installed in water-cooled nuclear power plants and which are provided with an emergency power source. The EFW pumps are class 3 pumps; however, only the ANO-2 electric driven EFW pump (2P7B) has an emergency power supply per se. Consequently, the steam driven EFW pump (2P7A) is not directly a requirement of ASME Section XI.

Plant procedures have for some time tested 2P7A under ANO's ASME Section XI testing program. Procedure 1015.06 documents this fact, and a copy of the applicable sections of this procedure are enclosed for your review. This procedure requires monthly testing of 2P7A per procedure 2106.06 which in turn has specific requirements upon flow rate and discharge pressure. A copy of the section of this procedure pertaining to the monthly testing of 2P7A is enclosed for your reference.

The ANO-2 Inservice Testing program is documented in AP&L letter to NRC 2-068-17 dated June 15, 1978. This letter does not commit AP&L to testing 2P7A under the Section XI IST program. However, AP&L will continue to test 2P7A under the Section XI program and will include this commitment in our next periodic update of the program to NRC.

1.0 PURPOSE

The purpose of this procedure is to:

1.1 Describe the methods used to implement, and update the ASME Section XI Inservice Test Program for pumps and valves.

1.2 Define the responsibilities of ANO personnel associated with the Inservice Test Program and for implementing procedures development and revisions.

2.0 SCOPE

This procedure defines methods and responsibilities associated with the Inservices Test Program of pumps and valves at Arkansas Nuclear One. The Inservice Test Program complies with requirements contained in ASME, B&PV with exceptions specifically authorized by the NRC.

3.0 REFERENCES

3.1 References used to develop this procedure:

3.1.1 ASME B&PV Code, Section XI, 1974 edition, summer 1975 addenda.

3.1.2 APL-TOP1A, Rev. 5, "Quality Asssurance Manual Operations"

3.1.3 AP&L Memo 1-019-6, Manager of Licensing to NRC dated 1/15/79

3.1.4 AP&L Memo 2-063-017, Manager of Licensing to NRC dated 6/15/79

3.1.5 Plant Technical Specifications

3.2 References to be used to implement the requirements of this procedure:

3.2.1 1001.01, "Scheduling or Routine Station Activities"

3.2.2 1000.09, "Surveillance Test Control"

3.3.3 1000.08, "Regulatory Reporting and Communications"

3.3 Related ANO Administrative Procedures

None

4.0 DEFINITIONS

Inservice Test Program

The Inservice Test Program is a list of pumps, valves and general testing criteria that meets the requirements of ASME B&PV Code, Section XI, Subsection IWP and IWV with NRC approved modifications.

5.0 RESPONSIBILITIES

5.1 Plant Analysis Superintendent

5.1.1 Assign a qualified individual(s) to act as Pump and Valve IST Coordinator(s).

5.1.2 Provide independent technical review and approval of all Program submittals of the Pump and Valve IST Program to the Licensing group. Identify to the responsible individuals the need for performance test criteria revisions necessitated by changes in analysis and systems performance requirements.

5.1.4 Provide interface with AP&L General Office Engineering. Submit request for relief from ASME Seciton XI Code Requirements with justification to the Licensing Manager.

5.1.5 Establish testing requirements necessary to provide new performance data following replacement, maintenance or other modifications of ASME Section XI related components or their systems which could affect system performance characteristics as may be deemed necessary.

5.2 IST Coordinator(s)

5.2.1 Maintains the cross reference between the ASME Section XI pumps and valves and the plant procedure in which these test requirements are incorporated (Attachments 1 and 2 of this procedure).

5.2.2 Ensure the establishment and maintenance of a file for each ASME Section XI affected pump or valve. These files should contain:

(1) Any necessary calculations or other information relating to the limiting values for operability of the associated components;

(2) A tabulation of the components performance data (usually in a graph type format) for use in trend analysis and;

(3) Any information pertaining to special Section XI relief requests, NRC correspondence or other information relating to the pump or valve which affects its Section XI status.

5.2.3 Periodically Review selected performance data associated with the ASME Section XI pump and valve tests for trending purposes and assist in identifying needed in-depth assessment or corrective action as indicated by this review.

5.2.4 Review recommended additions, deletions and/or justifications for changes to the IST program.

5.2.5 Inform the appropriate supervisors of the necessity for procedure changes required as a result of revisions to the IST Program.

5.2.6 Prepare and coordinate the 20 month IST program submittal to the NRC.

5.3 Planning and Scheduling Supervisor

Procedures developed to perform IST of pumps and valves required by ASME Section XI shall be incorporated into the ANO Surveillance Test Program. The Planning and Scheduling Supervisor is responsible for scheduling those surveillances and tests as required by 1000.09 "Surveillance Test Control".

5.4 Plant Engineering Superintendent

5.4.1 Assist the Operations Superintendents, I&C Superintendent and the ISI Coordinator in developing pump and valve test procedures or assist in incorporating section XI pump and valve tests into the ANO surveillance test procedure.

5.4.2 Identify all plant and design changes which may impact the pump and valve test program or its implementing procedures to the IST Coordinator, OPS Superintendent and I&C Superintendent.

5.4.3 Assist the IST Coordinator(s) in justifying requests for relief from ASME Section XI Code requirements.

5.5 Operations Superintendents

5.5.1 Submit input to the pump and valve IST Coordinator concerning changes in plant operation that affect the testing of those pumps and valves for their units as listed in attachment 1 or 2 of this procedure.

5.5.2 Provide review of operations surveillance test results to ensure those activities associated with ASME Section XI IST are performed and reviewed as required.

5.5.3 Incorporate the pump and valve testing requirements into those surveillance test procedures for which the Operations Department is directly responsible.

5.6 I & C Superintendent

5.6.1 Develop the procedures to perform Leak Rate tests required by ASME Section XI or incorporate the required leak rate tests into existing plant surveillance test procedures.

5.6.2 Review the completed data sheets associated with Leak Rate tests required by ASME Section XI to ensure program compliance.

5.7 Mechanical Engineering Supervisor

5.7.1 Develop the procedures necessary to verify setpoints of safety and relief valves subject to the IST program or incorporate the required setpoint verifications into existing surveillance test procedures.

5.7.2 Provides justification to the IST Coordinator for requests for relief from ASME Section XI requirements which are not practical or applicable.

5.7.3 Review completed data sheets associated with Section XI relief valve tests to ensure compliance with stated criteria.

5.8 Shift Supervisor

5.8.1 The Shift Supervisor shall identify to the IST Coordinator new baseline data for Section XI components generated as required following equipment maintenance or other plant system changes. A copy of the test data from the specific surveillance test run for baseline should be forwarded to the IST files conspicuously marked as being baseline data following maintenance/modifications.

6.0 REQUIREMENTS

6.1 IST Program Development

The AP&L/ANO Inservice Testing (IST) Program for pumps and valves subject to ASME B&PV Code, Section XI, 1974 Edition, Summer 1975 Addenda was developed through the joint effort of the ANO Operations Superintendent, Plant Analysis Superintendent, I&C Supervisor and Mechanical Engineering Supervisor. The AP&L Licensing Department has submitted the IST Program for ANO Units I and II to the Nuclear Regulatory Commission for their review and approval. The IST Program for ANO Unit I received interim approval by the NRC on September 13, 1978. Final approval on both units is currently pending.

6.2 IST Program Implementation

The pump and valve tests required by Section XI of the ASME Code shall be maintained in the surveillance test procedures as required by ANO Technical Specifications and scheduled thru ANO Surveillance Test Program. Cross reference between the pumps and valves subject to ASME Section XI and the ANO test procedure is included in Attachments 1 and 2 of this procedure.

6.3 General Requirements

6.3.1 The Plant Analysis Superintendent shall select an individual (s) to perform responsibilities of the IST coordinator(s) (see Section 5.2).

6.3.2 IST Coordinator(s) is responsible for maintaining documentation of the basis for the limiting values for operabiltiy applied to Section XI pumps and valves, and for initiating action to calculate those valves for new components added to systems which are subject to Section XI.

6.3.3 The Operations Superintendent(s) shall ensure procedures are developed as necessary to perform IST of pumps and valves to meet the requirements of ASME Section XI, subsection IWP and IWV as set forth in the approved IST Program submittal.

6.3.4 The Operations Superintendents shall prepare procedures (as necessary) to perform IST of pumps and valves to meet the requirements of ASME Section XI, subsection IWP and IWV (except as ammended by the NRC approved IST Program submittal).

6.3.5 The Mechanical Engineering Supervisor shall prepare procedures as necessary to verify the setpoint of safety and relief valves subject to ASME Section XI, subsection IWV, as set forth in the approved Section XI submittal.

6.3.6 The Operations Superintendent, I&C Superintendent and the Mechanical Engineering Supervisor shall provide the Planning and Scheduling Supervisor information necessary to schedule Inservice Tests at their required frequencies.

6.3.7 The Planning and Scheduling Supervisor shall schedule IST Program procedures as part of the Administrative Procedure 1000.09, "Surveillance Test Porgram Control".

6.4 IST Program Additions, Deletions and Revisions

6.4.1 Proposals for revisions to the ASME Section XI can be made by any responsible individual recognizing the necessity for revision. The originator shall propose the revision by completing the appropriate section of Form 1015.06A and forwarding the form to the IST Coordinator(s). (Refer to Attachment 3 for guidance in ASME Section XI classification).

6.4.2 The IST Coordinator shall review the proposed change and then forward the form to the Plant Analysis Superintendent with his comments.

6.4.3 The Plant Analysis Superintendent shall review the proposed change. He shall then request any necessary actions (such as procedure changes, program revisions, etc.) as a result of the change. Upon completion of his review the Plant Analysis Superintendent shall return the completed form to the IST Coordinator(s) for filing and any further action (such as program revision, license amendments, submittal for PSC approval, etc.) as appropriate.

6.4.4 Approved changes that are in full compliance with applicable portions of ASME Section XI can be incorporated into ANO test procedures under their responsibility by the IST Coordinator, Operations Superintendents, I&C Supervisor, Mechanical Engineering Supervisor and/or the Plant Analysis Superintendent.

6.4.5 If any requirement of the ASME B&PV Code, Section XI, can not be met due to plant design, safety, or impracticality of testing, a request for Code relief will be prepared by the Operations Superintendent, IST Coordinator, I&C Superintendent and/or the Mechanical Engineering Supervisor as appropriate. This request and associated justification shall be processed to the Plant Analysis Superintendent for review.

6.4.6 If any requirement of the ASME B&PV code, section XI, cannot be met due to plant design, safety, or impracticality of testing, a request for code relief will be prepared by the Operations Superintendents, IST Coordinator, I&C Supervisor, or the Mechanical Engineering Supervisor. This request and associated jurisdiction shall be processed to the Plant Analysis Superintendnet for review and approval prior to implementation.

6.4.7 All changes to IST program procedures shall be reviewed and approved in accordance with Administrative Procedure 1000.06, "Procedure Review, Approval and Revision Control".

6.5 IST Program Updates

6.5.1 Every 20 months from the date of commercial operation, a revision to the Inservice Test Program shall be prepared for submittal to the NRC.

6.5.2 The IST Coordinator, using input from the Operations Superintendents, I&C Superintendent and Mechanical Engineering Supervisor shall prepare the revision submittal approximately six (6) months prior to the end of the 20 month interval.

6.5.3 This submittal shall include a listing of all additions, deletions and revisions to the existing programs that have been implemented since the last submittal to the NRC. This submittal shall also include justification for any relief requests from specific requirements of the Section XI of the ASME Code.

6.5.4 The submittal shall be issued by the Plant Analysis Superintendent after approval by the PSC.

6.6 IST Program Records

The IST Coordinator(s) shall maintain a working file of Inservice Test Program records which include:

6.6.1 Pump and valve test data. This file should include selected data from the current and the previous year.

6.6.2 All request for additions, deletions and revisions to the IST Program.

6.6.3 Baseline test data for each re-run of baseline data as required following maintenance or modifications.

6.6.4 All NRC submittals and correspondence involving the IST program.

7.0 FORMS AND ATTACHMENTS

Form 1015.06A, IST Program - Revision Control

Attachment 1 - Unit One IST Program

Attachment 2 - Unit Two IST Program

Attachment 3 - Guidance for ASME Section XI category classification of pumps and valves.

ATTACHMENT 2

LISTING OF UNIT 2 PUMPS REQUIRING INSERVICE TESTING

PUMP NO.	DESCRIPTION	SURVEILLANCE PROCEDURE
2P4A	Service Water Pump	2104.29, Supp. 1A
2P4B	Service Water Pump	2104.29, Supp. 1B
2P4C	Service Water Pump	2104.29, Supp. 1C
2P7A	Emergency Feedwater Pump	2106.06 Supp. 1
2P7B	Emergency Feedwater Pump	2106.06 Supp. 2
2P35A	Containment Spray Pump	2104.05, Supp. 1
2P35B	Containment Spray Pump	2104.05, Supp. 2
2P36A	Charging Pump	2104.02, Supp. 1
2P36B	Charging Pump	2104.02, Supp. 2
2P36C	Charging Pump	2104.02, Supp. 3
2P39A	Boric Acid Makeup Pump	2104.03, Supp. 1
2P39B	Boric Acid Makeup Pump	2104.03, Supp. 2
2P60A	Low Pressure Safety Injection Pump	2104.40, Supp. 1
2P60B	Low Pressure Safety Injection Pump	2104.40, Supp. 2
2P89A	High Pressure Safety Injection Pump	2104.39, Supp. 1
2P89B	High Pressure Safety Injection Pump	2104.39, Supp. 2
2P89C	High Pressure Safetiy Injection Pump	2104.39, Supp. 3
2P136A	NaOH Addition Pump	2104.05, Supp. 4
2P136B	NaOH Addition Pump	2104.05, Supp. 5

SUPPLEMENT I

MONTHLY 2P7A TEST

INITIALS

1.0 PREREQUISITE CONDITIONS

1.1 Emergency feedwater system is aligned per Attachment "A" of this procedure. ______

1.2 Reactor power is ≥ 3% FP if this is to suffice as an annual test. ______

1.3 Steam generator pressures are > 865 psig. ______

1.4 Blowdown demineralizer effluent or condensate storage supply available for 2P7A suction. ______

1.5 Portable vibrometer available and currently calibrated; pyrometer available if this test is to suffice as an annual test. ______

1.6 Blowdown demineralizer in operation and capable of receiving emergency feedwater flush, or is bypassed. ______

2.0 TEST METHOD

2.1 Verify 2P7A pump inboard, outboard, and turbine bearings have oil levels between the minimum and maximum marks on the gage glasses. (Record Oil Reservoir Levels in Section 3.2) ______

2.2 Open flush valve 2EFW-11B, 2P-7A flush and test isolation valve. ______

CAUTION
2P7B CONDENSATE SUCTION 2CV-0789-1 MUST BE OPEN DURING 2P7A FLUSHING OPERATIONS TO PREVENT SUCTION PIPING OVER-PRESSURIZATION FROM 2P-7B DISCHARGE-CHECK LEAKAGE.

2.3 If SU/BD demineralizer supply MOV 2CV-0706 is closed due to power being >5%, and SU/BD demineralizer is in service, open 2CV-0706; otherwise, "N/A" this step. ______

2.4 Record the idle pump suction pressure in Section 3.0 as indicated on 2PIS-0795-2(local). ______

2.5 Place 2P7A in flush operation by opening its steam supply MOV 2CV-0340-2; verify pump accelerates to rated speed (~3580 RPM) and does not trip on overspeed. ______

2.6 Open 2CV-0798-1 flush valve to estsblish test recirc. flows; monitor B/D demineralizer flows (if in service) ______

INITIALS

2.7 Verify water supply 2SV-0317-2 opens by noting return drain flow at local floor drain. ________

2.8 Verify 2P7A room cooler 2VUC-6A starts and its SW supply valve 2CV-1529-2 opens. ________

2.9 Montior 2P7A discharge pressures and flows and adjust, if necessary, to meet Section 3.0 criteria. ________

> NOTE
>
> It may be necessary to trim flush valve 2CV-0798-1 position slightly with the handjack to adjust flow and speed to within normal values as indicated in Section 3.0.

2.10 After at least five minutes of stable operation, gather the necessary data and record measured values in the appropriate spaces provided in Section 3.0. ________

2.11 Is this an annual test? Yes /__/ No /__/

> NOTE
>
> If this test is to suffice as an annual test, continue flushing operation until pump bearing temperature is stable and record temperature on data sheet (otherwise, mark temperature slots "N/A"). Bearing temperature is considered stable when three consecutive readings at ten minute intervals vary no more than 3%.

2.12 After recording data, close flush valve 2CV-0798-1 and close and lock-closed 2EFW-11B. ________

2.13 Is EFW stop-check testing required? Yes /__/ No /__/

> NOTE
>
> Emergency FW stop checks 2EFW-7A and 2EFW-8A require quarterly stroke testing. If this test is the first test (not a retest after maintenance) which falls in the month of Jan., Apr., July or Oct., continue with Steps 2.14 through 2.17; otherwise, mark steps "N/A" and proceed to Step 2.18.

SUPPLEMENT I

INITIALS

2.14 Position the following valves for SG feed alignment: ________

2.14.1 Open SG emergency FW isolation valves 2CV-1037-2 and 2CV-1039-2. ________

2.14.2 Open SG emergency feed valve 2CV-1026-2; verify >560 gpm flow to "A" S/G. ________

CAUTION

LIMIT THE TIME FEEDING THE SG'S AT HIGH FLOW RATES WHILE AT LOW REACTOR POWER TO ONLY THAT TIME NECESSARY TO ASSURE STABLE FLOW AND GATHER DATA; TERMINATE FEEDING IF SG HI-HI LEVEL IS REACHED WITH BLOWDOWN AT MAXIMUM, OR IF RCS TAVE OR PRESSURE DECREASES SIGNIFICANTLY.

2.14.3 Immediately reclose 2CV-1026-2; verify EFW flow falls to minimum. ________

2.15 Close 2CV-1026-2. ________

2.16 Open 'B' SG feed valve 2CV-1076-2; verify >560 gpm flow to 'B' SG. ________

2.17 Immediately close feed valve 2CV-1076-2, then close isolation valves 2CV-1037-2 and 2CV-1039-2. ________

2.18 Close 2P7A turbine steam supply 2CV-0340-2; verify pump stops. ________

2.19 Verify speed controller output at 100% and verify overspeed trip alarm clear. ________

2.20 Stop 2VUC-6A and close its SW MOV 2CV-1529-2. ________

2.21 Close 2CV-0706 if opened in Step 2.3 and if reactor power is >5% and the main F.W. pump(s) is supplying feedwater; otherwise, "N/A" this step. ________

POOR ORIGINAL

3.0 TEST CRITERIA & DATA COMPARISON

3.1 Record the values observed during flushing operations and compare with limiting range of values for operability.

TEST QUANTITY	INSTRUMENT	MEASURED VALUES	NORMAL RANGE	LIMITING RANGE FOR OPERABILITY	IS DATA WITHIN LIMITING RANGE? (Circle Yes or No)
Idle Suct. Press.	2PIS-0795-2	psig	> 7 psig	> 7 psig	Yes No
Running Suct. Press.	2PIS-0795-2	psig	> 7 psig	> 7 psig	Yes No
Discharge Press.	2PIS-0713-2	psig	> 1200 psi	> 1200 psi	N/A
Emerg. FW Flow	2FIS-0713-2	gpm	> 560 gpm	> 560 gpm	Yes No
I.B. Brg. Vibration		MILS	< 2 MILS	< 4 MILS	Yes No
O.B. Brg. Vibration		MILS	< 2 MILS	< 4 MILS	Yes No
I.B. Brg. Temp.		°F	N/A	< 180 °F	Yes No
O.B. Brg. Temp.		°F	N/A	< 180 °F	Yes No
Pump Speed	2SIS-0334-2	RPM	3575	3600	Yes No

3.2 Record Oil Reservoir Levels

2P7A I.B. BRG. OIL LEVEL	
2P7A O.B. BRG. OIL LEVEL	
2K3 I.B. BRG. OIL LEVEL	

3.3 If "NO" is circled in above table, declare 2P7A inoperable and notify Operations Superintendent.

Performed By ______________________ Operator Date/Time ______________

Analyzed By ______________________ S/S Date/Time ______________

4.0 OPERATIONS SUPERVISOR REVIEW AND ANALYSIS

4.1 Do all measured values recorded in Section 3.0 fall within the specified normal range: YES NO (Circle one).

4.2 Has this equipment been proven operable per Section 3.0? YES NO (Circle one).

4.3 Describe action taken in the event either 4.1 or 4.2 is answered "NO":

____

____

____

____

____

____

____

____

____

____

____

____

____

Date,Time ______________________

Signed ______________________ Ops. Supv.

48 Describe the bases for the change from less than or equal to
(3/4 7-8) 0.10 micro curies per gram dose equivalent I-131 to less than or equal to 0.046 micro curies per gram.

Answer The present technical specification is based upon the "Standard Technical Specification for Combustion Engineering PWR's" (NUREG-0212), which assures that offsite doses would be only a fraction of 10CFR 100 offsite dose limits. Both the Standard Technical Specifications and the ANO-2 Tech Specs. specify a secondary coolant activity of less than or equal to 0.10 uCi/gram dose equivalent I-131. However, Combustion Engineering assumed a more conservative secondary coolant activity of 0.046 uCi/gram I-131 when they performed the safety analyses for a main steam line break and for a steam generator tube rupture and calculated associated offsite doses.

61 (B3/4 7-2) The recent problems you refer to with the electric driven feedwater pump suggest that the pumping capability, although capable of meeting revised safety analysis considerations, may have been reduced somewhat. Outline your plans for evaluating this matter, and provide a schedule for reporting to the staff the results of your evaluation and corrective actions to be taken as required.

Answer The electric driven emergency feedwater pump, 2P7B, was removed from the plant soon after cooldown following shutdown for the current refueling outage and has been shipped to the manufacturer's testing and repair facilities. On April 6, 1981, a test was performed at Byron-Jackson (B-J) on their test stand. This test showed that degradation of the pump's performance has occurred since its manufacture. Consequently, AP&L has authorized B-J to refurbish the pump in an effort to regain lost performance.

This pump will then be returned to ANO and reinstalled. In-plant testing will be performed, and a new head-flow curve generated. This testing is expected to occur prior to June 1, 1981. AP&L will be glad to submit the results of this testing to NRC within approximately ten days of completion of plant tests.

PART II - INSTRUMENTATIONS AND CONTROLS SYSTEM

Q-1 The staff feels that Arkansas Power & Light Company should propose Technical Specifications to assure that the CPC is not considered or .able when environmental conditions including cyclic or ramped temperature fluctuation exceed those for which the CPC has been qualified. Provide justification for the environmental limits proposed.

Response

The CPC system has been qualified by testing over a wide range of temperatures for which the CPCS has been shown to perform its intended safety function. AP&L's experience has indicated that temperature transients within the qualified range may have contributed to certain individual CPC component failures. In all observed cases, the failures which have been attributed to temperature cycling have resulted in conservative action by the CPCS. AP&L feels that it is neither necessary nor practical to attempt to quantify the impact of room cooling upon the CPCS since the failures which have been attributed to room cooling problems have only reduced channel availability and have had no adverse impact on safety.

AP&L does not believe that Technical Specification changes which define temperature limits for CPCS operability are necessary. However, a proposed Technical Specification to verify CPC operability upon high CPC room temperature is included for your review. Since this proposed change has not received the necessary level

of internal AP&L review, the proposed change to Tech. Spec. 4.3.1.1.6 should be considered a draft which is subject to change. For your information, we have included some discussion of the CPC room cooling problems and a description of the changes AP&L has made or has planned to improve CPCS cooling. We will continue to evaluate the causes of component failures and make changes to improve CPCS availability.

The original ANO-2 design provided CPC room cooling via use of the Control Room emergency chillers. Early experience showed that these chillers were unsatisfactory for CPC room cooling because they were oversized for cooling only the CPC room and would trip on freeze protection. As stated in ANO-2 LER 50-368-80-026, an engineering analysis is in progress to correct this problem. A separate CPC room cooling unit has been added to provide more dependable air conditioning for the room, and the Control Room emergency chillers provide backup cooling. Annunciation has been added in the Control Room on high CPC room temperature to inform the operators of loss of normal CPC room cooling so that the backup cooling will be initiated.

During the first refueling outage, which is currently in progress, a major heat load will be removed from the CPC cabinets. The original design included channelized incore detector amplifiers in each CPC cabinet. These detectors perform no protection function and are planned to be removed from the CPC cabinets. This change should provide considerable reduction of CPC cabinet heat generation.

If after these modifications the ANO-2 CPC system performance still does not meet desired levels, additional modifications such as a dedicated backup room cooling unit or separate cabinet cooling units will be considered.

Q-2 Table 3-4 of CEN 147(s)-P contains upper and lower proposed allowed bounds on addressable constants. These bounds as currently proposed would restrict the values of addressable constants entered into the CPC to avoid only very gross errors. Other small yet unacceptable values could be entered. For example, a negative value of a diagonal element of the shape annealing correction matrix does not seem justified, and such values should be rejected by the computer. Furthermore, there may be values of addressable constants within the current proposed bounds which if entered could lead to violation of DNBR or LPD limits even when the CPC is otherwise functioning properly.

Therefore, please adopt more restrictive bounds on the addressable constants to assure that values may not be entered which are physically unrealistic or which could lead to violation of DNBR or LPD limits even when the CPC is otherwise functioning properly.

A potentially acceptable way to satisfy the second element of this request is to adopt power range dependent DNBR and LPD pre-trip alarm setpoints along with Technical Specifications to require action upon initiation of either alarm.

Response

Referring to our response to CPC software question 492.23, it is our position that the Type II addressable constants are thoroughly checked and quality assured prior to entry and thus are not susceptible to entry error. The use of addressable constant disks should preclude subsequent re-entry errors. Thus our position is that the entry range bounds do not need to be made more restrictive for these constants.

The Type I addressable constants do require frequent changes and may not always receive the same level of review as the Type II changes. The following additional administrative control is proposed:

> Addressable constant values* outside of the below prescribed allowable values shall not be implemented without the prior review and concurrence of the Plant Safety Committee:

Addressable Constant	Allowable Value*
F_{c1}	≤ 1.15
F_{c2}	0.0
C_{INOP}	0, 1, 2 or 3
T_R	≥ 1.02
K_{CAL}	≥ 0.85
C_{TP}	≥ 0.90

*These values are plant specific.

It is AP&L's position that software changes to the allowable bounds for addressable constant limits more restrictive than those already implemented are not necessary due to the strict controls we have already introduced.

The proposed use of power range dependent DNBR and LPD pre-trip alarm setpoints does not appear to be a workable solution since these would only be useful for catching overly conservative entry errors and would not appear to be useful for those that would be non-conservative.

Q-3 Propose Technical Specifications to assure that (a) plant procedures shall be in effect to control modifications to CPC addressable constants, (b) these procedures are consistent with Approved Physics and Thermal Hydraulic Methods; the approved methods should be referenced in the bases to the Technical Specifications, (c) CPC addressable constants and their allowed ranges (i.e., upper and lower bounds) are identified in the Technical Specifications, (d) values of addressable constants outside the allowed range are not to be entered without approval of the Plant Safety Committee, (e) an independent verification shall be conducted to confirm that addressable constant modifications have been made as approved by the Plant Safety Committee or the Engineering staff (whichever is applicable), (f) modifications to the CPC addressable constants based on information obtained through the plant computer data links shall not be made without approval of the Plant Safety Committee.

Response

(a) See proposed ANO-2 Tech. Spec. 6.8.1.g attached.

(b) ANO plant procedures contain the methodology for addressable constant calculations. Addressable constants are calculated consistent with the requirements of documents referenced in the bases for ANO-2 Technical Specification 2.2.1.

(c) and (d) See proposed ANO-2 Tech. Spec. 2.2.2 and Table 2.2-2 attached.

(e) See proposed change to ANO-2 Technical Specification Table 4.3-1 and its table notation.

(f) See proposed note addition to proposed ANO-2 Technical Specification 6.8.1.g.

Please note that these proposed Technical Specifications have not yet received the required level of internal AP&L review to be considered final and thus should be considered as drafts which are subject to change.

SURVEILLANCE REQUIREMENTS (Continued)

2. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 8 volts DC.

b. For the optical isolators: Verify that the input to output insulation resistance is greater than 10 megohms when tested using a megohmmeter on the 500 volt DC range.

4.3.1.1.5 The Core Protection Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.1.6 The Core Protection Calculator System shall be su' ected to a CHANNEL FUNCTIONAL TEST to verify OPER..BILITY within 12 hours of receipt of a valid High CPC Room Temperature alarm.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants are defined in Table 2.2-2. Type I Addressable Constants are expected to change frequently during plant operation. Type II Addressable Constant values are determined (or confirmed) during PHYSICS TESTS following each fuel loading and are not expected to change during plant operation. Changes to Type I Addressable Constants outside the Allowable Value range require Plant Safety Committee review prior to implementation. Changes to Type II Addressable Constants made other than as a result of post fuel loading PHYSICS TESTS shall require Plant Safety Committee review prior to implementation unless the changes are required for Technical Specification Compliance.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1

ACTION: With a Core Protection Calculator Addressable Constant found to be non-conservative, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant ~~(Type I addressable constant~~)	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant ~~(Type I addressable constant~~)	0.0
62	CEANOP	CEAC/RSPT inoperable flag ~~(Type I addressable constant~~)	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance ~~(Type I addressable constant~~)	≥ 1.02
64	TPC	Thermal power calibration constant ~~(Type I addressable constant~~)	≥ 0.90
65	KCAL	Neutron flux power calibration constant ~~(Type I addressable constant~~)	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint ~~(Type I addressable constant~~	unrestricted
67	LPDPT	Local power density pretrip setpoint ~~(Type I addressable constant~~)	unrestricted

TABLE 2.2-2 (CONTINUED)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

II. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERRØ	Thermal power uncertainty bias ~~(Type II addressable constant)~~
69	BERR1	Power uncertainty factor used in DNBR calculation ~~(Type II addressable constant)~~
70	BERR2	Power uncertainty bias used in DNBR calculation ~~(Type II addressable constant)~~
71	BERR3	Power uncertainty factor used in local power density calculation ~~(Type II addressable constant)~~
72	BERR4	Power uncertainty bias used in local power density calculation ~~(Type II addressable constant)~~
73	EOL	End of life flag ~~(Type II addressable constant)~~
74	ARM1	Multiplier for planar radial peaking factor ~~(Type II addressable constant)~~
75	ARM2	Multiplier for planar radial peaking factor ~~(Type II addressable constant)~~
76	ARM3	Multiplier for planar radial peaking factor ~~(Type II addressable constant)~~
77	ARM4	Multiplier for planar radial peaking factor ~~(Type II addressable constant)~~
78	ARM5	Multiplier for planar radial peaking factor ~~(Type II addressable constant)~~
79	ARM6	Multiplier for planar radial peaking factor ~~(Type II addressable constant)~~
80	ARM7	Multiplier for planar radial peaking factor ~~(Type II addressable constant)~~
81	SC11	Shape annealing correction factor ~~(Type II addressable constant)~~
82	SC12	Shape annealing correction factor

TABLE 2.2-2 (CONTINUED)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

II. TYPE II ADDRESSABLE CONSTANTS (continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
83	SC13	Shape annealing correction factor ~~(Type II addressable constant)~~
84	SC21	Shape annealing correction factor ~~(Type II addressable constant)~~
85	SC22	Shape annealing correction factor ~~(Type II addressable constant)~~
86	SC23	Shape annealing correction factor ~~(Type II addressable constant)~~
87	SC31	Shape annealing correction factor ~~(Type II addressable constant)~~
88	SC32	Shape annealing correction factor ~~(Type II addressable constant)~~
89	SC33	Shape annealing correction factor ~~(Type II addressable constant)~~
90	PFMLTD	DNBR penalty factor correction multiplier ~~(Type II addressable constant)~~
91	PFMLTL	LPD penalty factor correction multiplier ~~(Type II addressable constant)~~
92	ASM2	Multiplier for CEA shadowing factor ~~(Type II addressable constant)~~
93	ASM3	Multiplier for CEA shadowing factor ~~(Type II addressable constant)~~

TABLE 2.2-2 (CONTINUED)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

II. TYPE II ADDRESSABLE CONSTANTS (continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
94	ASM4	Multiplier for CEA shadowing factor ~~(Type II addressable constant)~~
95	ASM5	Multiplier for CEA shadowing factor ~~(Type II addressable constant)~~
96	ASM6	Multiplier for CEA shadowing factor ~~(Type II addressable constant)~~
97	ASM7	Multiplier for CEA shadowing factor ~~(Type II addressable constant)~~
98	CORR1	Temperature shadowing correction factor multiplier ~~(Type II addressable constant)~~
99	BPPCC1	Boundary point power correlation coefficient ~~(Type II addressable constant)~~
100	BPPCC2	Boundary point power correlation coefficient ~~(Type II addressable constant)~~
101	BPPCC3	Boundary point power correlation coefficient ~~(Type II addressable constant)~~
102	BPPCC4	Boundary point power correlation coefficient

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is > 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The correct values of addressable constants (See Table 2.2-2) shall be verified to be installed in each OPERABLE CPC.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2 and *
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *
14. Core Protection Calculators	S, W(9)	D(2,4),R(4,5)	M,R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Director, Nuclear Operations and to the SRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Director, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants

Note: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data links shall not be made without prior approval of the Plant Safety Committee.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

PART III - OTHER ISSUES

Q-1 Your letter of August 29, 1980, requested an extension from 12 to 24 hours for setting the pressurizer code safety valves during Mode 3. The following information is needed to enable our review.

a) Was the subject testing performed as part of the ANO-2 ASME Code Section XI Inservice Testing and Inspection program?
b) How many tests have been conducted on these valves to date, and what was the time required to do each of these tests?
c) How does Arkansas Power & Light Company's experience with the testing of these valves compare to general industry practice?
d) During the testing, are both valves rendered inoperable at the same time?

Response

a) Yes, the ANO-2 ASME Section XI Inservice Testing program was the governing requirement for the subject valves.
b) The two pressurizer code safety valves are numbered 2PSV-4633 and 2PSV-4634. 2PSV-4634 has been tested only once during pre-operational testing. The time to perform this test is not documented and had no time limit since it was performed prior to licensing. 2PSV-4633 has been tested six times. The first was performed prior to licensing. The second and third tests were performed prior to criticality ("radiation-free") and required less than 12 hours to complete. The fourth test was performed in March 1980 to verify setpoint

only and required no adjustment. The fifth test followed disc replacement in April 1980 after a small leak past the disc had developed. This test encountered no problems and required approximately ten hours to complete. The sixth and most recent test (July 1980) followed minor repairs on the valve but required approximately 16 hours to complete due to test apparatus problems and was the event which caused AP&L to request a Tech. Spec. change.

It should be pointed out that there is a tremendous incentive for AP&L to complete testing in as short a time as possible since the reactor must be kept hot and subcritical to complete this testing. Pressurizer code safety valve testing, when required, is generally a critical path activity which must be completed in order to begin electricity generation.

c) Since ASME Section XI and the standardized Technical Specifications do not specify a required time period for completion of such testing, few other units have a time limit. Consequently, documentation of time required to complete pressurizer code safety valve testing is not readily available. There does not appear to be a need for a time limit at all in Technical Specification 3.4.3, thus our request to extend the allowable time to 24 hours should cause no concern.

d) Testing is performed on only one valve at a time, thus only one valve is rendered inoperable as a result of the ASME Section XI tests.

ATTACHMENT B

<u>AFFIDAVIT PURSUANT</u>

<u>TO 10 CFR 2.790</u>

Combustion Engineering, Inc.)
State of Connecticut)
County of Hartford) SS.:

I, A. E. Scherer depose and say that I am the Director, Nuclear Licensing, of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations and in conjunction with the application of Arkansas Power and Light Company, for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

Response to Second Round Questions on Statistical Combination of Uncertainties, CEN-139(A) -P and CEN-124(B) -P, Part 2.

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure are the C-E thermal margin analysis methodology and the thermal hydraulic characteristics of C-E Cores, which is owned and has been held in confidence by Combustion Engineering.

2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a substantial competitive advantage to Combustion Engineering.

3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F.M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject documents herein are proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because.

a. A similar product is manufactured and sold by major pressurized water reactors competitors of Combustion Engineering.

b. Development of this information by C-E required thousands of manhours of effort and tens of thousands of dollars. To the best of my knowledge and belief a competitor would have to undergo similar expense in generating equivalent information.

c. In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of methods for the statistical combination of the uncertainties in thermal margin analysis.

d. The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e. The information consists of the C-E thermal margin analysis methodology and the thermal hydraulic characteristics of C-E cores, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included.

The ability of Combustion Engineering's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

g. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

A. E. Scherer
Director
Nuclear Licensing

Sworn to before me

this 28th day of April, 1981

Carey J. Wenzel
Notary Public

CAREY J. WENZEL, NOTARY PUBLIC
State of Connecticut No. 59962
Commission Expires March 31, 1985