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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Response to First Request for Additional Information from the NRC Reactor Systems Branch Regarding the ANO-2 Power Uprate License Application

Gentlemen:

By application dated December 19, 2000, Entergy Operations, Inc. submitted an "Application for License Amendment to Increase Authorized Power Level." On May 21, 2001, NRC personnel from the Reactor Safety Branch requested responses to 22 questions regarding the application. The attachment to this letter contains the responses to the staff's questions. The response was delayed due to personnel changes involving the lead NRC reviewer which caused delays in discussing the questions via teleconference. This submittal contains no regulatory commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 17, 2001.

Very truly yours,

Glenn R. Ashley

Glenn R. Ashley
Manager, Licensing

GRA/dwb
Attachment/enclosure

A001

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**Request for Additional Information
from Reactor Systems Branch Personnel Regarding the
Arkansas Nuclear One, Unit 2 (ANO-2) Power Uprate License Application**

General

NRC Question 1

Several instances in the Power Uprate Licensing Report (PULR) refer to a "107.5% power uprate." Please change this phrase to either a "7.5 percent power uprate" or an "uprated power of 3026 megawatts thermal."

ANO Response

The 7.5% uprate was incorrectly referred to as a 107.5% uprate in four instances. Other references referred to a power level of 107.5% or were otherwise appropriate. Clarification is provided for the following statements:

Attachment to Letter dated December 19, 2000 (2CAN120001)

Page 14 of 16, paragraph 3 - "This surface area permits a 107.5% uprate" should have stated "This surface area permits a 7.5% uprate."

2CAN120001, Enclosure 5, "Power Uprate Licensing Report"

Page 2-1, paragraph 2 - "The BOP SSCs have been evaluated for the impact of the 107.5% power uprate..." should have stated "The BOP SSCs have been evaluated for the impact of the 7.5% power uprate..."

Page 2-2, paragraph 1 - "This surface area permits a 107.5% power uprate..." should have stated "This surface area permits a 7.5% power uprate..."

Page 6-5, section 6.4.6 - "...as a result of the replacement steam generator and 107.5% power uprate..." should have stated "...as a result of the replacement steam generator and 7.5% power uprate..."

PULR Section 4.1.1 - Reactor Coolant System

NRC Question 2

The PULR states on page 4-2 that the original design T_{hot} was 612 °F and that the Cycle 16 T_{hot} would be 609 °F. Streaming effects from low leakage cores can cause a stratified temperature profile in the hot legs with the peak temperature being higher than the average temperature. Have you observed hot leg streaming effects in ANO-2? What is

the impact of the peak hot leg temperature exceeding the design temperature due to present or future hot leg streaming effects?

ANO Response

ANO-2 has observed a range of hot leg temperatures. Current average hot leg temperature is 604.9° F with an occasional maximum temperature of 612° F in individual resistance temperature detectors (RTDs).

No impact is expected from peak hot leg temperatures exceeding the original design temperature. For the structural integrity analysis, thermal loads were evaluated assuming a 10% increase in core ΔT, which gives a normal operating hot leg temperature of 618.5° F. Section 5 of the PULR provides additional information. Refer to Section 5.4 and Table 5-5. Although the table lists a T_{hot} value of 617.7° F, the paragraph below the table explains that a conservative increase of 10% was used as a guideline, not the 8.62% listed in the table.

Discussion is provided in Section 6 of the PULR regarding the design transients used for the structural design of the reactor coolant system (RCS). This analysis assumed a normal operating hot leg temperature of 617.7° F. This increase in temperature has no affect on the component material properties. Neither the tensile strength nor the allowable S_m experiences any significant change between 600° F and 650° F. A total of a 6° F increase in T_{hot} is anticipated following the power uprate effort; a 4° F increase in T_{hot} from uprate plus a 2° F increase in T_{hot} due to a 2° F increase in T_{cold}. The maximum hot leg temperature will remain consistent with the analyses values and the average hot leg temperature will be less than the original design consideration of 612° F.

SRP Section 5.2.2 - Overpressure Protection

NRC Question 3

Page 4-3 of the PULR states that the effect of power uprate and the replacement steam generators on the low temperature overpressure protection (LTOP) analysis was discussed in correspondence dated December 21, 1999 (2CAN129907). The referenced correspondence does not discuss the effect of the power uprate. Please confirm that an LTOP analysis was performed that accounts for the power uprate and provide the steam generator tube plugging limits for which the analysis is valid.

ANO Response

The technical specification bases change request dated December 21, 1999 (2CAN129907), changed the LTOP event based on an analysis which included the effects of the replacement steam generator and power uprate. Although not specifically

discussed in the submittal, this analysis assumed a decay heat load commensurate with a 7.5% power uprate.

Up to 10% steam generator tube plugging was considered for the LTOP analyses (mass addition event and energy addition event), but 0% was found to be conservative for the limiting event, which became the energy addition event. With the larger surface area available when no tubes are plugged, the energy transfer from the steam generator is maximized. There is also a pressure correction factor calculation which considers the pressure drop from the pressurizer to the limiting vessel location. This pressure drop is larger and more conservative with the higher flow rates associated with no plugging.

The LTOP analysis may be affected by changes to the pressure-temperature curves necessitated by the results of the surveillance capsule analysis. This is mentioned in Section 8.4 of the PULR. A new LTOP analysis may be performed based on the revised pressure-temperature curves. Any such effort will be addressed with the revised pressure-temperature curves.

NRC Question 4

Page 5-28 of the PULR discusses the report on overpressure protection. Please provide the assumptions and results (including the steam generator tube plugging limits for which these analyses are valid) for the bounding pressure excursion transients that were used to evaluate the adequacy of the sizing for the primary and secondary safety valves at the uprated power level.

ANO Response

The bounding pressure excursion transient used to evaluate overpressure protection is the loss of condenser vacuum (LOCV) event. When this event was reanalyzed for the replacement steam generators, the uprated power level was assumed. Therefore, the effect of power uprate on the LOCV analysis (and hence the overpressure protection report) was discussed in the submittal dated November 29, 1999 (2CAN119901), in Enclosure 4, Section 1.4.1 (pages 28-33 of 172). The input assumptions and results are included in this discussion. Steam generator tube plugging limits from 0-10% were considered; zero tubes plugged is limiting.

SRP Section 5.4.7 - Residual Heat Removal System

NRC Question 5

Section 4.1.4, page 4-7 of the PULR states that an evaluation of the shutdown cooling system was performed that is comparable to that described in Safety Analysis Report (SAR) Section 9.3.6.6. Please provide your evaluation that demonstrates the adequacy of the shutdown cooling system at the uprated power level.

ANO Response

Operation at a higher power level increases the decay heat that must be removed from the RCS in a cooldown from normal operating conditions to cold shutdown. The adequacy of the shutdown cooling (SDC) system was verified by an evaluation performed by ABB Combustion Engineering (now CE Nuclear Power, LLC or CENP) using the DESCENT computer analysis code. DESCENT models time-dependant SDC system performance during a cooldown. This analysis was done to support the replacement steam generators as well as the 7.5% power uprate.

The DESCENT analysis verified that cold shutdown (less than 200° F) can be reached within the 36-hour time requirement of the technical specifications assuming the most limiting single failure. The failure assumed is the loss of one emergency diesel generator, which results in using only one pump and one heat exchanger for the cooldown.

Input assumptions were similar to the analysis described in Amendment 15 of SAR Section 9.3.6.3. The PULR has a typographical error in the SAR reference to Section 9.3.6.6; the correct reference should be 9.3.6.3. Four and one-half hours are allowed to reach 300° F/300 psig in the RCS, which is the point at which SDC can be initiated. The RCS volume used is 9770 ft^3. Service water temperature is conservatively assumed to remain constant at 121° F, the peak temperature for the emergency cooling pond. The effective area per shutdown cooling heat exchanger is 5220 ft^3, which contains an increased allowance for tube plugging. Primary flow through the tubes is assumed to be 3000 gpm at the beginning of the cooldown with a step change to 4500 gpm when RCS temperature reaches 220° F. This is consistent with operator actions to maximize primary flow through the heat exchangers to maximize the cooldown.

Given these conditions, DESCENT predicted that cold shutdown would be reached in about 32 hours from the time of the reactor trip. This is consistent with the analysis currently described in the SAR and well within the time limit in the technical specifications.

PULR Section 4.1.2 – Chemical and Volume Control System (CVCS)

NRC Question 6

Page 4-5 of the PULR states that the design requirement for the CVCS system to provide for letdown or makeup for a 75 °F/hr heatup or cooldown was met except for a momentary deficit. This indicates that there must be a revised design requirement that

was met without exception. Please state the true design requirements and verify that they are met without exception.

<u>ANO Response</u>

The design criterion for the CVCS stated in the SAR (Section 9.3.4.1.2) is "to provide the required makeup using two of three charging pumps when the reactor coolant is cooled at the rate of 75° F/hr." Section 9.3.4.1.1 lists the function requirement to "maintain the required volume of water in the RCS by compensating for coolant contraction or expansion resulting from changes in reactor coolant temperature and for other coolant losses or additions." The CVCS meets these requirements without exception. The system is not required to maintain a constant pressurizer level during such transients. Power uprate has no affect on this requirement.

Because of the change in RCS volume due to the replacement steam generators, there was a slight increase in makeup requirements from Cycle 14 to Cycle 15, the first cycle with the replacement steam generators. During the first 12 minutes of a cooldown from 545° F at the rate of 75° F/hr, an average of 86 gpm of charging flow would be needed to maintain pressurizer level. This slightly exceeds the capacity of two charging pumps, which is 44 gpm per pump with four gpm for controlled bleedoff. This causes only a momentary drop in pressurizer level because of the limited duration. Since there is no change in RCS volume from Cycle 15 to Cycle 16, this requirement is not affected by power uprate.

SRP Chapter 15 – Accident Analysis

<u>NRC Question 7</u>

Page 7-105 of the PULR states that the power measurement uncertainty was reduced. Justify the reduction in power uncertainty from 3 percent to 2 percent.

<u>ANO Response</u>

With the change in power rating due to power uprate, the analyses were changed to use the standard power measurement uncertainty of two percent defined in 10CFR50.46, "Acceptance Criteria for emergency core cooling systems for light water nuclear power reactors." A two-percent power measurement uncertainty is required in an Appendix K, "ECCS Evaluation Models" LOCA analysis. The actual instrument uncertainty associated with the power measurement for ANO-2 is less than two percent.

<u>NRC Question 8</u>

Table 8.3-1 of the PULR states that the peak rod axial average burnup is 67,300 megawatt days per metric ton uranium (MWD/MTU). This value is greater than the NRC-approved burnup limit for your fuel and is outside that range of approval and validity of your fuel rod evaluation codes. Please provide a list of all safety analyses that

are affected by this assumption and provide information to show that the analyses are conservative when they are done within the valid limits of fuel burnups.

ANO Response

The fuel mechanical design calculations: stress, strain, fatigue, clad collapse, shoulder gap, and hold down margin all yield worse results with increased burnup. The results based on 67,300 MWD/MTU are more conservative than if the burnup limit of 60,000 MWD/MTU was used. The 67,300 MWD/MTU limit was obtained from a generic analysis which calculated the maximum burnup achievable before reaching the criteria limits. The limit for clad strain was the first limit reached at 67,300 MWD/MTU.

The following is a summary of the burnup used for various analyses:

Topic	Report Section	Burnup (MWD/MTU)
Cladding Collapse	8.3.1.1	67,300
Clad Fatigue	8.3.1.2	67,300
Clad Stress	8.3.1.3	67,300
Clad Strain	8.3.1.4	67,300
Rod Maximum Internal Pressure	8.3.1.5	65,000 (1)
Waterside Corrosion	8.3.1.6	60,000

Note 1: The rod maximum internal pressure analysis was performed to rod average burnups of 65,000 MWD/MTU. The analysis was performed to rod average burnup of 60,000 MWD/MTU per the burnup topical, with extra time steps added to achieve a rod average burnup of 65,000 MWD/MTU in anticipation that higher burnups may be allowed in the future. For the present, however, the licensed burnup for ANO-2 fuel remains at 60,000 MWD/MTU.

NRC Question 9

Your report does not list the fuel bundle designs that will be present in the core after the power uprate. Provide a list and description of the fuel bundle designs that will be used in your core. If more than one fuel bundle design is used, how are mixed core effects evaluated?

ANO Response

With the exception of the center assembly, Power Uprate cores will be utilizing the same standard Combustion Engineering (CE) 16x16 fuel assembly design bundles which are currently used in ANO-2 cores. The center assembly is from a prior ANO-2 core utilizing a previously used assembly design. The current ANO-2 cores use this older design in the

center assembly. Power uprate cycles will be using Erbia as a burnable absorber, the current ANO-2 cores use Gadolinia.

NRC Question 10

Page 7-105 of the PULR states that the charging pump flow was changed from 44 gallons-per-minute (gpm) to 46 gpm for the Chapter 15 safety analyses. What is the impact of raising the CVCS flow rate from 44 gpm to 46 gpm? What safety analyses are affected by this change? Why is there no technical specification change if this new value is required to meet the safety analysis?

ANO Response

Actual charging pump flow has not changed nor has a requirement been added for a minimum charging pump flow. Charging flow is typically not credited in safety analyses where it would provide a benefit; it is assumed in analyses where it makes the situation worse. No credit is taken for charging flow in the non loss-of-coolant accident (LOCA) analyses discussed in section 7.3 or the LOCA analyses in section 7.1 of the PULR. Two events in which charging flow is modeled are an uncontrolled boron dilution incident (see the PULR, Section 7.3.4) and the steam generator tube rupture (see the PULR, Section 7.3.13). Increasing the assumed charging flow makes these analyses more conservative. A higher charging flow is also conservatively considered in determining offsite releases associated with event generated iodine spikes.

NRC Question 11

Verify that your analyses use approved methodologies and that your analyses and inputs meet all restrictions in the approved methodologies. For example, a fuel burnup of 67,300 MWD/MTU would not meet the restrictions of your approved fuel rod modeling methodology.

ANO Response

The approved methods and verified input data used in the transient analyses is discussed in the "Analysis Overview" subsection in Section 7.3 of the PULR for the non-LOCA events and in the "Methodology" subsection of Section 7.1 for LOCA.

NRC Question 12

The tube plugging limits in your transient, accident, and loss-of-coolant accident (LOCA) analyses are set at 10 percent (i.e., pages 7-14 and 7-18 of the PULR). What is the maximum percentage of plugged tubes allowed in any single steam generator? Was the effect of this allowed asymmetry (if any) evaluated for all transient, accident, and LOCA analyses?

ANO Response

The LOCA and non-LOCA analyses considered 0 to 10% plugging in each steam generator for relevant events. Asymmetric plugging up to 10% in one generator and 0% in the other was also considered for relevant events as indicated in the discussion of the respective analyses. For the following events, explicit tube plugging considerations were made:

Event	Plugging (%)
Large Break LOCA	10
Feedwater Line Break	0
Coastdown data – Seized Rotor and Loss of Flow	0 / 10*
Loss of External Load / or Turbine Trip	0
Instantaneous Closure of a Single Main Steam Isolation Valve	0 / 10*
Main Steam Line Break	0
Subcritical, Hot Zero Power, and Hot Full Power CEA Withdrawal	10
Boron Dilution	10
Loss of Feedwater	0
Excess Heat Removal	0
Steam Generator Tube Rupture	10

* These events considered asymmetric plugging limits up to 0% in one steam generator and 10% in the other steam generator.

NRC Question 13

Please provide the initial steam generator mass and the basis for that value for all Chapter 15 transient and accident analyses.

ANO Response

The initial steam generator mass is calculated by CENTS based on the event specific defined RCS initial conditions (temperature, pressure and flow) and the initial steam

generator level. All Chapter 15 events are based on 70% indicated level at hot full power conditions and 60% level at hot zero power conditions. The one exception is the feedwater line break analysis which is based on an inventory of 164,400 lbm. For this event, a more conservative inventory based on the high level alarm limit of 78% indicated level was assumed.

PULR Section 7.3.11.2 – Feedwater Line Break Accident

NRC Question 14

Provide the location of the feedwater line inlet in your steam generator.

ANO Response

The centerline of the inlet nozzle is 361" above the top of the tubesheet. SAR Figure 5.5-7 (Amendment 16) shows the relative position of the elevated feed ring. The J nozzle outlet is 386" above the top of the tubesheet.

NRC Question 15

Justify that the low level trip occurs with at least 40,000 pounds mass (lbm) of liquid remaining in the faulted steam generator (page 7-135 of the PULR). The justification should be based on the accuracy of the instrumentation under the conditions and the physics of two phase flow. What is the impact of not being able to take credit for this trip?

ANO Response

The instrument uncertainty calculations have taken into consideration the steam generator conditions when determining the mass of inventory in the steam generator at time of trip. The blowdown effects of density changes and velocities following a feedwater line break (FWLB) have been accounted for. An inventory of 40,000 lbm credited in the FWLB is conservative with respect to the approximate 78,000 lbm at the low level trip setpoint credited in the loss of feedwater analysis.

Credit for low level indication during a FWLB on the affected steam generator is similar to the credit taken by Westinghouse plants as presented in WCAP 9230, "Report on the Consequences of a Postulated Main Feedline Rupture" (January 1978) and WCAP 9236, "NOTRUMP: A Nodal Transient Steam Generator and General Network Code" (September 1977). The replacement steam generators for ANO-2 are Westinghouse designed steam generators.

The 40,000 lbm was determined consistently and conservatively with the methods documented in WCAP 9230 and WCAP 9236 using the NOTRUMP code. This 40,000

lbm assumption was then used in the CENP CENTS code for determination of the effects on the reactor coolant system versus the Westinghouse LOFTRAN code.

Not crediting the low level setpoint in the affected steam generator will result in a limited range of feedwater line breaks potentially overfilling the pressurizer.

10 CFR 50.62 – Anticipated Transients Without Scram (ATWS)

NRC Question 16

Please submit an analysis of an ATWS at the uprated power level to show that peak pressures and the percentage of cycle with an unfavorable moderator temperature coefficient are consistent with those considered by the staff in deliberations leading to promulgation of the ATWS rule.

ANO Response

The ATWS Rule, 10CFR50.62, required that the ANO-2 design be modified to include a diverse scram system (DSS), diverse turbine trip (DTT) and diverse emergency feedwater actuation system (DEFAS). Paragraph (c)(2) of the rule required the installation of a DSS system for CE and Babcock and Wilcox manufactured plants. These system designs were approved by the NRC in safety evaluations dated June 21, 1989 (2CNA068902) and May 1, 1990 (2CNA059001) based on their reliability, independence and diversity from the plant protection system. Power uprate does not modify the DSS, DTT or DEFAS designs, and therefore, these systems continue to comply with the ATWS Rule. Consistent with the respective safety evaluations approving these designs, the actuation setpoint for DSS/DTT remains above the reactor protection system high pressurizer pressure setpoint and below the pressurizer safety valve opening set pressure. The actuation setpoint for DEFAS is below the plant protection system setpoint for the emergency feedwater actuation system. The ATWS Rule imposed system design requirements, but ATWS events did not become design basis events requiring re-analysis.

SRP Section 15.6.5 - LOCA

NRC Question 17

Please provide your analysis of the switch over from refueling water storage tank injection to sump recirculation to show that the core remains at an adequately cool temperature during any flow reduction or interruption that may occur during switch over. The analysis assumptions should be consistent with your emergency operating procedures.

ANO Response

The supply of water used for emergency core cooling system (ECCS) injection initially comes from the refueling water tank (RWT) and automatically transfers to the reactor building sump once the RWT water is exhausted. This automatic switchover is based upon level in the RWT and the timing is such that no air is entrained from the RWT, which could damage the ECCS equipment or impact the ability to adequately cool the core. The switchover from RWT suction to sump is accomplished with a continuous supply of water for suction by opening the sump suction valves as the RWT supply valves close. For power uprate no increase to ECCS system flows are required and as such no changes were required to the analysis documenting adequacy of the valve timing for switchover to recirculation. The recirculation mode of the containment spray system is discussed in SAR section 6.2.3.2.2.2.

Question 18

Page 7-6 of the PULR states that the long-term cooling model is different than the one referenced in the ANO-2 SAR. Please provide your long term cooling analysis as required by the Safety Evaluation approving topical report CENPD-254-P-A when it is first applied to a plant application referencing the report.

ANO Response

This response includes information discussed in a conference call with the NRC staff on September 27, 2001.

Section 7.1.5 of the PULR describes the post-LOCA long term cooling (LTC) analysis that was performed at power uprate conditions. The analysis consists of a boric acid precipitation analysis for a large cold leg break LOCA. The analysis uses the Westinghouse boric acid precipitation evaluation model for Combustion Engineering designed pressurized water reactors, from CENPD-254-P-A, "Post-LOCA Long Term Cooling Evaluation Model," June 1980. The CENPD-254 methodology uses the BORON computer code for calculating the boric acid concentration in the core following a large break LOCA.

The power uprate LTC analysis replaces the LTC analysis documented in Section 6.3.3.15 of the ANO-2 SAR. That analysis also consists of a boric acid precipitation analysis for a large cold leg break. The methodology used in the analysis is briefly described in Section 6.3.3.15. A more detailed description of the methodology is contained in a letter dated April 5, 1978 from D.H. Williams (AP&L) to J.F. Stolz (NRC), "Arkansas Power & Light Company, Arkansas Nuclear One-Unit 2, Docket No. 50-368, ECCS Long Term Cooling." The letter is enclosed. The methodology was the original methodology developed by CE in 1975 for addressing boric acid precipitation following a large break LOCA.

The basic assumptions and differential equations for calculating boric acid concentration in the core that were used in the 1975 methodology formed the basis for the CENPD-254 methodology contained in the BORON computer code.[1] The following are several of the common assumptions and features of the two methodologies.

1. No credit is taken for subcooling of the safety injection flow.
2. No credit is taken for entrained liquid leaving the core (only steam leaves the core).
3. No credit is taken for boric acid volatility.
4. Prior to the initiation of simultaneous hot and cold side injection, the only injection into the reactor vessel credited is that which is required to replace boil-off.
5. Maximum boric acid concentrations are represented for all sources of boric acid.
6. Maximum initial liquid volumes are represented for all sources of boric acid.
7. No credit is taken for increased boric acid solubility due to boiling point elevation.
8. Credit is taken for mixing of liquid in the core and lower plenum. This credit is based on the results of a post-LOCA boric acid concentration test.

Although the two methodologies are very similar, they are not identical. The following three differences between the two methodologies, as applied to ANO-2, are worth noting.

1. The 1975 methodology assumed a boric acid solubility limit of 32 wt%, which is based on an RCS pressure of 20 psia. The ANO-2 power uprate boric acid precipitation analysis assumed a more conservative boric acid solubility limit of 27.6 wt%, which is based on an RCS pressure of 14.7 psia.
2. The two methodologies used different "mixing volumes".[2] In the 1975 methodology, the mixing volume is comprised of the liquid in the lower plenum, core, and outlet plenum below the elevation of the bottom of the hot leg. The lower plenum is assumed to be filled with single phase liquid while the core and outlet plenum contain two-phase fluid. In the CENPD-254 methodology, the mixing volume is different.
3. As described in the response to Question 3 in Appendix E to CENPD-254, the decay heat model used in the CENPD-254 methodology is based on the 1973 version of ANS Standard 5.1, including the recommended uncertainties of 1.2 up to 1000 seconds and 1.1 thereafter. As described in both Section 6.3.3.15 of the ANO-2 SAR

[1] CENPD-254 was submitted to the NRC in August 1977. The NRC issued the Safety Evaluation Report for CENPD-254 in July 1979, and the "-A" version of the topical was issued in June 1980.
[2] The "mixing volume" is the volume of liquid inside the reactor vessel within which the boric acid concentrates.

and in the April 5, 1978 letter, the 1975 methodology uses maximum decay heat values in compliance with 10CFR50, Appendix K. Based on a review of the existing documentation from that era, it is not known whether the 1975 methodology used a single multiplier of 1.2, as explicitly identified in Appendix K, or the two multipliers described above, which are cited in the American Nuclear Society standard referenced by Appendix K.

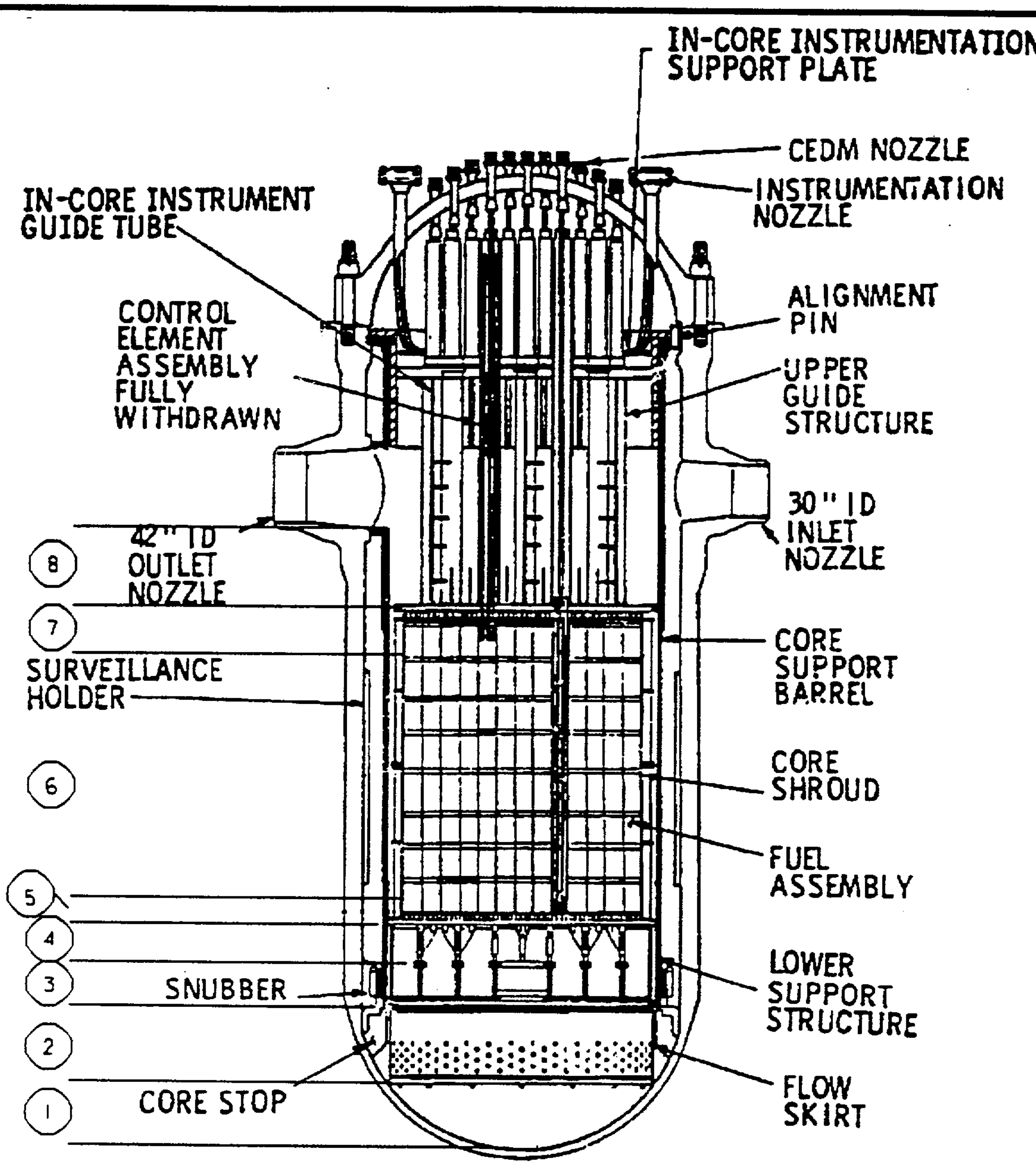
During the September 27, 2001, conference call, the staff requested the value of the mixing volume used in the ANO-2 power uprate boric acid precipitation analysis, as well as various ANO-2 reactor vessel water volumes. A value of 10,532 gallons (1408 ft^3) was calculated for the ANO-2 mixing volume. As a discretionary conservatism (i.e., a conservatism not required by the methodology), the input value used in the BORON code was reduced to 10,000 gallons (approximately a 5% reduction). Table 1 lists water volumes for the lower plenum, core, and outlet plenum regions of the ANO-2 reactor vessel. From the information presented in Table 1 on the following page, the total water volume that is inside the core support barrel (excluding the water between the core support barrel and the core shroud) and between the top of the core support structure and the bottom of the core barrel outlet nozzles (i.e., Regions 5 through 8), is 968.1 ft^3.

Table 1
ANO-2 Reactor Vessel Water Volumes

No.	Description	Water Volume, ft^3
1	Reactor vessel bottom head below flow skirt	138.0
2	Flow skirt region (inside of flow skirt, from bottom of flow skirt to bottom of lower support structure (LSS))	313.1
3	Lower support structure bottom region (inside of core support barrel from bottom of LSS to top of beam flange)	137.0
4	Lower support structure top region (inside of core support barrel from top of beam flange to bottom of core support plate)	84.0
5	Lower inactive core (inside of core shroud from bottom of core support plate to bottom of active core)	28.0
6	Active core (inside of core shroud from bottom of active core to top of active core)	579.6
7	Upper inactive core (inside of core shroud/core barrel from top of active core to top of fuel alignment plate)	100.8
8	Lower outlet plenum region (inside of core support barrel from top of fuel alignment plate to bottom of core support barrel outlet nozzles)	259.7
9	Core bypass region (region between core support barrel and core shroud)	168.0
10	Annular region between core support barrel and LSS cylinder	12.6

Note:

1. See ANO-2 SAR Figure 4.1-1 (next page) that has been annotated with circles drawn around numbers 1-8 to depict the top and bottom elevations of Regions 1 through 8.



SAR FIGURE NO. 4.1-1

AMENDMENT-15

ARKANSAS NUCLEAR ONE UNIT 2 RUSSELLVILLE, ARKANSAS	ENTERGY	SCALE : NONE DRAWN : CHARLEY RANKIN DESIGN : ENTERGY CAD NO : 4f1-01.sar
REACTOR VERTICAL ARRANGEMENT	BASED ON DRAWING NO	SHEET: 1 / REV:

<u>*NRC Question 19*</u>

Please provide the results of your pump trip analysis for a small-break LOCA as required by Item II.K.3.5 of NUREG-0737, "Clarification of TMI Action Plan Requirements," to determine the maximum time allowed to trip the reactor coolant pumps.

<u>Response</u>

For CE plants, the determination of the maximum time allowed to trip the reactor coolant pumps was resolved by CEN 268, "Justification of Trip Two/Leave Two Reactor Coolant Pump Trip Strategy During Transients," (March 1984) which cites Case P14 of CEN-114, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems," to show an infinite time to trip the pumps. CEN 268 is applicable to all current CE plants. A generic plant was analyzed in this analysis. The safety evaluation report (SER) for Item II.K.3.5 accepts the determination of the maximum time allowed to trip the reactor coolant pumps for all CE plants. The SER, dated May 29, 1986, states that "...the time available to the operator to trip the RCP for a small-break LOCA is unlimited." The maximum time allowed to trip the reactor coolant pumps meets the criteria of Generic Letter 83-10a, Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (February 8, 1983). Raising the power level at ANO-2 does not invalidate the applicable bases for the determination of an infinite time. No new analysis is required for power uprate.

10CFR50.63 - Station Blackout

<u>NRC Question 20</u>

Please submit the ANO-2 coping analysis for a station blackout at the uprated power level to show that the plant is able to cope with a station blackout for the ANO-2 specific duration.

<u>ANO Response</u>

As documented in the "Supplemental Safety Evaluation for the Arkansas Nuclear One Units 1 and 2 (ANO-1&2) Station Blackout Rule (10 CFR 50.63) (TAC Nos. 68508 and 68509)" dated October 24, 1991 (0CNA109111), no coping analysis was performed or required for ANO-2 because an alternate AC diesel generator was installed. Per 10CFR50.63(c)(2) and the guidance provided in NUMARC 87-00, "NUMARC Initiatives for Addressing Station Blackout at Nuclear Power Plants," the alternate AC diesel generator has been demonstrated, by testing, to be available to power the shutdown buses within 10 minutes of the onset of station blackout. Since the analysis of record for emergency AC power is bounding for power uprate (see Sections 2.2.4 and 2.2.5 of the PULR), no additional analysis is required.

SRP Section 4.2 - Fuel System Design, PULR Chapter 8 – Nuclear Fuel

<u>*NRC Question 21*</u>

Table 8.3-1 of the PULR lists the peak rod axial average burnup as 67,300 MWD/MTU both for current conditions and for uprated conditions. Please explain this, considering that the maximum approved 1 pin burnup is 60 megawatt days per kilogram uranium per references 8.3-8 and 8.3-9.

<u>ANO Response</u>

See response to Question 8.

<u>*NRC Question 22*</u>

Section 8.1.1.1 of the PULR gives a brief description of the rod bow penalties. Please expand on this description. In particular, please discuss why the value given for burnups up to 33 gigawatt days per metric ton uranium (GWD/MTU) is valid and provide a more detailed justification of no penalty for burnups beyond 33 GWD/MTU.

<u>ANO Response</u>

Avoidance of thermally induced fuel damage during normal steady state operation and during anticipated operational occurrences is the principal thermal-hydraulic design basis. Steady state DNBR [departure from nucleate boiling ratio] analyses of the bounding cycle design at the rated power level of 3026 MWt have been performed using the TORC computer code described in Reference 1, the CE-1 critical heat flux correlation described in References 2 and 10, the simplified TORC modeling methods described in Reference 3, and the CETOP code described in Reference 4 and approved in Reference 5.

Effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analysis in the manner discussed in References 5, 6, 7, and 8. The penalty used for this analysis, 0.6% of minimum DNBR, is valid for assembly burnups up to 33 GWD/MTU. The Modified Statistical Combination of Uncertainties (MSCU) methodology presented in Reference 9 was applied with the rod bow penalty, the calculational factors listed in Table 8.1-1 of the PULR, and other uncertainty factors at the 95/95 confidence/probability level to define a design limit of 1.25 on CE-1 minimum DNBR. For assemblies with burnup greater than 33 GWD/MTU, sufficient available margin exists to offset rod bow penalties due to the lower radial power peaks in these higher burnup batches. Consistent with the practice described in ANO-2 Technical Specifications basis 3/4.2.4, the margin associated with the lower radial peak offsets the increase in rod bow penalty in the higher burnup bundles. Hence, the rod bow penalty

based upon Reference 8 for 33 GWD/MTU is applicable for all assembly burnups expected for the power uprate.

References for the response to NRC question 22:

1. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.

2. CENPD-162-P-A, "Critical Heat Flux Correlation for CE-1 Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution," April 1975.

3. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," June 1981.

4. CEN-214(A)-P, Rev. 1-P, "CETOP Code Structure and Modeling Methods for Arkansas Nuclear One-Unit 2," July 1982.

5. Robert A. Clark (NRC) to William Cavanaugh III (AP&L), "Operation of ANO-2 During Cycle 2," July 21, 1981 (Safety Evaluation Report and Licensing Amendment No. 26).

6. CEN-139(A)-P, "Statistical Combination of Uncertainties: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Arkansas Nuclear One - Unit 2," November 1980.

7. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1979.

8. CEN-289(a)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One - Unit 2," December 1984.

9. CEN-356(V)-P-A, Rev. 01-P-A, "Modified Statistical Combination of Uncertainties," May 1988.

10. CENPD-207-P-A, "C-E Critical Heat Flux, Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-uniform Axial Power Distribution," December 1984

Enclosure

Photocopy of Letter Dated April 5, 1978 from Daniel H. Williams to Mr. J. F. Stolz

Subject: **Arkansas Power & Light Company**
Arkansas Nuclear One-Unit 2
Docket No. 50-368
ECCS Long Term Cooling
(File: 2-1510)

(Note: Due to the age of the letter, only a microfilmed version is available. The document is the best quality available.)

HELPING BUILD ARKANSAS

ARKANSAS POWER & LIGHT COMPANY

PO BOX 551 LITTLE ROCK, ARKANSAS 72203 • (501) 371-4000

April 5, 1978

2-048-4

RECEIVED
APR 10 1978
ARKANSAS POWER & LIGHT CO.
ARKANSAS NUCLEAR ONE

Director of Nuclear Reactor Regulation
ATTN: Mr. J. F. Stolz, Chief
Light Water Reactors Branch #1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Arkansas Power & Light Company
Arkansas Nuclear One-Unit 2
Docket No. 50-368
ECCS Long Term Cooling
(File: 2-1510)

Gentlemen:

In response to a verbal request from your Mr. Glenn Kelley, we have enclosed information describing our long term cooling post-LOCA boron precipitation calculations. As we understand it, this information is all that is needed to allow Mr. Kelly to complete his review.

Very truly yours,

Daniel H. Williams
Manager, Licensing

DHW:dr

Enclosure

7 8 1 0 0 0 8 1 1 4 7 0 4

1. Background

The following discussion focuses on the performance of the Emergency Core Cooling System (ECCS) during extended periods of time following a loss-of-coolant accident (LOCA). Long-term residual heat removal is accomplished by continuous boil-off of fluid in the reactor vessel. As borated water is delivered to the core region via safety injection and virtually pure water escapes as steam, unacceptably high concentrations of boric acid and other solution additives may accumulate in the reactor vessel.

For a hot leg break, safety injection flow introduced via the cold legs will travel down the annulus, through the core, and out the break. A flushing path is established through the reactor vessel, precluding the buildup of solids in the core region. However, for a cold leg break, only that amount of injected water required for decay heat removal is delivered to the core; the remainder spills out the break. Therefore, because of the geometry of the Reactor Coolant System, there is no flushing flow through the core for a cold leg break and boric acid concentration will increase.

2. Solution

A circulation flow through the reactor vessel should be established to flush solids from the core region and insure continued operability of the ECCS independent of break size or break location. A minimum core through-flow of 20 gpm is required within 4 hours post-LOCA. Ultimately, subcooled cooling of the core is achieved and a shutdown cooling mode can be initiated.

3. Mathematical Model and Assumptions

A mathematical model and computer code has been developed to predict the buildup of boric acid in the post-LOCA reactor vessel. In addition, this model is used to perform parametric studies to determine required core through-flows and flush initiation times.

The concentration of boric acid in the reactor vessel has been modeled as a function of time by establishing a boric acid mass balance for the reactor vessel. Results of this model are presented in Figure (1).

Separate mass balances have been developed for the injection and recirculation modes of ECCS operation. In each mass balance, it is conservatively assumed that the flow rate of influent to the reactor is only that required to replace boil-off, and that all boric acid in the influent remains in the vessel. The flow required to match boil-off is a function of decay heat and RCS pressure and temperature. (For long-term ECCS operation, a core flushing flow is superimposed on the system.)

Boric acid buildup calculations have been performed for both large and small cold leg breaks. The different system pressures in these limiting cases result in several competing factors which affect the time until precipitation. A high system pressure causes a higher rate of boil-off, resulting in a somewhat faster buildup of boric acid. However, this faster buildup rate at high pressure is more than offset by a substantial increase in boric acid solubility at the associated higher temperature. See Figure (2). The net result is that the large break is found to be the worst case; a RCS pressure of 20 psia is assumed.

The effects of various durations of the ECCS injection mode have been considered. The point at which recirculation is initiated has been found to have a negligible effect on the total time elapsed until precipitation occurs. In spite of this, conservative assumptions are made for those factors affected by the recirculation initiation time (in particular the boric acid inventory in the sump water) to insure that the worst case has been considered.

For this analysis, only boric acid is presented. Other solution additives, specifically trisodium phosphate (TSP) or sodium hydroxide (NaOH), may be present in the reactor fluid and will influence the precipitation of solids. The presence of TSP, in particular, will contribute an additional specie to the dissolved solids inventory but is expected to increase boric acid solubility. There is no direct quantitative data on this subject; however, the following qualitative observations can be stated:

The solubility of boric acid when neutralized with a given quantity of sodium hydroxide (or TSP) has been predicted from References 1-3. The data in these references pertains to solutions of sodium borate salts. Such solutions have been shown (References 4-6) to contain the same distribution of ionic and molecular species as boric acid solutions neutralized with sodium hydroxide to the appropriate Na/B ratio. The references show that all the sodium borate salts have solubilities greater than that of pure boric acid, indicating that the addition of sodium hydroxide (or trisodium phosphate) to the ECCS water will result in an increase in boric acid solubility. The species of dissolved solids which result from TSP hydrolysis are given in Reference 7. It is shown for the appropriate solution pH that TSP hydrolyzes (approximately 100%) to form monosodium phosphate, disodium phosphate, and sodium hydroxide. The solubility of these species (Reference 8) is more than an order of magnitude greater than their maximum possible concentrations. Thus, it appears that the presence of TSP will result in a net increase in boric acid solubility.

A special effort was extended to model the post-LOCA reactor vessel liquid inventory as a function of time. (This is the solvent for the boric acid solution.) The reactor vessel has been divided into three regions, as shown in Figure 3. Region "A" is defined as the free volume from the lower plenum to the bottom of the core (excluding the annulus); Region "B" is the free volume of the core; and region "C" is the region from the top of the core to the bottom of the hot leg. Region "A" is assumed to be entirely liquid. Regions "B" and "C" are two-phase regions with liquid masses calculated as a function of time based upon the two-phase model of Reference 9. A mean void fraction is determined for Region "B"; the surface void fraction at the interface of Regions "B" and "C" is conservatively assumed to be the void fraction throughout Region "C".

Precipitation of solids is assumed to occur in the reactor vessel when the concentration matches the solubility at the solution temperature. On Figure (1), developed for a conservative system pressure of 20 psia, the boric acid solubility is approximately 32 wt%. Due to the complexities involved in predicting the nature, location and extent of precipitation, no precipitation will be tolerated and a core flushing flow will be initiated prior to reaching this solubility limit.

The following assumptions were made:

a. A "large" cold leg break is considered as the worst case (low RCS pressure yields lowest boric acid solubility). A reactor coolant system pressure of 20 psia is assumed.

b. Only boric acid is considered. The effects of chemical additives such as NaOH or TSP are neglected, as well as the effects of dirt, paint, and general post-LOCA containment debris.

c. The mass of liquid in the reactor vessel is calculated as a function of time by adding the liquid masses of Regions "A", "B", and "C" of Figure 3. It is further assumed that there is complete mixing of the liquid inventory in Regions "A", "B", and "C" of Figure 3. This has been confirmed by laboratory testing.

d. Region "A" is assumed to be completely liquid. Regions "B" and "C" are two-phase regions. It is assumed that there is complete mixing of the liquid inventories in Regions "A", "B", and "C". No credit is taken for liquid above the bottom of the hot leg.

e. Uniform concentration in the containment sump is assumed. Entrapment of sump fluid in isolated cavities is not considered.

f. Maximum decay heat values for appropriate core type are used in compliance with 10CFR50, Appendix K.

g. No credit is taken for increased solubility due to boiling point elevation from high solids concentration.

h. Without flushing flow, the only injection into the reactor vessel is that required to replace boil-off. No credit is taken for subcooling of the injection flow.

i. No credit is taken for boric acid volatility.

j. Carryover is neglected.

k. Blowdown and refill occurs instantaneously at t = 0.

l. Recirculation mode of operation is initiated at one hour post-LOCA, at which time it is assumed that the entire boron inventory has been discharged into the reactor coolant/containment sump system.

m. Maximum boric acid concentrations and inventories are used for all sources. Assumed invento[illegible] re as follows:

Source	Boric Acid Concentration (wt%)	Liquid Mass (lb)
Reactor Coolant System	0.7	470,000
Refueling Water Tank	1.3	3,403,000
Safety Injection Tanks	1.3	376,000
Concentrated Boric Acid Storage Tanks	12.0	194,200

The mathematical model is defined by the following equations:

Injection Flow to Core

To match boil-off: $\dot{m}(t) = \dot{Q}(t)/\Delta h$

Flushing flow: $\dot{F}$

where $Q(t)$ = decay heat (Btu/hr)

Δh = heat of vaporization (Btu/lb)

Liquid Mass in Reactor Vessel $M(t)_{RV}$

$$M(t)_{RV} = M_A + M_B + M_C$$

$$M_A = (\rho_f)\,(V_A)$$

$$M_B = (\rho_f)\,(V_B)\,(1-\alpha_B)$$

$$M_C = (\rho_f)\,(V_C)\,(1-\alpha_C)$$

where α = void fraction

$$\alpha_B = 1 - \frac{K}{\dot{m}(t)} \operatorname{Ln}\left(\frac{\dot{m}(t) + K}{K}\right)$$

$$\alpha_C = \frac{\dot{m}(t)}{\dot{m}(t) + K}$$

ρ_f, ρ_g = Liquid, steam densities

K = $(V_D)\,(\rho_g)\,(A)$

V_D = drift velocity (conservatively assumed to be 2 fps)

A = core flow area

__Boric Acid Concentration in Reactor Vessel__ $C_{RV}(t)$

$$C_{RV}(t) = \frac{M_{BA}(t)}{M_{RV}(t)}$$

__Boric Acid Mass Balance__

__Injection Mode:__ $M_{BA}(t) = (M_{BA})_o + \int \dot{m}(t) C_{inj}\, dt$

where: $(M_{BA})_o$ = LBS BA in reactor vessel at $t = 0$

C_{inj} = BA concentration of injection fluid.

__Recirculation Mode:__ $M_{BA}(t) = (M_{BA})_{RAS} + \int \dot{m}(t) C_S(t)\, dt$

where: $(M_{BA})_{RAS}$ = LBS BA in reactor vessel at initiation of recirculation.

$C_S(t)$ = SUMP BA CONC = $\dfrac{(M_{BA})_{Total} - (M_{BA})_{RV}}{M_{SUMP}}$

__Recirculation with Hot Leg Injection:__

$$M_{BA}(t) = (M_{BA})_{FL} + \int \dot{m}(t) C_S(t)\, dt + \int \dot{F}[C_S(t) - C_{RV}(t)]\, dt$$

where: $(M_{BA})_{FL}$ = LBS BA in reactor vessel at initiation of core flush.

__Recirculation with Hot Leg Suction:__

$$M_{BA}(t) = (M_{BA})_{FL} + \int \dot{m}(t)\, C_{inj}(t)\, dt + \int \dot{F}[C_{inj}(t) - C_{RV}(t)]\, dt$$

where: $C_{inj}(t) = \left[\dfrac{Q_{LP} + Q_{HP}}{Q_{LP} + Q_{HP} + \dot{F}}\right] C_S(t) + \left[\dfrac{\dot{F}}{Q_{LP} + Q_{HP} + \dot{F}}\right] C_{RV}(t)$

Q_{LP} = low pressure safety injection flow.

Q_{HP} = high pressure safety injection flow.

4. References to Attachment A

(1) Byrnes, D. E., Foster, W. E., WCAP-1570, January 1961.

(2) Cohen, "Water Coolant Technology of Power Reactors", Gordon and Breach, New York 1969.

(3) U. S. Borax and Chemical Corporation Technical Bulletin.

(4) Muetterties, E. L., The Chemistry of Boron and Its Compounds, Wiley, New York 1967.

(5) Nachtrieb, N. H., Nomii, R., Inorganic Chemistry, 6, 1189, 1967.

(6) Smith, D., Jr., Wiersema, R. J., Inorganic Chemistry, Vol. II No. 5, 1972, 1152-1154.

(7) Klein, H. A., "Use of Coordinated Phosphate Treatment to Prevent Caustic Corrosion in High Pressure Boilers", Combustion, October 1962.

(8) Handbook of Chemistry and Physics, 50th Edition, 1969-1970.

(9) CENPD-137P C-E Topical Report (Proprietary)"Calculative Method for the C-E Small Break LOCA Evaluation Model" August 1974.

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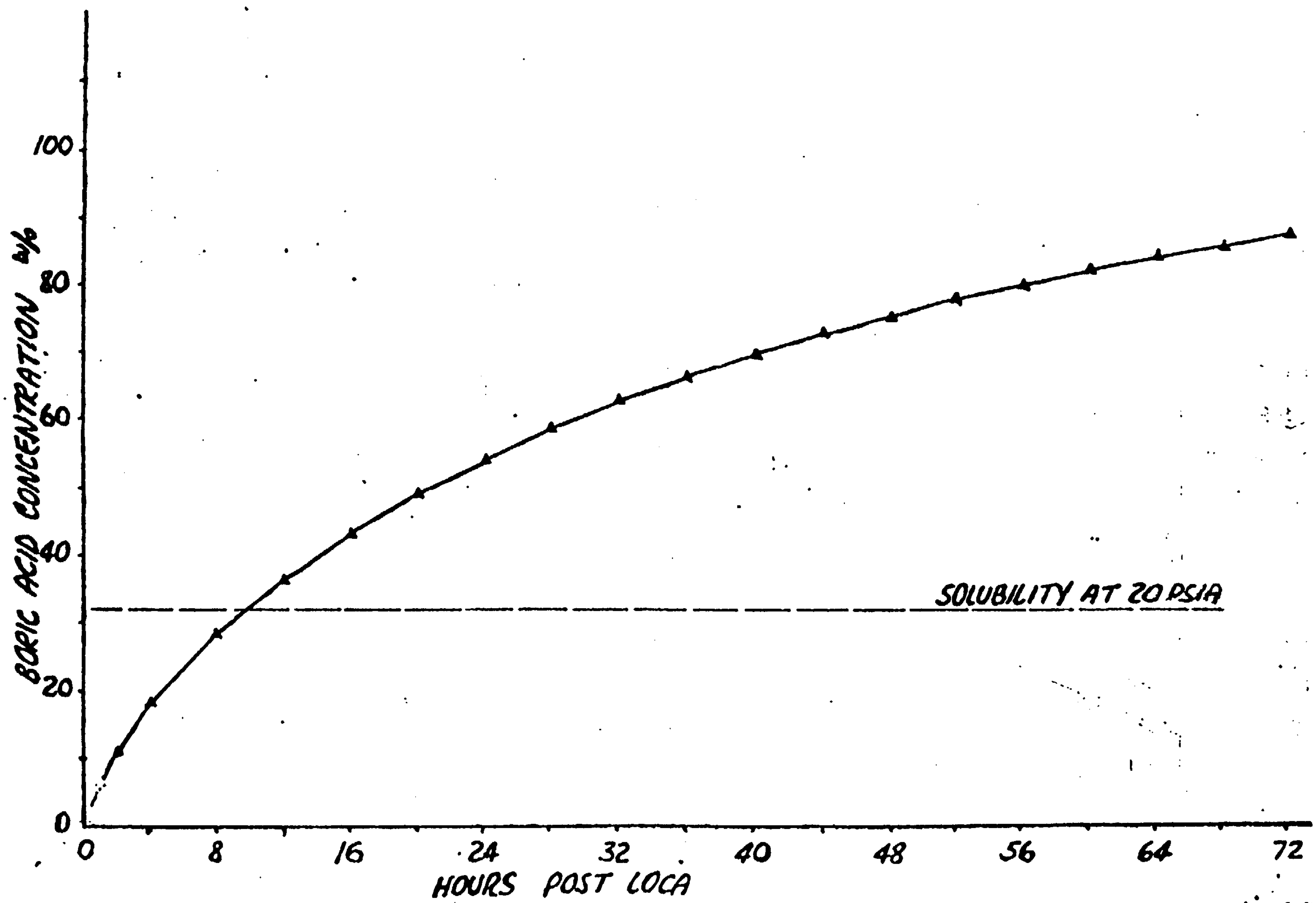
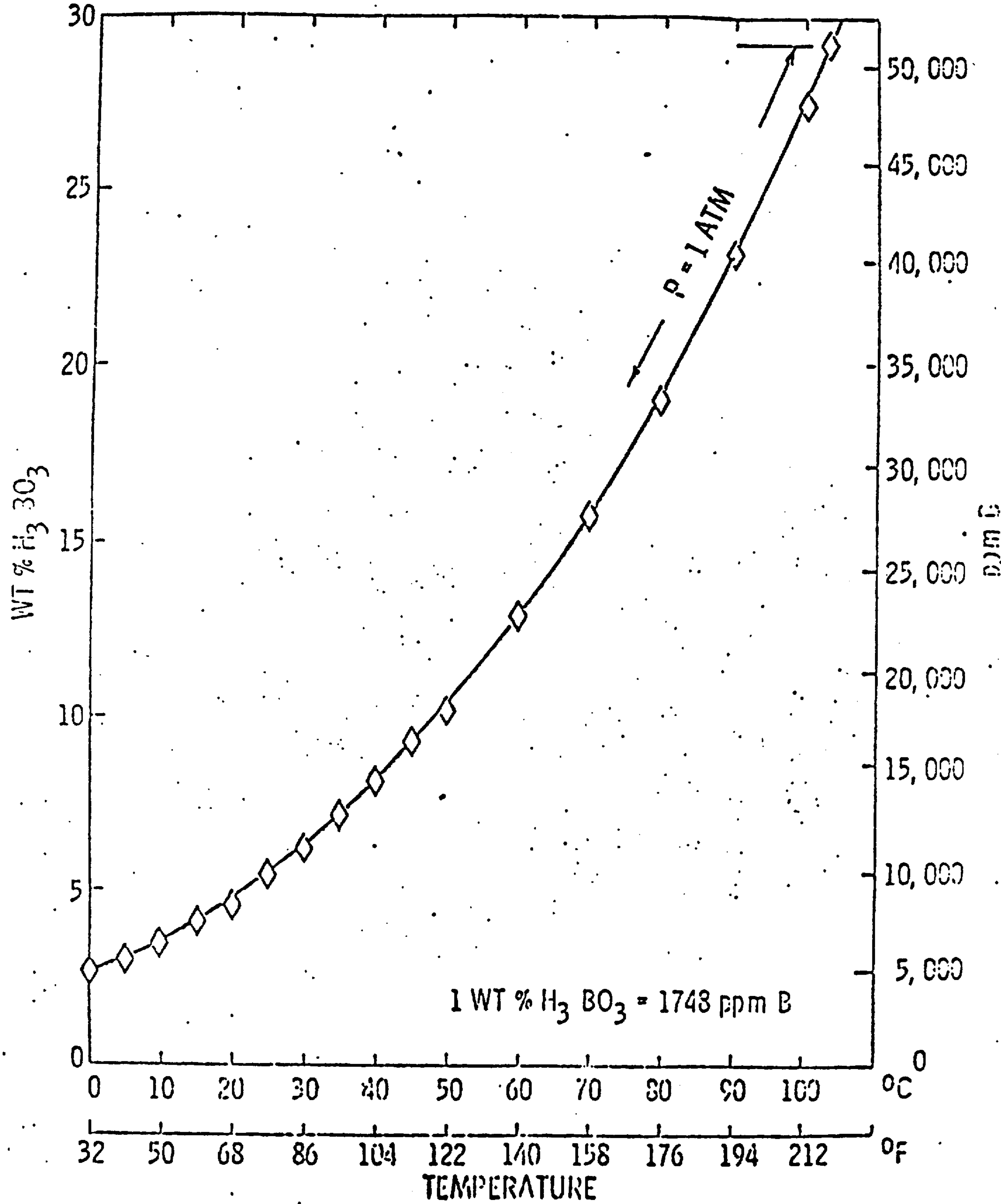


FIGURE (1)

IGURE (2A) OF

SOLUBILITY OF BORIC ACID IN WATER vs TEMPERATURE
COHEN P. 221 (Ref.)



COHEN P. 221 (Ref. 1)

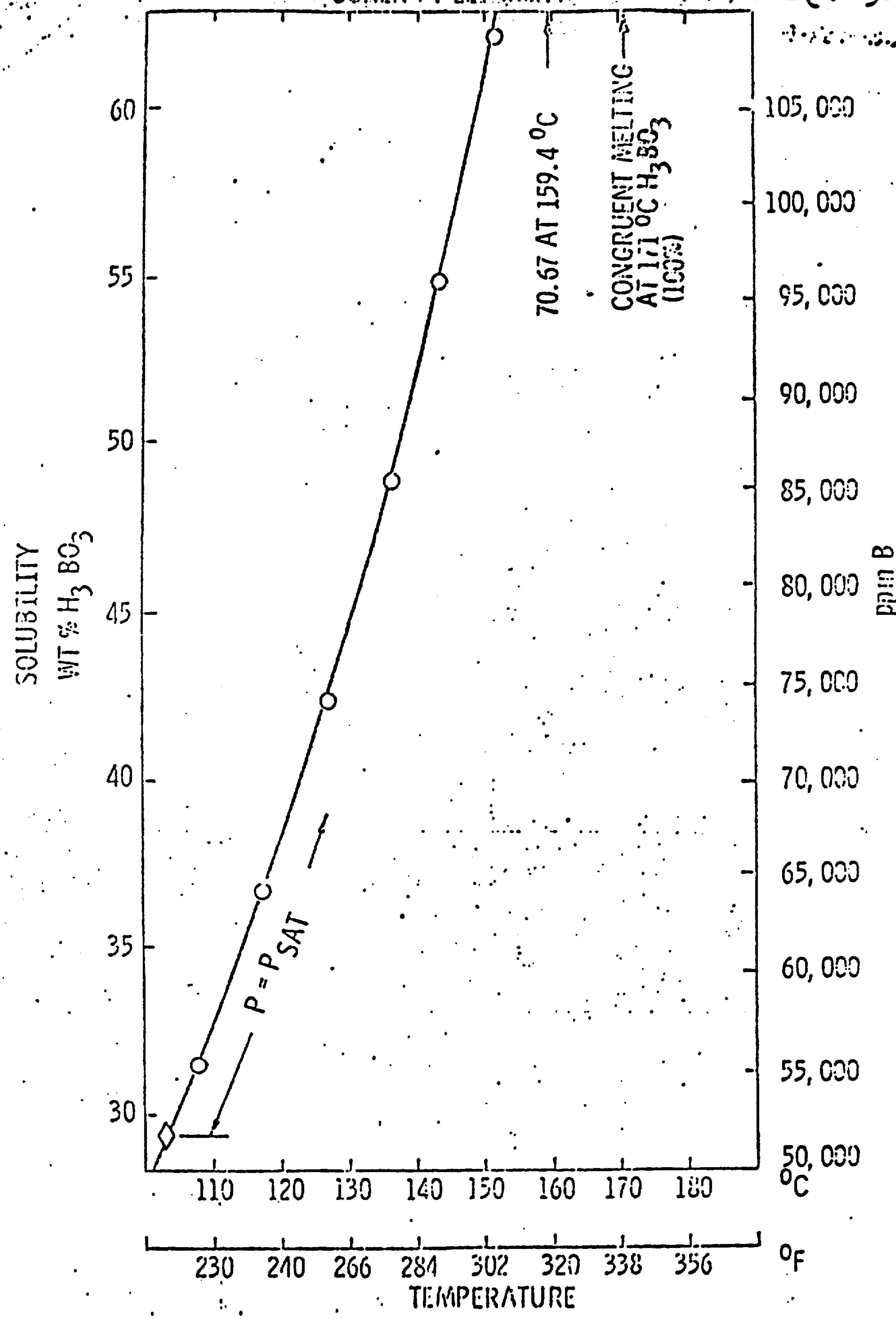
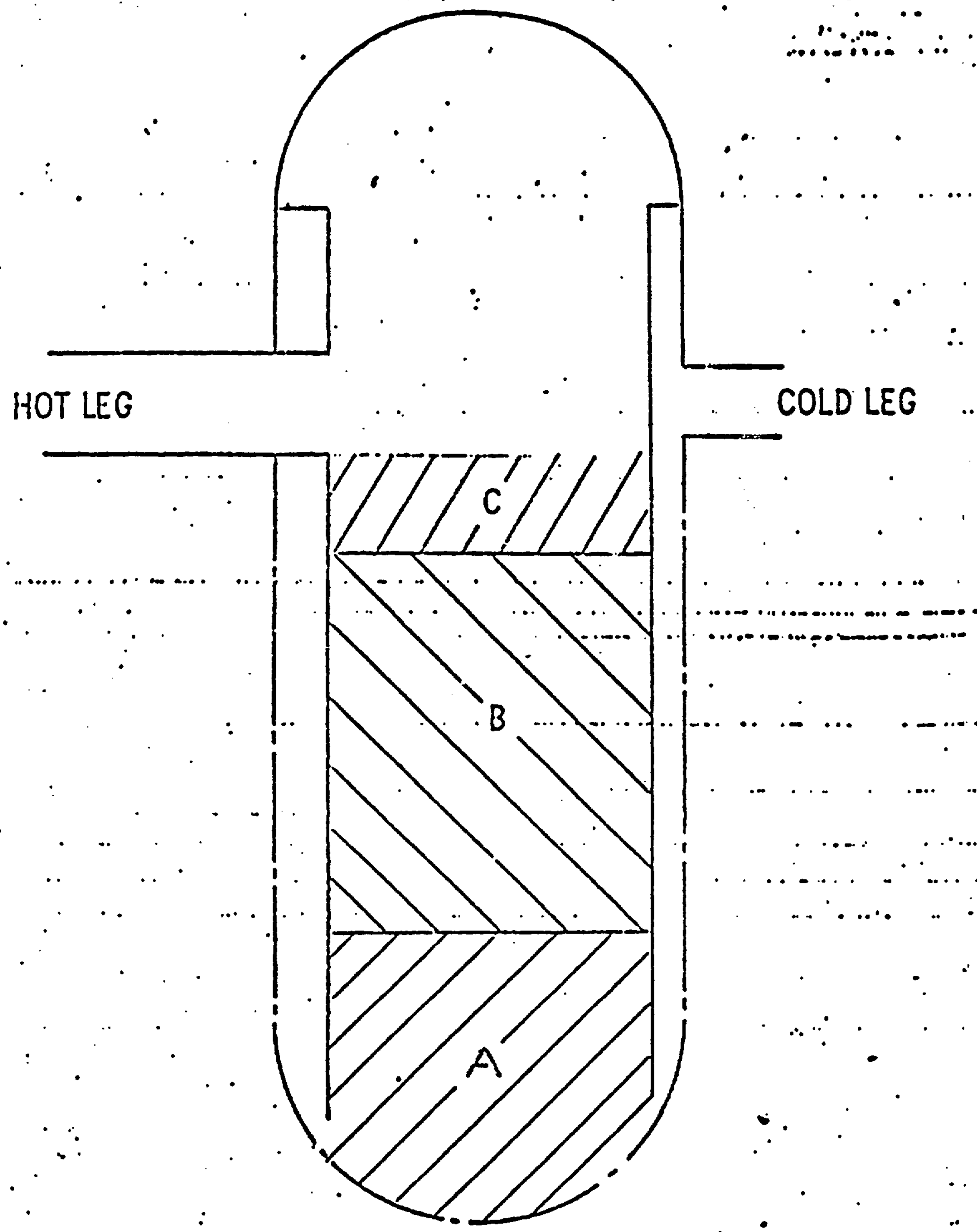


FIGURE (3) REACTOR VESSEL REGIONS



"A" = LOWER PLENUM TO BOTTOM OF CORE (EXCLUDING ANNULUS)
"B" = CORE REGION
"C" = TOP OF CORE TO BOTTOM OF HOT LEG