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2CAN100102

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Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Response to Second and Third Requests for Additional Information from the Reactor Systems Branch and Follow-up Information from Letter 2CAN100110 dated October 17, 2001, Regarding ANO-2 Power Uprate

Gentlemen:

In a letter dated December 19, 2000 (2CAN120001), Entergy Operations, Inc. submitted an "Application for License Amendment to Increase Authorized Power Level." On May 21, 2001, Nuclear Regulatory Commission (NRC) personnel from the Reactor Systems Branch requested responses to 22 questions. Responses were provided in a letter dated October 17, 2001 (2CAN100110). On September 4, 2001, Reactor Systems Branch personnel requested responses to 23 additional questions. The responses to these 23 questions are contained in Attachment 1.

The responses to NRC questions 11b and 11c contain information proprietary to Westinghouse Electric Company, LLC. Attachment 1 is a non-proprietary version of the response; therefore, the proprietary information has been removed. Brackets [ ] are used to indicate areas where proprietary information has been removed.

Attachment 2 contains the proprietary response, as well as an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses the considerations listed in paragraph (b)(4) of Section 2.790 of the *Code of Federal Regulations.* The affidavit applies to Attachment 2 although the text of the affidavit refers to 2CAN100102, Attachment 1. In the letter from Westinghouse to Entergy that transmitted the proprietary information and affidavit, Attachment 1 was the correct citation. During the development of this NRC submittal, which was subsequent to receipt of the affidavit, it was

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determined to be less confusing to include the proprietary information and affidavit in Attachment 2, not Attachment 1.

On September 13, 2001, Reactor Systems Branch personnel requested a response to one additional question. The response to that question is contained in Attachment 3.

Additionally, Attachment 4 provides supplemental information in regard to the response to NRC Question 18 from the first set of Reactor Systems Branch questions (letter 2CAN100110 dated October 17, 2001). A portion of the response cites three differences between the original (1975) methodology and the CENPD-254 methodology for addressing boric acid precipitation following a large break loss of coolant accident. Only two sentences of the five page response to the question were not included because the details of determining the mixing volumes for the CENPD-254 were classified by Westinghouse as proprietary information. The proprietary information was discussed with the NRC in a follow-up telephone call on October 23, 2001. During the telephone call, NRC personnel stated that the additional details were needed to adequately resolve the Staff's question. Attachment 4 is the non-proprietary version of the supplemental information.

Attachment 5 is the proprietary version of the supplemental information concerning NRC Question 18. An affidavit signed by Westinghouse, the owner of the information, is included with the attachment. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses the considerations listed in paragraph (b)(4) of Section 2.790 of the *Code of Federal Regulations*. The text of the affidavit refers to Enclosure 1 of letter LTR OA 01 24 dated October 8, 2001, the letter from Westinghouse to Entergy that transmitted the proprietary information to Entergy for inclusion in the letter to the NRC. Accordingly, it is respectfully requested that the proprietary information in Attachments 2 and 5 be withheld from public disclosure in accordance with 10CFR2.790.

Correspondence regarding the proprietary aspects of the information contained in Attachments 2 and 5 should be addressed to Mehran Golbabai, Project Manager, ANO-2 Power Uprate, Westinghouse Electric Company, CE Nuclear Power, LLC, 2000 Day Hill Road, Windsor, CT 06095.

This letter contains no regulatory commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 31, 2001.

Very truly yours,

Glenn R. Ashley

Glenn R. Ashley
Manager, Licensing

GRA/dwb
Attachments

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Attachment 1

Response to Second Request for Additional Information from the Reactor Systems Branch Regarding the ANO-2 Power Uprate License Application

Response to Second Request for Additional Information from the Reactor Systems Branch Regarding the ANO-2 Power Uprate License Application

NRC Question 1

The proposed changes to Technical Specification (TS) 3.5.4 will only specify the refueling water tank (RWT) volumes assumed in the accident analysis and move the required RWT indicated water level to plant procedures. This proposal will not provide sufficient information in the TSs for operators control and Nuclear Regulatory Commission (NRC) enforcement of this safety requirement. Please modify your proposed TSs to keep the indicated RWT water level of 91.7% to 100% in TS 3-5.4 as that in the current TSs.

ANO Response

Part of the intended change to refueling water tank (RWT) inventory technical specification 3.5.4 submitted in our license application dated December 19, 2000 (letter 2CAN120001), was to remove the indicated level. This change was requested to assist in minimizing the updates to this specification for adjustments in instrument uncertainties and other conversion factors. To allow for ease of operator control and NRC enforcement of this safety requirement, the following sentence will be added to refueling water tank technical specification bases section 3/4.5.4: "An RWT indicated level between 100% and 91.7%, in combination with the RAS {recirculation actuation signal} setpoint, ensures that the analysis assumptions with respect to available borated water volume is maintained." Since this additional information will be in the bases, it will assist in minimizing updates to the specification yet provide for ease of operator control and NRC enforcement.

NRC Question 2

Section 2.4.6.1 of the application evaluates the emergency feedwater system. Please describe the affect of power uprate on the condensate storage capacity required to meet the requirement of Branch Technical Position RSB 5-1, using safety grade equipment to achieve cold shutdown, and coping of a station blackout.

ANO Response

The ANO-2 Operating License predates the requirements of Branch Technical Position RSB 5-1. ANO-2 is considered a Class 3 plant based on an operating license issue date of July 18, 1978. Notwithstanding the above, the service water system provides the emergency feedwater system with a long term safety grade supply of water. This assured source of feedwater has been evaluated and has adequate capacity for power uprate conditions. Additionally, the condensate storage capacity is based on maintaining hot standby conditions for one hour followed by a cooldown to hot shutdown conditions.

As documented in the "Supplemental Safety Evaluation for the Arkansas Nuclear One Units 1 and 2 (ANO-1&2) Station Blackout Rule (10 CFR 50.631) (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.

NRC Question 3

In Table 3-1, the differences between the minimum TS values and the minimum analytical values of the reactor coolant system (RCS) flow rate, core inlet temperature, and pressurizer pressure are very small. Please discuss the method used to determine the uncertainties of these parameters.

ANO Response

ANO recently submitted information concerning its instrument setpoint methodology in a letter dated June 26, 2001 (2CAN060107). The primary reason the instrument uncertainty allowances are small is due to the fact that the parameters (RCS flow rate, core inlet temperature and pressurizer pressure) are being monitored for initial pre-accident RCS conditions under normal environmental conditions. Therefore, higher uncertainties associated with an accident are not applicable to the times the parameters are checked for technical specification compliance.

The RCS flow limit in technical specification 3.2.5 is based on the analytical assumptions. A more restrictive limit, which accounts for instrument uncertainties is implemented in procedures to ensure the minimum RCS flow limit is protected. The technical specification limits for the reactor coolant cold leg temperature (technical specification 3.2.6) and pressurizer pressure (technical specification 3.2.8) are based on the analytical limits adjusted for instrument uncertainties. The bases for technical specification 3.2.8 clarify this approach for pressurizer pressure. Safety analyses cover a pressure range from 2000 psia to 2300 psia. The upper and lower allowable limits (2275 and 2025 psia) are adjusted by 25 psi to bound pressure instrumentation measurement uncertainty. In a similar fashion, although not clarified in the bases, reactor coolant cold leg temperature is based on safety analyses assuming a range from 540 to 556.7 °F. The upper and lower allowable limits (542 and 554.7 °F) are adjusted by 2 °F to bound instrumentation measurement uncertainty. The above approach for accounting for instrument uncertainties on RCS flow, pressurizer pressure and cold leg temperature is currently utilized for ANO-2 and is consistently used in the power uprate efforts.

NRC Question 4

In Section 6.4.5, the licensee stated that Table 6-6 presents transient lifetime occurrences for test conditions. Leak testing is covered under Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Section XI permits leak tests in lieu of hydrostatic tests. Consequently, the hydrostatic tests are no longer *required to be analyzed for fatigue requirements. The licensee also stated that since leak testing at nominal operating pressure is done in conjunction with normal plant operation, there is no requirement to analyze leak testing with respect to fatigue considerations, except for the special secondary side tests associated with the steam generator. There is no discussion of how the results and measurements of these tests will be acceptable for the proposed power uprate. Provide such a discussion with regard to the fatigue usage and leak considerations.*

ANO Response

The replacement steam generators (RSGs) were subjected to a primary side pressure test after installation in accordance with ASME Section XI Code Case N-416-1, "Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2 and 3 Section XI, Division 1." In addition, a post-outage pressure test will be performed after each refueling outage, in accordance with the requirements of ASME Section XI, Table IWB-2500-1, category B-P ('92 Edition, with portions of '93 Addenda, wherein the requirement for 10 year interval hydrostatic testing has been replaced by a system leakage test following each refueling outage). All of these tests are performed at nominal RCS operating pressure (2200 psia). Fatigue considerations of these leakage tests are therefore accounted for in the plant heatup and cooldown transients listed in Table 6-1 of the Power Uprate Licensing Report (Enclosure 5 to letter 2CAN120001 dated December 19, 2000). Power uprate is not affecting nominal RCS operating pressure, and accordingly, there is no effect of power uprate on RCS leak/pressure testing or its associated fatigue analysis.

The nominal operating pressure of the secondary side of the RSGs is increasing as a result of power uprate. Re-rated items are subjected to a pressure test at nominal operating pressure for the new service condition, if the resulting test pressure would be higher than the pressure of previous pressure tests (including construction pressure tests). In this case, the construction pressure test would bound the re-rated condition of the secondary side. This is consistent with later editions of Section XI which have been approved by the NRC. Secondary side pressure testing was also performed after RSG installation in accordance with ASME Section XI Code Case N-416-1. This test was performed after RCS heatup, prior to power operation, at which time the steam pressure is higher than normal operating pressure and thus is bounding for power uprate operating pressure. In addition to the post installation pressure test, system leakage tests are performed on the secondary side at nominal operating pressure in accordance with the requirements of ASME Section XI, Table IWC-2500-1, category C-H. Fatigue considerations of these leakage tests are accounted for in the plant heatup and cooldown transients, and accordingly, there is no effect of power uprate on fatigue usage or leak testing of the RSG secondary side.

The other test conditions relative to the RSG design are the shop hydrostatic tests (primary and secondary) and the tube leak pressure tests as listed in Table 6-6 of Enclosure 5, Power Uprate License Report, of the power uprate license application dated

December 19, 2000 (letter 2CAN120001). The primary and secondary shop hydrostatic test pressures are determined by the system design pressures, which are not being changed as a result of power uprate. The tube leak tests are performed during shutdown conditions, and thus are not related to power level.

In summary, there are no effects of power uprate on RSG primary or secondary leak testing, and the associated fatigue analyses are bounding for uprate operation.

NRC Question 5

In Section 6, there are tables which list number of cycles for various plant transients during the life of the plant for the purpose of mechanical design. Please compare these data with the current design basis associated with the original steam generators and discuss the reason of the changes.

ANO Response

The following table compares the replacement steam generator (RSG) and original steam generator (OSG) design transients that are different, and provides notation to explain the basis for the difference.

Transient	OSG Cycles	RSG Cycles	Basis Note #
Plant heatup, 100 °F/hr	500	350	1
Plant cooldown, 100 °F/hr	500	350	1
Plant loading, 5%/min	15,000	12,000	1
Plant unloading, 5%/min	15,000	12,000	1
10% step load increase	10^6	2,000	2
10% step load decrease		2,000	2
Normal plant variation		10^6	2
Cold FW following hot standby	Supplier to determine maximum no. cycles	Info supplied in original PUR application, Table 6-2	3
Loss of FW flow	8	20	4
Hydrostatic test, primary	10	1	5
Hydrostatic test, secondary	10	1	5
Primary side leak test	200	NA	6
Secondary side leak test	200	NA	6
Tube leak tests (Cases 1-4), described in original PUR application, Table 6-6	NA	Case 1-400 Case 2-200 Case 3-120 Case 4-80	6

Note 1: These key transients had significant impact on RSG fatigue analyses and were adjusted to provide an equivalent number of cycles for an RSG design life of 40 years vs. an

assumed life extension design life for the rest of the RCS of 60 years (i.e., the remainder of the RCS, other than RSGs, maintains the original steam generators' (OSGs') number of design cycles).

Note 2: The OSG design combined the step load increase/decrease transients with the normal plant variation transient (total of 10^6 cycles). For RSG design, the normal plant variation cycles were kept the same as the combined cycles for the OSG, and the description of the transient was adjusted (variations in feedwater temperature with step load changes, previously ignored, was taken into account for RSGs; manway stud/inspection cover bolt variations were adjusted based upon plant operating data as follows: primary pressure +/- 50 psi, temperature +/- 5 °F; secondary pressure +/- 40 psi, and feedwater temperature +/- 25 °F). Additionally for RSG design, the step load increase/decrease transients were broken out as separate transients (2,000 cycles each) in order to reflect the different thermal hydraulic performance of the RSGs, and to account for power uprate.

Note 3: The original definition of this transient did not effectively describe the various operating modes (i.e., feedwater flowrates, durations, cycles, etc.) that later evolved for operating in hot standby and low power modes. Plant operating data was reviewed and the different operating scenarios were characterized such that the analyses of the feedwater nozzles reflect the actual expected operating modes of the plant with RSGs installed.

Note 4: The original number of cycles assumed was non-conservative, and was adjusted accordingly based upon historical plant performance.

Note 5: The OSG design assumed periodic hydrostatic testing (primary and secondary) would be required over the life of the unit, per Section XI requirements at the time. Subsequent Section XI Code requirements allow in-service leak testing in lieu of hydrostatic testing as discussed in the response to question #4. The RSG hydrostatic test condition is for the testing performed by the RSG manufacturer in their shop. Thus the number of hydrostatic tests was reduced from 10 to 1.

Note 6: OSG leak testing modes were eliminated since they do not reflect the current methodology used to test for leakage. Current normal operating leak testing is described in the response to question # 4. The RSG leak testing listed in Table 6-6 represents the expected mode of any future required tube leak testing. The four cases represent step increases in secondary side pressure that might be required to find very small leaks.

NRC Question 6

Please expand Section 7.3 to address all changes of reactor protection system (RPS) trip delays, including the reasons of the changes. Confirm that the changes of RPS trip delay have been factored in all the re-analyses of affected events with acceptable consequences.

ANO Response

The changes in reactor protection system trip delays are discussed in Section 7.3 of the Power Uprate Licensing Report and in the respective accident analysis sections under the "Analysis Overview" subsection. The trip delays assumed are consistent with those defined in Table 7.3.0.1-2 or conservative with respect to these values. The values in Table 7.3.0.1-2 are consistent with those in Final Safety Analysis Report (FSAR) Table 15.1.0-1. Changes in the trip delays as defined in the power uprate submittal are typically increases in response times to add conservatism to the analysis and allow for potential future increases in response times. Or, the changes are increases to make the analysis assumptions consistent with the limits defined in Table 7.3.0.1-2 and FSAR Table 15.1.0-1. For example, the feedwater line break (FWLB) trip delay on high pressurizer pressure was increased to 0.9 seconds for conservatism. The current limit as defined by Table 7.3.0.1-2 is 0.65 seconds. This value was reduced to 0.65 seconds as part of the replacement steam generator effort to gain margin in the analysis results. The analysis assumption for the FWLB analysis has been restored to 0.9 seconds although some analyses are still based on 0.65 seconds, which is the limit defined in FSAR Table 15.1.0-1.

NRC Question 7

Please confirm that all computer codes (CENTS, HERMITE, etc.) used in the reanalyses have been reviewed and approved by NRC for their applicability at ANO-2. Provide a discussion that explains how all limitations have been satisfied.

ANO Response

Section 7.3.0.4 of the Power Uprate Licensing Report discusses the computer codes used to evaluate the non loss-of-coolant accident (LOCA) analyses. These computer codes (CENTS, HERMITE, CETOP, TORC and STRIKIN-II) have been reviewed and approved by the NRC. References to the approved topicals can be found in Section 7.3.19 of the Power Uprate Licensing Report. The CENTS code topical report is listed in Reference 7.3-2. This methodology was approved for use at ANO in Amendment 182 and included in technical specification 6.9.5.1 as a reference for the core operating limits report (COLR). HERMITE is used in the current analysis of record for the 4-pump loss of flow analysis and the approved topical is noted in Reference 7.3-4. Approval for the use of HERMITE at ANO is documented in Amendment 190. CETOP is used in the departure from nucleate boiling ratio (DNBR) and DNB thermal margin analyses. Reference 7.3-3 is an approved topical for the use of CETOP at ANO. The TORC code was used in the pump shaft seizure event and approved in Reference 7.3-7. STRIKIN-II is used in the control element assembly (CEA) ejection analysis, which is consistent with the current analysis methods. The STRIKIN-II code is documented in Reference 7.3-9. The implementation of STRIKIN-II into the CEA ejection analysis is covered in

Reference 7.3-11, which is also currently a reference in technical specification 6.9.5.1 as a COLR reference.

The above methods were used in the non-LOCA analyses to support the ANO-2 power uprate effort. Verification of proper implementation with consideration of the methodology limitations was performed as part of the development of the calculations to support the non-LOCA analyses.

NRC Question 8

Please address the following areas regarding the reactor coolant pump (RCP) shaft seizure accident described in Section 7.3.5:

a) *Explain why a concurrent loss of offsite power is not assumed with a RCP shaft seizure.*

b) *Describe the method used to determine the amount of failed fuel and state the number of failed fuel in this event.*

ANO Response

a) A concurrent loss of off-site power was not considered in the original licensing analyses for ANO-2; hence, it was not considered during the power uprate effort.

b) The methods used to determine the amount of failed fuel are defined in Section 7.3.5.2.4. The results in Figure 7.3.5.2-6, which present minimum DNBR for fuel pins of various radial peaks, will be used to determine the number of pin failures. This figure is used in conjunction with the number of fuel pins in the core having any given radial peak. The probability of DNB versus DNBR is overlaid with the above information to determine the total fuel failure based on DNB. Although, the Cycle 16 reload efforts are not complete at this time, it will be verified that the total fuel failures will be less than 14%.

NRC Question 9

Provide the methods used in determining the allowable power level with inoperable main steam safety valves.

ANO Response

The methods used to determine the allowable power level with inoperable main steam safety valves is defined in Section 1.4.1 of Enclosure 4 to our letter dated November 29, 1999 (2CAN119901). The methods and analyses presented in the November 29, 1999, letter are utilized to define the new allowable power levels. Technical Specification Table 3.7.1 and Figure 3.7-1 are based on a percentage of rated thermal power. Each of the data points in Table 3.7.1 and Figure 3.7-1 of the technical specifications is supported by an explicit evaluation of the loss of condenser vacuum event based on an initial thermal power. The technical specification limits reflect the ratio of the analysis assumed initial thermal power to the rated thermal power. No new analyses were performed to support power uprate. The proposed technical specification Table 3.7.1 and Figure 3.7-1 limits are developed from the initial thermal power assumptions for the analyses discussed in the November 29, 1999, letter and adjusted by the uprated power level.

NRC Question 10

Please address the following areas regarding the feedwater line break accident analysis described in Section 7.3.11.2:

a) Explain the need for the change in methodology for determining the most limiting break size. Provide discussion on why the feedwater line break analysis submitted by your letter dated November 29, 1999 (Enclosure 4, Page 40 of 172) is no longer valid.

b) Explain why the proposed method would [be] able to determine a most limiting break size which could bound the spectrum of potential break sizes including a double ended main feedwater line break.

c) Is the proposed method of determining the most limiting feedwater line break size consistent with that used in the Combustion Engineering (CE) System 80+ design? Has the proposed method been applied in any other CE-designed pressurized water reactors? Provide the citation for staff approval of the revised methodology and its applicability to ANO-2.

d) Discuss the instrument used in the RPS to initiate a reactor trip on low water level (with 40,000 lbs of water remaining) in the failed steam generator. Is this level measurement reliable during the dynamic transient conditions of a steam generator?

e) Discuss the single failure assumed in the feedwater line break analysis.

ANO Response

a) The analysis for the replacement steam generator effort (see letter 2CAN119901 dated November 29, 1999) was not performed at the uprated power level; therefore a new analysis was necessary. The only change in determining the limiting break size relates to the new assumption of crediting the low-level trip in the affected steam generator. 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. As a result of this new method a break spectrum was assessed. The new break size of 0.1492 ft^2 is only slightly smaller than the current limiting break size of 0.1798 ft^2 assumed in the replacement steam generator effort.

b) We have looked at a range of break sizes as shown in Figure 7.3.11.2-1 demonstrating the bounding nature of the smaller break sizes.

c) See the response to Question 15 in our letter dated October 17, 2001 (2CAN100110).

d) See the response to Question 15 in our letter dated October 17, 2001 (2CAN100110).

e) A single failure of an emergency feedwater pump is assumed consistent with the current analysis assumptions.

NRC Question 11

Please expand Section 7.3.13 to discuss the following:

a) The most limiting single failure assumed in the steam generator tube rupture (SGTR) analysis.

b) Confirm the operator actions assumed in the SGTR analysis are consistent with emergency operating procedures at ANO-2.

c) Describe operator actions relative to steam generator overfill during a SGTR event.

ANO Response

a) ANO-2 does not consider single failures for the SGTR event consistent with the original licensing basis analysis.

b) Although the emergency operating procedures are not written explicitly to require it, the ANO-2 Operations staff is trained to cool the ruptured steam generator to less than 520 °F T_{hot} and isolate the steam generator within 30 minutes of diagnosis of a steam generator tube rupture. The initiating time is generally considered to begin when the event is diagnosed following completion of standard post trip actions (SPTAs). The diagnosis and SPTAs typically require 10-15 minutes to complete. Even if the initiating time was considered to be when the rupture actually occurs (leak greater than charging pump capacity), then the time to isolate the steam generator would be well within 60 minutes. The operations crews are graded on their ability to accomplish this task during evaluated simulator sessions. In addition to the 30-minute operator action analyses presented in the power uprate submittal, an analysis based on 60 minutes was also evaluated. Operator response within 1 hour is sufficient time to diagnose this event and secure the affected steam generator.

The CENTS code was run for the 60-minute case consistent with the presented analysis at 30 minutes, with the assumption that the operator secured the affected steam generator within 1 hour. The results of this analysis with respect to the dose considerations had the following differences:

Table 1
Comparison on EAB and LPZ Radiological Dose Input

Parameter	Units	Operator Action @ 30 minutes	Operator Action @ 60 minutes
8 hour cooldown rate	°F/hr	35.5	38
Primary to Secondary Leakage	lbm	70,000	120,400
Secondary Release Prior to Operator Action	lbm	250,000	362,000
Total Secondary Release (2 hrs)	lbm	675,000	635,000
Total Secondary Release (8 hrs)	lbm	1,772,000	1,772,000
Flashing Fraction		see Table 2	see Table 3

Table 2
Flashing Fraction, Mass Release and Time Interval for Operator Action at 30 minutes

Time Interval, seconds	Flashing Fraction. %	Flashing Fraction Mass Release, lbm
0 to 60	[ ]	[ ]
60 to 1800	[ ]	[ ]
	total	70,000

Table 3
Flashing Fraction, Mass Release and Time Interval for Operator Action at 60 minutes

Time Interval, seconds	Flashing Fraction. %	Flashing Fraction Mass Release, lbm
0 to 60	[ ]	[ ]
60 to 300	[ ]	[ ]
300 to 700	[ ]	[ ]
700 to 900	[ ]	[ ]
900 to 1300	[ ]	[ ]
1300 to 1400.3	[ ]	[ ]
1400.3 to 1800	[ ]	[ ]
1800 to 2554.9	[ ]	[ ]
2554.9 to 2700	[ ]	[ ]
2700 to 3200	[ ]	[ ]
3200 to 3600	[ ]	[ ]
	Total	120,400

The SGTR with concurrent loss of alternating current (LOAC) power event results for the 30 minute and 60 minute operator actions are documented in Table 4. The 60 minute operator action results for the exclusion area boundary (EAB) and low population zone (LPZ) thyroid no-spiking, generated iodine spike (GIS), and pre-existing iodine spike (PIS) dose values are slightly higher than the 30 minute operator action time dose values. Also, the 60 minute operator action EAB whole body dose no-spiking, PIS, and GIS dose values are slightly higher than the 30 minute operator action time dose values. Although the results are slightly higher based on a 60 minute operator response time, the radiological dose results from the SGTR with Concurrent LOAC are within their respective criterion.

Table 4
Comparison of 30 Minute versus 60 Minute Operation Action

	Thyroid Dose (rem)		**Whole Body Dose (rem)**	
Event	**EAB**	**LPZ**	**EAB**	**LPZ**
30 minute SGTR case results				
SGTR – No Spiking	1.4	<0.1	0.6	<0.1
SGTR – PIS	70.0	3.5	0.9	<0.1
SGTR – GIS	21.4	1.2	0.7	<0.1
60 minute SGTR case results				
SGTR – No Spiking	1.5	<0.1	1.	<0.1
SGTR – PIS	73.2	3.6	1.3	<0.1
SGTR – GIS	30.	1.7	1.1	<0.1

c) Steam generator overfill is not a significant issue for ANO-2 due to the large secondary volume in the steam generators. The results of the loss of AC SGTR analysis performed to support 60-minute operator action reflects a maximum steam generator inventory of 40 feet at the end of the first hour, which is below the narrow range upper reference tap height of 41.5 feet. The volume above the upper reference tap height is approximately [ ]. At the end of an hour the leak rate is about 33 lbm/sec assuming no operator action. Based on this leak rate and assuming an inventory in the steam generator at the upper tap location, it will take at least an additional 35 minutes to finish filling the steam generator with liquid. However, operator action within 1 hour to secure the primary to secondary leakage will prevent the steam generator inventory from reaching the upper reference tap height.

NRC Question 12

To show that the referenced generically approved loss-of-coolant accident (LOCA) analysis methodologies apply specifically to ANO-2, provide a statement that ANO-2 and *its vendor have ongoing processes which assure that LOCA analysis input values for peak cladding temperature-sensitive parameters bound the as-operated plant values for those parameters.*

ANO Response

The emergency core cooling system (ECCS) performance analysis for ANO-2 documented in the December 19, 2000, license application was performed to conservatively bound the expected consequences of a LOCA. The analysis was performed according to the approved CENP evaluation models and conforms to the conservative deterministic methods outlined in 10 CFR 50, Appendix K, "ECCS Evaluation Models." By the nature of the evaluation model, the analysis provides substantial margin over realistic conditions that will bound variances in sensitive parameters.

In addition to the inherent conservatism in the CENP evaluation models, significant analysis input parameters were selected to bound as-operated plant values including instrument drift, uncertainties and inaccuracies. The key parameters used in the ANO-2 ECCS performance analysis are listed in Tables 7.1.3-1 and 7.1.4-1 of the Power Uprate Licensing Report. The parameters in these tables consist of parameters controlled by the core operating limiting supervisory system/core protection calculators (COLSS/CPCs), fuel specific parameters, and parameters controlled by technical specifications. For the parameters monitored by COLSS/CPC, reload specific uncertainties are applied to ensure these parameters are bounded by the safety analysis. The values controlled under technical specifications are monitored to ensure the as-operated plant values are bounding by considering instrument drift, uncertainties, and inaccuracies. One discrepancy on the control of the RWT temperature does exist. This issue was identified as part of our station corrective action program and resolution will be controlled consistent with this program. Current operating restrictions do, however, ensure that the LOCA assumptions, including uncertainties and inaccuracies, are bounded. The fuel specific parameters are determined consistent with the approved ABB-CE methodology. Other parameters, such as RCS pressure, are used at their nominal/reference point, since peak cladding temperature (PCT) is not sensitive to variation in these parameters.

ANO-2 uses a reload specific "groundrules" process with the non-physics assessment checklist (NPAC) to ensure the key safety analysis (including small and large break LOCA) input parameters and assumptions remain bounding on a cycle-to-cycle basis. This process assures that any proposed, or actual, changes in plant configuration are appropriately verified to remain bounded by the safety analysis.

NRC Question 13

The ANO-2 power uprate submittal references CENPD-137, Supplement 2-P-A, April 1998, as the generically approved small-break LOCA (SBLOCA) methodology as the one which will become the methodology to be included in licensing documentation and which *was used to perform the ANO-2 SBLOCA licensing analyses for the uprated power. The NRC approved CENPD-137, Supplement 2-P-A invoking [sic] unique criteria for the specific methodology and the then-existing or then-proposed plant conditions. Show that this methodology continues to be applicable to ANO-2 at the uprated power.*

ANO Response

The methodology described in the NRC-approved topical report CENPD-137, Supplement 2-P-A, April 1998 and referred to as the S2M methodology was reviewed and approved by the NRC for the ECCS performance analyses of the SBLOCA transient of Combustion Engineering (CE) designed plants. The ANO-2 CE designed plant is included under the full range of operating conditions allowed by the methodology. ANO-2 applied this approved methodology for the power uprate effort (see Reference 7.1-27 on page 7-12 of the Power Uprate Licensing Report). Additionally, as indicated above in response to Question 12, for peak cladding temperature (PCT) sensitive parameters, conservative analysis input values are verified through an ongoing process between ANO and CENP to bound the as-operated plant values for those parameters. For example, the high pressure safety injection (HPSI) pump surveillance requirements defined in technical specification 4.5.2 do not account for instrument uncertainties. This detail is clarified in the Bases; hence, procedures account for uncertainties in measured parameters prior to comparison to the technical specification limits.

The last application of the CENPD-137, Supplement 1-P methodology (S1M) to the ANO-2 plant yielded a peak clad temperature (PCT) of 2011 °F. The initial application of the S2M methodology to ANO-2 accompanied the request for technical specification changes to support increasing the main steam safety valve (MSSV) tolerance. This analysis yielded a PCT of 1798 °F, an improvement of 213 °F over the previous analysis.

The next application of the S2M methodology for ANO-2 was associated with the analysis of RSGs at the current power level. This analysis yielded a PCT of 1905 °F. The increased heat transfer area and primary side water volume due to the RSGs would lower the resulting PCT compared to the PCT obtained from 30% tube plugging case. However, the analytical HPSI flow rate used in the analysis was lowered compared to the previous application to include some additional margin for future HPSI flow measurements. The power uprate analysis, yielding a PCT of 2066 °F, indicates a 161 °F PCT increase relative to the RSG analysis, due to the 7.5% power uprate.

Therefore, the power uprate analysis PCT of 2066 °F, when compared to the last S1M analysis PCT of 2011 °F, reflects the use of the S2M methodology revised heat transfer margin gain plus the gain from additional RSG heat transfer area and primary side water volume relative to OSGs with 30% plugging. These gains are used to help offset the effects of a 7.5% power uprate, a lower analytical HPSI flow, and higher uncertainties on MSSV opening setpoints.

Version	Power (MWt)	MSSV Tolerance (%)	HPSI Flow @ 990 psia	Limiting PCT (°F)
S1M – 30% Plugging	2900 (1)	1	Base	2011
S2M – 30% Plugging	2900	3	-3.2 %	1798
S2M – RSG	2900	3	-5.9 %	1905
S2M – Power Uprate	3087(2)	3	-5.9 %	2066

Notes:

(1) nominal +3% power measurement uncertainty
(2) nominal +2% power measurement uncertainty

NRC Question 14

Page 1 of the cover letter, last paragraph: Has the Westinghouse Topical Report WCAP-10263 been approved by the NRC? If not, please provide technical justification (quantitative and qualitative) for its selection.

ANO Response

Westinghouse Topical Report WCAP-10263 has not been approved by the NRC; however, it was used as guidance for the ANO-2 power uprate for the following reasons:

1) SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," dated February 18, 1997, recommends its use. Section 3, "Power Uprate Review Process" provides guidance to the NRC staff for the review and approval of power uprate requests from licensees. In particular, section 3.5, "Recommendations," states in part that the scope and depth of review of uprate applications should be based on "... (3) uprate submittals that were based on the GE and Westinghouse topical reports on uprates."

2) Several power uprates have been approved by the NRC for licensees who utilized this document as guidance when performing their power uprate analyses. The following is an excerpt from the introduction of the NRC Safety Evaluation issued in response to Commonwealth Edison's (ComEd) Byron and Braidwood power uprate license applications (Amendment 119 for Byron, 113 for Braidwood): "ComEd stated that the power uprate analyses were performed consistent with the guidelines set forth in Westinghouse Energy Systems Report, WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant." This WCAP methodology, although not formally reviewed and approved by the NRC, was followed by North Anna, Salem, Indian Point 2, Callaway, Vogtle, Turkey Point, and Farley for their core power uprates, and those uprates were found to be acceptable."

3) Based on several discussions with the NRC staff during the early stages of preparation of the ANO-2 license application, the staff recommended using the Farley license application as a template when preparing the ANO-2 application. Farley utilized the Westinghouse topical during the preparation of their power uprate license application. The practice of using the Farley power uprate application as a template is discussed in the Byron and Braidwood power uprate NRC safety evaluation and also in SECY-01-0124, "Power Uprate Application Reviews," dated July 9, 2001.

NRC Question 15

Page 7-105, Section 7.3.0.1, list of input parameters: Please provide technical justification for Items 2 and 3.

ANO Response

Item 2 relates to power measurement uncertainties changing from 3% to 2%. This question is similar to question 7 of the first set of questions from the Reactor Systems Branch (see the response in letter 2CAN100110 dated October 17, 2001). 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 for an Appendix K LOCA analysis. The actual instrument uncertainty associated with the power measurement for ANO-2 is less than two percent.

Item 3 relates to a change in linear heat rate. The basis for the change in linear heat rate is discussed on page 11 of the attachment to letter 2CAN120001. A combination of the LOCA and non-LOCA analyses described in Sections 7.1 and 7.3 of the Power Uprate Licensing Report assume the new 13.7 kW/ft limit. Typically, the LOCA analysis is limiting with respect to linear heat rate limit considerations. A value of 13.7 kW/ft was assumed in this analysis as indicated in Tables 7.1.3-1 and 7.1.4-1. As indicated in Table 7.3.0.1-1, a value of 13.7 kW/ft was assumed in the non-LOCA analyses discussed in Section 7.3 of the Power Uprate Licensing Report when applicable.

The limiting consideration for linear heat rate is discussed in Section 8.3.1.5 of the Power Uprate Licensing Report. The maximum fuel rod internal pressure analysis also assumed a linear heat rate of 13.7 kW/ft until 50 gigawatt days per metric ton of uranium (GWD/MTU) and a lower value of 13.0 kW/ft for higher burnups.

NRC Question 16

The plant parameter changes stated in the last paragraph of page 7-105 (and continuing on to page 7-106): The first four changes were not provided with any technical bases. Please provide technical justifications (quantitative and qualitative) for the selection of these parameters.

ANO Response

The list of items on page 7-105 and 7-106 of the Power Uprate Licensing Report is a summary of major plant parameter changes discussed later in the submittal as part of the respective analyses. These limits were extracted from the report to help highlight plant parameter and input assumption changes used in the accident analyses discussed in Section 7.3. The first item listed relates to the increased operator response time (1 hour to 2 hours) allowed for a CEA misalignment event for inward deviations (CEA drop). The justification for this change is discussed in Section 7.3.3 of the Power Uprate Licensing Report. In this section of the report, a re-evaluation of the CEA misoperation event is presented with the 2-hour operator response time incorporated into the analysis. This 2-hour time frame is incorporated into COLR Figure 2. As discussed on page 12 of the attachment to the power uprate license application dated December 19, 2000, the increased time allows for better operator control of the ramp and reduces the risk of a reactor trip.

The second item relates to the scram worth trade-off for hot zero power (HZP) which is discussed in Section 7.3.11. A trade-off study of CEA worth was performed for the HZP main steam line break analysis. This study indicated that an incremental shutdown margin of 1.29% $\Delta\rho$ can be credited in future HZP analyses without exceeding the DNBR limit or peak linear heat rate limit.

The third item relates to the conservative assumption used for charging flow in the boron dilution event discussed in Section 7.3.4. No credit is taken for charging flow in the Non-LOCA analyses discussed in Section 7.3 or the LOCA analysis is Section 7.1 of the Power Uprate Licensing Report. The boron dilution event is conservatively based on a maximum charging flow (3 pumps with a capacity of 46 gpm versus 44 gpm). Additional information with respect to the increased charging flow has been discussed in response to question 10 of the first set of questions from the Reactor Systems Branch (letter 2CAN100110 dated October 17, 2001). The increased charging pump flow was chosen to conservatively bound as-operated plant conditions.

The fourth item relates to the increased low reactor coolant pump shaft speed trip response time assumed in the loss of reactor coolant flow analysis discussed in Section 7.3.5.1. The FSAR 4-pump loss of flow analysis presented in Table 15.1.5-9 reflects a 0.3 second reactor coolant pump shaft speed trip response time. This response time was increased to 0.4 seconds in Cycle 15 as part of the RSG effort by an evaluation presented in letter 2CAN119901 dated November 29, 21999 (page 20 of Enclosure 4). The analysis presented in Section 7.3.5.1 of the power uprate submittal also incorporates this new response time of 0.4 seconds.

NRC Question 17

In the first sentence in the first paragraph on page 7-110, it is stated that the power uprate could result in a small degradation of the calculated thermal margin. How small is the degradation? How much margin is left?

ANO Response

The statement on page 7-110 relates to the small degradation of thermal margin in the uncontrolled CEA withdrawal from subcritical conditions discussed in Section 7.3.1. An explicit effort was not undertaken to assess the impact of only power uprate on the subcritical CEA withdrawal event. Rather, the combined effect of the changes discussed in Section 7.3.1 were considered in the analysis. The results of this analysis are discussed in Section 7.3.1.5 of the Power Uprate Licensing Report.

The results in Tables 7.3.1-2 and 7.3.1-3 can be compared to the results in FSAR Tables 15.1.1-5 and 15.1.1-6. The combined effects of the increase in power, increase in RCS flow and change in core design (Erbia burnable poison versus Gadolinia) resulted in an increase in thermal margin. A minimum DNBR of 1.4 was obtained for the 2.5×10^{-4} $\Delta\rho$/sec reactivity insertion rate (RIR) case at uprated conditions versus a DNBR of 1.27. For the 2.0×10^{-4} $\Delta\rho$/sec RIR case, a DNBR of 2.0 was calculated for the uprated conditions versus 1.42. The small degradation of thermal margin attributed to the increase in power has been more than offset by the increase in RCS flow assumption and core design.

NRC Question 18

On page 7-113, under the subheading of hot full power, item 5 states that a moderator temperature coefficient (MTC) of $0.0 * 10^{-4}$ Δp/°F is more conservative than a MTC of negative $0.2 * 10^{-4}$ Δp/°F at beginning-of-cycle. Please explain.

ANO Response

The hot full power CEA withdrawal event results in an increase in temperature. A less negative moderator temperature coefficient (MTC) is assumed in this analysis for conservatism to minimize the negative reactivity being added as the temperature increases. An MTC of 0.0×10^{-4} $\Delta\rho$/°F was assumed in this analysis rather than the limit of -0.2×10^{-4} $\Delta\rho$/°F as the negative reactivity feedback is lower using this assumption.

NRC Question 19

Item 6 on page 7-113 states that the response time was increased to 0.40 seconds. Please justify.

ANO Response

Some of the older analyses (Cycle 12 analysis discussed in FSAR section 15.1.2.4.2.1) used a response time of 0.39 seconds for the ex-core neutron detectors. A more conservative and bounding value used in current analyses is 0.4 seconds. This statement was made to clarify that the power uprate analyses were performed using the more conservative value of 0.4 seconds.

NRC Question 20

Item 7 on page 7-113: Was this reactivity insertion rate changed from a prior value, and why?

ANO Response

The reactivity insertion rate has not changed from the analysis of record. A reactivity insertion rate of 1×10^{-4} $\Delta\rho$/sec has been used in the hot full power CEA withdrawal analysis since Cycle 13.

NRC Question 21

On page 7-118, the first sentence states that the impact of the above changes result in no violation of the specified acceptable fuel design limits. Please explain. Also in the same paragraph, it states that acceptable limits were not violated. Please explain. What are these acceptable limits?

ANO Response

The first paragraph of Section 7.3.4.2 of the Power Uprate Licensing Report describes the acceptance criteria used in the boron dilution event. The purpose of the boron dilution analysis is to demonstrate that the specified acceptable fuel design limits (SAFDLs, centerline-to-melt and DNBR limits) are not violated. This is indirectly demonstrated by ensuring that the uncontrolled criticality does not occur within the specified times for operator action. In this way the centerline-to-melt and DNBR limits are not challenged. Section 7.3.4.2 also defines the acceptable time limits for operator action. For the dilution events initiated from subcritical conditions, the time from an alarm until the loss of shutdown margin must exceed 15 minutes or 30 minutes for events during refueling.

NRC Question 22

Item B in the first paragraph on page 7-121 states that credit was taken for the temperature difference between the modes. Please explain.

ANO Response

Dilution of the volumes noted in Tables 7.3.4-4a, 7.3.4-4b, 7.3.4-5a, and 7.3.4-5b for Hot Shutdown and Hot Standby is converted to the respective mass inventory based on the density associated with the mode. The colder temperatures associated with Hot Shutdown increases the mass, hence, relaxes the requirements.

NRC Question 23

On page 7-121, why was the minimum response time changed from 30 minutes to 31 minutes?

ANO Response

An additional minute was added to the acceptance criterion of 30 minutes for conservatism only. The analysis conservatively used 31 minutes versus 30 minutes. Using 31 minutes as an acceptance criterion for operator action, acceptable critical boron concentration/inverse boron worth limits are determined as discussed in Section 7.3.4.4.

Proprietary Affidavit Pursuant to 10CFR2.790 for Attachment 2 (1 page)

AFFIDAVIT PURSUANT TO 10 CFR 2.790

I, Philip W. Richardson, depose and say that I am the Licensing Project Manager, Westinghouse Electric Company LLC (WEC), 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 ENTERGY Operations, Inc. for withholding this information.

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

2CAN100102, Attachment 1 – "Response to NRC Request for Additional Information Nos. 11b and 11c", October, 2001

This document has been appropriately designated as proprietary.

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

Pursuant to the provisions of Section 2.790(b)(4) 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, is owned and has been held in confidence by WEC. It consists of Steam Generator Tube Rupture safety analysis methodology details.
2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to WEC.
3. The information is of a type customarily held in confidence by WEC and not customarily disclosed to the public. WEC 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.
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 WEC because:
   a. A similar product is manufactured and sold by major pressurized water reactor competitors of WEC.
   b. Development of this information by WEC required thousands of dollars and hundreds of man-hours of effort. 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 to develop Steam Generator Tube Rupture safety analysis methodology details.
   d. The information consists of Steam Generator Tube Rupture safety analysis methodology details, 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 WEC, take marketing or other actions to improve their product's position or impair the position of WEC's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.
   e. In pricing WEC's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of WEC's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.
   f. 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 WEC's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

Philip W. Richardson
Licensing Project Manager

Sworn to before me
this **22** day of **October**, 2001

Joan C. Hastings
Notary Public
My commission expires: **JOAN C. HASTINGS**
NOTARY PUBLIC
MY COMMISSION EXPIRES SEP. 30, 2002

Attachment 3

Response to Third Request for Additional Information from the Reactor Systems Branch Regarding the ANO-2 Power Uprate License Application

Response to Third Request for Additional Information from the Reactor Systems Branch Regarding the ANO-2 Power Uprate License Application

NRC Question

The recent experience from Calvert Cliffs has shown that the cladding corrosion is worse in the high-burnup regime and is consistently underestimated by the CENP corrosion model. Provide updated information of corrosion during power uprate and assess the potential impact for fuel operation at ANO-2.

ANO Response

Westinghouse CENP (W CENP) recognizes that recent high duty fuel performance data from the Calvert Cliffs, Waterford 3, and Palo Verde nuclear generating stations have indicated that OPTIN cladding corrosion for some high duty fuel rods is more adverse than originally expected. Increased corrosion and limited oxide spalling have been observed in recent high duty fuel inspections at Calvert Cliffs and Palo Verde and in past high burnup test assemblies at Calvert Cliffs, Palo Verde, and Waterford 3. Increased core crudding has also been observed in poolside measurements for high duty fuel at Palo Verde. As a result of these observations, preliminary models for predicting corrosion, the threshold for spalling, and steaming rates associated with crudding, which include consideration of the above-mentioned developments at Combustion Engineering plants, have been developed by W CENP. These preliminary models have been applied as needed to assess high duty operation of operating W CENP plants. The NRC has been made aware of these developments.

W CENP has reassessed the corrosion performance of ANO-2 under power uprated conditions with the new models, and has established and applied additional fuel management guidelines for corrosion to ANO-2 on a cycle-specific basis, beginning with the first uprated cycle. These fuel management guidelines limit the maximum oxide thickness, fuel duty, steaming rate and core crudding. The preliminary corrosion models discussed above were applied to assess conformance with these guidelines. Adjustments were made to the Cycle 16 core design to accommodate these new guidelines. This consideration required a notable change in the core loading pattern. In particular for ANO-2 Cycle 16, four fuel assemblies were added.

The corrosion performance assessment included analysis of select limiting power fuel rods (including assembly peripheral rods) from ANO-2 uprated fuel management depletions. The fuel management was constructed explicitly to model the more adverse expected core uprated operation. The preliminary models developed based on the high duty corrosion performance data were applied and show that predicted maximum oxide thickness is less than 100 microns. A 100 micron limit is imposed on other fuel vendors/cladding by the NRC and the limit is expected to be imposed on OPTIN cladding for low duty high burnup when CENPD-388-P is approved. CENPD-388-P "Extension of the 1-pin Burnup Limit to 65 MWD/kgU for ABB PWR Fuel With OPTIN™ Cladding" (February 1998) is a document developed by the Combustion Engineering Owners Group.

In summary, reassessment of the planned power uprate cores which use the new fuel management guidelines, utilizing the preliminary corrosion model which includes consideration of recent experiences at Combustion Engineering plants, shows acceptable corrosion performance for the planned ANO-2 uprate cores.

Attachment 4

Supplemental Information (Non-Proprietary) Regarding the ANO Response to NRC Question 18 from Letter Dated October 17, 2001 (2CAN100110)

Supplemental Information (Non-Proprietary) Regarding the ANO Response to NRC Question 18 from Letter Dated October 17, 2001 (2CAN100110)

In a letter dated October 17, 2001 (2CAN100110) Entergy Operations, Inc. provided responses to 22 questions from the NRC staff. On page 12 of 18 of the attachment to the letter, three differences were cited between the original (1975) methodology and the CENPD-254 methodology for addressing boric acid precipitation following a large break loss of coolant accident. The last sentence of item 2 stated that the mixing volume for the CENPD-254 methodology was different but provided no details because the details were considered proprietary information. In a follow-up telephone call with the NRC staff on October 23, 2001, the proprietary information regarding the mixing volume was discussed. NRC personnel stated that the additional details were needed to adequately resolve the Staff's question. Therefore, the paragraph is repeated below; however, additional details have been added to the end of the paragraph. Proprietary information is denoted with brackets [ ].

2. The two methodologies used different "mixing volumes". 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 equal to the volume corresponding to [ ] Note that in neither methodology is water in the [ ] included in the mixing volume.

Proprietary Affidavit Pursuant to 10CFR2.790 for Attachment 5 (2pages)

I, Norton L. Shapiro, depose and say that I am the Advisory Engineer of CE Engineering Technology, Westinghouse Electric Company LLC (WEC), duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and described below.

I am submitting this affidavit in conjunction with the application by Entergy Operations Incorporated and in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information. I have personal knowledge of the criteria and procedures utilized by WEC in designating information as a trade secret, privileged, or as confidential commercial or financial information.

The information for which proprietary treatment is sought, and which document has been appropriately designated as proprietary, is contained in the following:

- Enclosure 1 to letter LTR-OA-01-24 dated October 8, 2001

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

i. The information sought to be withheld from public disclosure is owned and has been held in confidence by WEC. It consists of details of the post-LOCA long term cooling analysis methodology for the power uprate for Arkansas Nuclear One, Unit 2 (ANO-2).

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

iii. The information is of a type customarily held in confidence by WEC and not customarily disclosed to the public.

iv. 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.

v. 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 that provide for maintenance of the information in confidence.

vi. Public disclosure of the information is likely to cause substantial harm to the competitive position of WEC because:

   a. A similar product is manufactured and sold by major competitors of WEC.

   b. Development of this information by WEC required tens of thousands of dollars and hundreds of manhours of effort. A competitor would have to undergo similar expense in generating equivalent long term cooling analysis methodology.

   c. The information consists of details of the post-LOCA long term cooling analysis methodology for the power uprate for ANO-2, the application of which provides WEC a competitive economic advantage. The availability of such information to competitors would enable them to design their product to better compete with WEC, take marketing or other actions to improve their product's position or impair the position of WEC's product, and avoid developing similar technical analysis in support of their processes, methods or apparatus.

d. In pricing WEC's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of WEC's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

e. Use of the information by competitors in the international marketplace would increase their ability to market comparable analytical services by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on WEC's potential for obtaining or maintaining foreign licenses.

Norton L. Shapiro
Advisory Engineer

Sworn to before me this 8 day of October, 2001

Notary Public
My Commission expires: ____________

JANEY BRUNO
NOTARY PUBLIC
MY COMMISSION EXPIRES APR. 30, 2006