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November 24, 1996

2CAN119610

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Technical Specification Change Request Concerning
Small Break Loss of Coolant Accident

Gentlemen:

Attached for your review and approval is a proposed Arkansas Nuclear One Unit 2 (ANO-2) Technical Specification amendment request which adds specifications to section 6.9.5. In order to support an increased steam generator tube plugging limit beyond 10%, reference to the small break loss-of-coolant accident methodology CENPD-137, Supplement 1-P and its approval letter are being added to section 6.9.5.1.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

Entergy Operations requests that the effective date for this change be upon issuance. Although this request is neither exigent nor emergency, your prompt review is requested.

A001/1

030085

Very truly yours,

CRH/nbm
Attachments

To the best of my knowledge and belief, the statements contained in this submittal are true.

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for Johnson **County and the State of Arkansas, this** 24 **day of** November**, 1996.**

Juana M. Tapp
Notary Public
My Commission Expires 11-8-2000

OFFICIAL SEAL
JUANA M. TAPP
NOTARY PUBLIC - ARKANSAS
JOHNSON COUNTY
My Commission Expires: 11 - 8 - 2000

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<u>ATTACHMENT</u>

<u>TO</u>

<u>2CAN119610</u>

<u>PROPOSED TECHNICAL SPECIFICATION</u>

<u>AND</u>

<u>RESPECTIVE SAFETY ANALYSES</u>

<u>IN THE MATTER OF AMENDING</u>

<u>LICENSE NO. NPF-6</u>

<u>ENTERGY OPERATIONS, INC.</u>

<u>ARKANSAS NUCLEAR ONE, UNIT TWO</u>

<u>DOCKET NO. 50-368</u>

DESCRIPTION OF PROPOSED CHANGES

Technical specification 6.9.5.1 lists the analytical methods used to determine the core operating limits. Reference to CENPD-137, Supplement 1-P, "Calculative Methods for the CE Small Break LOCA Evaluation Model" and its approval letter, respectively, are being added in specifications 6.9.5.1.9 and 6.9.5.1.13. In specification 6.9.5.1.8 a typographical error was corrected, and specifications 6.9.5.1.10 through 6.9.5.1.14 are being renumbered to accommodate these changes.

BACKGROUND

The current ANO-2 small break loss-of-coolant accident (SBLOCA) analysis is performed utilizing the methodology described in CENPD-137-P. This methodology is the original method with which ANO-2 was licensed. Very few changes have occurred over the past 11 cycles which have required the SBLOCA analysis to be revisited. In preparation for anticipated steam generator tube repairs and the potential reactor coolant system (RCS) flow reduction associated with these repairs, efforts were initiated early in 1996 to update the SBLOCA analysis. To support this effort, the approved SBLOCA evaluation model, CENPD-137, Supplement 1-P, was chosen as the preferred evaluation method. This methodology has been applied with a steam generator tube plugging limit of 30% and an associated 10% reduction in RCS flow.

The current ANO-2 SBLOCA analysis of record using CENPD-137-P supports a steam generator tube plugging limit of up to 10%. Due to an increased number of degraded steam generator tubes removed from service over the last few outages, the steam generator tube plugging limit will need to be increased above 10% in the very near future. Based on the results of the new SBLOCA analysis, a steam generator tube plugging limit beyond 10% can be established utilizing the CENPD-137, Supplement 1-P evaluation model. CENPD-137, Supplement 1-P has previously been reviewed and approved by the NRC and is the current ABB-Combustion Engineering (CE) SBLOCA analysis methodology.

Generic Letter 88-16 requires the NRC-approved methodologies to be listed in technical specifications for limits that are placed in the Core Operating Limits Report. Currently, CENPD-137-P is utilized as the ANO-2 SBLOCA methodology. The SBLOCA methodology CENPD-137, Supplement 1-P will be utilized in future ANO-2 safety analyses.

DISCUSSION OF CHANGE

The ANO-2 SBLOCA analyses have been performed by ABB-CE utilizing the current approved methods in CENPD-137, Supplement 1-P. The supporting calculations for the ANO-2 SBLOCA input parameters and analyses are maintained by ABB-CE in Windsor, Connecticut. Because this is a first time application for ANO-2, a plant specific input deck was developed. The input parameters generated in the development of the plant specific

ANO-2 model were formed utilizing standard ABB-CE practices. These input assumptions were reviewed to ensure the present plant design conditions were correctly modeled and potential future plant changes were considered.

The following table compares the relevant input changes made to the general system parameters. Conservative input parameters, with respect to the technical specification limits, were utilized when modeling parameters such as the high pressure safety injection (HPSI) response time, safety injection tank (SIT) pressures, RCS flow, and reactor power. As previously noted, the CENPD-137, Supplement 1-P analysis addresses a 30% steam generator tube plugging limit and a 10% reduction in RCS flow.

Input Parameter	CENPD-137-P	CENPD-137, Sup. 1-P	Units
Reactor power level	2882	2900	MWt
Peak linear heat generation rate (PLHGR)	16.0	13.5	kW/ft
Gap conductance at PLHGR	1666	1582	BTU-hr-ft^2-°F
Fuel centerline temperature at PLHGR	3828	3334	°F
Fuel average temperature at PLHGR	2382	2115	°F
Hot rod gas pressure	1206	1123	psia
MTC at initial density	+0.5 E-4	0.0 E-4	delta rho/°F
RCS flow rate	120.4 E+6	108.4 E+6	lbm/hr
Core flow rate	116.2 E+6	104.6 E+6	lbm/hr
RCS pressure	2250	2250	psia
Cold leg temperature	557.5	556.7	°F
Hot leg temperature	618.8	622.7	°F
Plugged tubes per steam generator	10	30	%
Low pressurizer pressure reactor trip setpoint	1625	≤1625	psia
Low pressurizer pressure SIAS setpoint	1625	≤1578	psia
Safety injection tank pressure	615	550	psia
HPSI response time	30	40	seconds

In accordance with the methodology established in CENPD-137, Supplement 1-P, the limiting conditions with respect to offsite power availability and single failures were assumed. A loss of offsite power coincident with a reactor trip is assumed. This condition has been determined to be limiting as it requires the HPSI pumps to wait until the emergency diesel generators startup and load-sequence the pumps. The limiting single failure of an emergency diesel generator to start is also assumed. Based on this assumption, only one train of emergency core cooling is available.

A break spectrum consistent with the limiting break sizes and locations in the Combustion Engineering Standard Safety Analysis Report (CESSAR) and Waterford-3 applications of CENPD-137, Supplement 1-P has been evaluated for ANO-2. From this spectrum, the new limiting break size for ANO-2 is a 0.05 ft^2 (previously 0.1 ft^2) break in the reactor coolant pump discharge leg. The change in limiting break size is attributed primarily due to the change in methodology. Based on the assumption of a reactor coolant pump discharge leg

break, only 75% of the HPSI pump flow is credited. The other 25% of the flow is assumed to spill out the break.

A consistent approach for all input parameters between break sizes was the original intent of this analysis effort; however, in order to ensure acceptable results the following deviations in input parameters were made. For the 0.05 ft^2 and the 0.06 ft^2 break sizes, the main steam safety valve (MSSV) first bank opening pressure was conservatively assumed to be 1125 psia versus the 1103.5 psia opening pressure assumed in the 0.02 ft^2 and 0.04 ft^2 break sizes. The 1103.5 psia setpoint is consistent with the current technical specification limit of 1078 psig plus a 1% tolerance. The low pressurizer pressure reactor trip and safety injection actuation system (SIAS) setpoints were conservatively assumed to be 1400 psia for the 0.02 ft^2, 0.05 ft^2, and 0.06 ft^2 break sizes. The low pressurizer pressure reactor trip was assumed to be 1625 psia, and the low pressurizer pressure SIAS setpoint was assumed to be 1578 psia for the 0.04 ft^2 break size. A low pressurizer pressure reactor trip and SIAS setpoint analytical limit of 1625 psia is currently used to define the 1717.4 psia technical specification value which accounts for instrument uncertainty. In all cases, the system parameters and initial conditions are within the current technical specification limits. The following table summarizes these deviations.

Parameter	TS Analysis Limit	0.02 ft^2	0.04 ft^2	0.05 ft^2	0.06 ft^2
MSSV setpoint (psia)	1103.5	1103.5	1103.5	1125	1125
Low pzr. press. reactor trip (psia)	1625	1400	1625	1400	1400
Low pressurizer press. SIAS (psia)	1625	1400	1578	1400	1400

The results are summarized in the attached rewrite of the applicable portions of the ANO-2 SAR. For the limiting break size, the peak cladding temperature (PCT) was 2011 degrees F which is below the 10CFR50.46 acceptance criteria of 2200 degrees F, the maximum cladding oxidation was 5.47% which is below the 10CFR50.46 acceptance criteria of 17%, and the core-wide cladding oxidation was less than 0.835% which is below the 10CFR50.46 acceptance criteria of 1.0%. It should be noted that the PCT, maximum cladding oxidation, and core-wide cladding oxidation values are greater than those calculated utilizing the current CENPD-137-P methodology (reference current ANO-2 SAR section 6.3.3.2.4); however, they remain within the limits of 10CFR50.46 and remain bounded by the current large break loss-of-coolant accident (LBLOCA) results. A comparison of the new results to the current SAR results is provided in the table below. The increase in the results can be attributed to the net effect of the input assumptions utilized being more conservative and the change in methodology. Although there has been no explicit effort to quantify the effects of the input changes versus the methodology change, it is believed that most of the increase in PCT relates to the change in methods, based on applications of the evaluation model at other plants.

Parameter	10CFR50.46 Acceptance Criteria	Current SBLOCA SAR Results	New SBLOCA Results	Current LBLOCA Results
PCT (°F)	2200	1460	2011	2142
Maximum cladding oxidation (%)	17	0.205	5.47	8.9
Core wide cladding oxidation (%)	1.0	< 0.027	<0.835	<0.843

Per 10CFR50.46 licensees are required to report significant changes (i.e., calculated PCT being different by more than 50 degrees F) to an acceptable evaluation model within 30 days. By utilizing CENPD-137, Supplement 1-P for the ANO-2 SBLOCA analysis, the PCT did increase by more than 50 degrees F; however, this increase is primarily attributed to the change in methodology. Because NRC approval to use a different evaluation model for the ANO-2 SBLOCA analysis is being requested, it is Entergy Operations position that the reporting criteria of 10CFR50.46 does not apply.

For additional information, a comparison of the ANO-2 results from the NRC-approved CENPD-137, Supplement 1-P (S1M) was made with the results from CENPD-137, Supplement 2-P (S2M) and CEN 420-P, the Realistic Evaluation Model (REM). Both the S2M and REM methodologies have been submitted to the NRC for review. The input parameters were effectively the same as were used in the S1M model for the 0.06 ft^2 and 0.05 ft^2 break sizes except the S2M model assumes a slightly more conservative HPSI pump performance curve (lower flow rates). The REM uses the same input parameters as the S2M model except a significantly more conservative HPSI pump performance curve is assumed, a more conservative moderator temperature coefficient was assumed, and 15% steam generator tube plugging (based on projected tube plugging four years ago) was modeled in the REM. The comparison of the PCT results are shown in the table below.

PCT (degrees F)

Break	S1M	S2M	REM
0.06 ft^2	2003	1666	1574
0.05 ft^2	2011	1798	1572
0.04 ft^2	1870	1704	1569
0.02 ft^2	1671	1564	1402

Based on this comparison, it is evident that the S1M results are based on a conservative modeling methodology.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change to reference CENPD-137, Supplement 1-P is administrative in nature. The current referenced SBLOCA methodology is being supplemented with a more recently approved methodology which has demonstrated acceptable results with respect to 10CFR50.46 for the ANO-2 SBLOCA analysis. CENPD-137, Supplement 1-P has been independently reviewed and approved by the NRC. Technical specifications will continue to require operation within the core operational limits for each cycle reload calculated by the approved reload design methodologies. Cycle-specific evaluations performed in accordance with 10CFR50.59 demonstrate that changes in fuel cycle design do not involve an unreviewed safety question. Although there is an increase in the results (PCT, maximum cladding oxidation, and core-wide cladding oxidation) of the SBLOCA analysis, the increase is primarily due to the methodology change. The more recently approved methodology allows steam generator tube plugging up to 30% for SBLOCA analysis, but the increase in the results due to steam generator tube plugging is very small when compared to the increase due to the methodology change. The safety analyses will continue to be performed utilizing NRC-approved methodologies, and specific reload changes will be evaluated per 10CFR50.59.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change to reference the current NRC-approved SBLOCA methodology is administrative in nature. The more recently approved methodology has demonstrated acceptable results for ANO-2. No changes to plant operating procedures or operating parameters are proposed. The safety analyses will continue to be performed utilizing NRC-approved methodologies, and specific reload changes will be evaluated per 10CFR50.59. No new equipment is being introduced, and no equipment is being operated in a manner inconsistent with its design.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed change to reference the NRC-approved CENPD-137, Supplement 1-P SBLOCA methodology is administrative in nature. The margin of safety as defined by 10CFR50.46 has not been significantly reduced. There is an increase in the results (PCT, maximum cladding oxidation, and core-wide cladding oxidation) of the SBLOCA analysis utilizing this methodology; however, the increase is primarily due to the methodology change and remains within the limits specified in 10CFR50.46. The more recently approved methodology allows steam generator tube plugging up to 30% for SBLOCA analysis, but the increase in the results due to steam generator tube plugging is very small when compared to the increase due to the methodology change.

The development of limits for a particular cycle will continue to conform to the methods described in NRC-approved documentation. Technical specifications will continue to require that the core be operated within these limits and specify appropriate actions to be taken if the limits are violated. Each reload undergoes a 10CFR50.59 safety review to assure that operation of the unit within the cycle-specific limits will not involve an unreviewed safety question. The safety analyses will continue to be performed utilizing NRC-approved methodologies.

Therefore, this change does <u>not</u> involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does <u>not</u> involve a significant hazards consideration.

PROPOSED TECHNICAL SPECIFICATION CHANGES