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July 24, 2001

2CAN070105

U. S. Nuclear Regulatory Commission Document Control Desk Mail Station OP1-17 Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 2 Docket No. 50-368 License No. NPF-6 Part 2 of the Probabilistic Safety Assessment Information to Support the ANO-2 License Amendment Request for Power Uprate

Gentlemen:

The purpose of this letter is to provide the remaining information requested by the NRC staff regarding the probabilistic safety assessment (PSA) portion of the license application dated December 19, 2000 (2CAN120001). The license amendment requests an increase in the authorized power level for Arkansas Nuclear One, Unit 2 (ANO-2) from 2815 megawatts thermal to 3026 megawatts thermal. A partial response was provided in a letter to the staff dated June 28, 2001 **(2CAN0601** 10). Information regarding the following risk areas was provided in the June 28, 2001, letter: 1) Level 1 Internal Events - Initiating Event Frequencies, 2) Level 1 Internal Events - Component Failure Rates, 3) Individual Plant Examination of External Events (IPEEE) Internal Fire Analysis, 4) IPEEE Seismic Analysis, 5) IPEEE Other External Events Analysis, and 6) Shutdown Risk.

The attachment contains the remaining information requested by the staff: 1) Level 1 Internal Events - Success Criteria and Operator Actions, including effects on core damage frequency, 2) Large early release fraction, and 3) Quality of the PSA Model.

Level 1 Internal Events - Success Criteria

A detailed review was performed to identify the effect of the 7.5% increase Arkansas Nuclear One - Unit 2 (ANO-2) thermal power level on the system success criteria credited in the ANO-2 Probabilistic Safety Analysis (PSA) Level-i Internal Events model. These success criteria specify the requirements of the plant systems to address critical safety functions.

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U. S. NRC July 24, 2001 2CAN070105 Page 2

With one exception, the power uprate was determined to have no impact on the success criteria associated with each of these safety functions. The one exception is the high pressure safety injection success criteria for long term reactor coolant system heat removal following a large break loss-of-coolant accident. The success criterion was increased from one high pressure safety injection pump injecting into the reactor coolant system through two of four open injection flow lines (one of which may contain a pipe break) for the current power level to one high pressure safety injection pump injecting into the reactor coolant system through three of four open injection flow lines (one of which may contain a pipe break) for the uprated power level. The difference is due to the slightly larger decay heat levels at the time of recirculation. As a result of this assessment, the effect of the 7.5% Power Uprate was estimated to increase the internal events core damage frequency from 1.70E-05/reactor-year to 1.97E-05/reactor-year (an increase of about 16%).

#### Level 1 Internal Events - Operator Actions

Power uprate impacted the operator failures in that it had the general effect of reducing the time available for the operator to complete the recovery action. This change is due to the higher decay heat level after power uprate. Calculations were performed and computer modeling was used to estimate the time available for operators to complete required actions. For conservatism, core uncovery time, not core melt time, was selected as the basis for available time for operator action. Since core uncovery preceeds core damage, this approach yields conservatively high operator failure probabilities for operator actions tied to averting core damage. Therefore, the operator failure probabilities applied in the assessment of the risk impact of the power uprate are conservatively high.

#### Large Early Release Fraction

The 7.5% Power Uprate increases the large early release fraction roughly proportionally to the increase in the core damage frequency. The large early release fraction value remains at approximately 2% of the new core damage frequency. The major contributors to large early releases remain the same as indicated in the pre-power uprate analysis and include steam generator tube ruptures and interfacing system loss of coolant accidents.

#### Quality of the Model

Regarding the quality of the model, the ANO-2 Probabilistic Safety Analysis Level-i Model was initially developed in response to NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities." This model was developed by safety analysis design engineering personnel with support from other design engineering groups and from operations. A peer review of the IPE results was performed. Since its submittal to the NRC staff, this model has been updated several times to maintain it consistent with the as built plant. Design engineering calculations and reports document the development of all major elements of the initial and updated versions of the model. These calculations have been independently reviewed and are retained as quality records for the life of the plant. The PSA evaluation performed in support of the ANO-2 power uprate has similar quality U. S. NRC July 24, 2001 2CAN070105 Page 3

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attributes. Engineering personnel who performed the original GL 88-20 Independent Plant Examination submittal and others who are familiar with the model performed this evaluation. The calculations and reports which document this model were independently reviewed and are retained as quality records for the life of the plant.

This submittal contains no regulatory commitments. Should you have questions, please contact me.

I declare under penalty of perjury that the foregoing is true and correct.

Very truly yours,

Jimmy D. Vandergrift Director, Nuclear Safety Assurance

JDV/dwb Attachment

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Attachment to 2CAN070105 Page 1 of 33

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# Probabilistic Safety Assessment for Power Uprate - Part 2

### **TABLE** OF **CONTENTS**



Attachment to 2CAN070105 Page 2 of 33

#### **INTRODUCTION**

An application for a license amendment to increase the authorized power level was submitted to the NRC staff on December 19, 2000 (2CAN120001). Subsequently, an evaluation was the NRC staff on December 19, 2000 (2CAN120001). performed to assess the potential effects on plant risk of an ANO-2 power uprate to 3026 MWt, which represents a 7.5 % increase in reactor thermal power above the currently licensed core power rating of 2815 MWt.

This report addresses the following risk areas: 1) Level 1 Internal Events – Success Criteria and Onerator Actions. 2) Level 2 Internal Events, and 3) Ouality of the Model. Section 4) Operator Actions, 2) Level 2 Internal Events, and 3) Quality of the Model. summarizes the conclusions of the discussion.

The remaining risk areas assessed are summarized in a submittal to the staff dated June 28, 2001 **(2CAN0601** 10): 1) Level 1 Internal Events - Initiating Event Frequencies, 2) Level 1 Internal Events - Component Failure Rates, 3) IPEEE Internal Fire Analysis, 4) IPEEE Seismic Analysis, 5) IPEEE Other External Events Analysis, and 6) Shutdown Risk.

#### **1** LEVEL 1 INTERNAL EVENTS - **SUCCESS** CRITERIA **&** OPERATOR **ACTIONS**

The Level-i internal events risk impact assessment of the 7.5% Power Uprate was performed via the revision and quantification of the existing ANO-2 PSA Master Model to account for power uprate related plant hardware modifications and to account for the effect of power uprate on the plant success criteria and operator failure probabilities.

#### 1.1 Plant Modifications

Plant hardware modifications related to power uprate were identified and incorporated, as appropriate, into the existing ANO-2 PSA Level-i Internal Events Model. The only modification related to power uprate identified to affect the ANO-2 PSA Level-1 Internal Events was the installation of a signal actuated by high containment pressure to close Main Feedwater Isolation Isolation/Block Valves, trip the Main Feedwater Pumps, close the Feedwater Regulating Valves and Main Feedwater Regulating Bypass Valves, trip the Condensate Pumps on either a Main Steam Isolation Signal (MSIS) or Containment Spray Actuation Signal (CSAS), and trip the Feedwater Heater Drain Pumps on either MSIS or CSAS signals. In addition, this modification also removed the MSIS close signal from the Main Feedwater Regulating Bypass Valves.

#### 1.2 Success Criteria

A detailed review was performed to identify the effect of the 7.5% increased thermal power level on the system success criteria credited in the ANO-2 Probabilistic Safety Analysis (PSA) Level-1

Attachment to 2CAN070105 Page 3 of 33

Internal Events model. These success criteria specify the requirements of the plant systems to address critical safety functions. These safety functions are as follows:

- $\bullet$  Reactivity Control (K)
- RCS Pressure Control/Pressure Boundary Integrity (Q)
- RCS and Core Heat Removal (B)
- $\bullet$  Once-Through-Cooling (F)
- " RCS Inventory Control (U)
- $\bullet$  Long term RCS Inventory Control and Heat Removal  $(X)$

With one exception, the power uprate was determined to have no impact on the success criteria associated with each of these safety functions. The one exception is that the High Pressure Safety Injection (HPSI) success criteria for long term reactor coolant system (RCS) heat removal following a Large Break Loss of Coolant Accident (LBLOCA) was increased from one HPSI pump injecting into the RCS through two of four open injection flow lines (one of which may contain a pipe break) for the current power level to one HPSI pump injecting into the RCS through three of four open injection flow lines (one of which may contain a pipe break) for the uprated power level. The difference is due to the slightly larger decay heat levels at the time of recirculation.

#### 1.3 Operator Actions

A detailed assessment was performed to account for the effect of the 7.5% increase in thermal power level on the probability of operator failure. The analysis involved the development of realistic available times for post-initiator operator actions using the ANO-2 CENTS model to simulate a variety of accident conditions. The CENTS calculations were performed for both the current and uprated power levels. These available times were then input into the ANO-2 PSA Human Reliability Analysis (HRA) model to assess the effect of the power uprate on the operator actions used in the model. A comprehensive list of the post-initiator operator actions whose probability changed as a result of the power uprate is provided in Table 1, below. The table provides the operator failure event or Human Failure Event (HFE), its description, the available time and the mean probability of the operator failure for both pre-power uprate and post-power uprate conditions.

		<b>Pre-Power Uprate</b>			<b>Post-Power Uprate</b>		
<b>Event Name</b>	<b>Description</b>		Available   <b>Time</b>	<b>Mean</b> <b>Probability</b>	Available Time		Mean Probability
EHF2A1A2SP	Failure to Re-energize 2A1/2A2 from ST2 (SBLOCA or SGTR)	42	min	$1.9E-1$	39	min	$2.9E-1$
	EHF2A1A2TP Failure to Re-energize 2A1/2A2 from ST2 (Transient)	80	min	$1.6E-2$	68	min	$2.0E-2$
	EHF2A3A4XP Failure to reduce loads and cross-tie $ 4160v$ buses 2A3 and 2A4	80	min	$1.1E-2$	68	min	$2.8F - 2$

Table 1: Post-Initiator Operator Failure Events Affected by Power Uprate

#### Attachment to 2CAN070105 Page 4 of 33



It should be noted that other operator failures occur in the model and model cutset results. These operator failures include pre-initiator operator failure events (e.g., "failure to restore after maintenance") and post-initiator operator failures which did not change as a result of the power uprate.

Power uprate impacted the operator failures in that it had the general effect of reducing the time available for the operator to complete the recovery action. This change is due to the higher decay heat level after power uprate. Table 1 provides an estimate for the time available to complete

Attachment to 2CAN070105 Page 5 of 33

each operator action both prior to and after power uprate. These estimates were based on CENTS calculations performed at both pre- and post-power uprate conditions. CENTS is a best estimate deterministic thermal-hydraulic analysis code which accounts for both the primary and secondary system response to plant transient events, including small break LOCAs. The ANO-2 CENTS model was exercised to estimate the latest time for successful operator action to avert two-phase core uncovery for a representative set of accident scenarios which the PSA model identified as important to risk. These CENTS-generated core uncovery times were used as the pre- and post-power uprate available time for operator actions which were tied to core damage. Since core uncovery preceeds core damage, use of core uncovery time as the basis for available time for operator action yields conservatively high operator failure probabilities for operator actions tied to averting core damage. Thus, the operator failure probabilities applied in the assessment of the risk impact of the ANO-2 power uprate are conservatively high.

#### 1.4 Core Damage Results

The Level-I internal events risk impact assessment of the 7.5% power uprate was performed via the revision and quantification of the existing ANO-2 PSA Master Model to account for the above-noted changes in the plant hardware, its success criteria, and its operator failure probabilities. Two ANO-2 PSA models were used in the assessment of the risk impact of the power uprate:



ANO-2 PSA Model, Rev-2B - post-power uprate model using post-power uprate success<br>criteria and post-power uprate operator failure criteria and post-power probabilities

Both models were quantified and their results compared in order to assess the affect of the power uprate on the internal events Level-1 risk. As a result of this assessment, the effect of the 7.5% Power Uprate from 2815 MWt to 3026 MWt was estimated to increase the internal events Core Damage Frequency (CDF) from 1.70E-05/rx-yr to 1.97E-05/rx-yr (an increase of about 16%). Table 2 provides the effect of the power uprate on the CDF results by accident sequence. Tables 3 and 4 provide listings of the dominant (top 50) Rev-2A and Rev-2B core damage cutset contributors, respectively. Tables 5 and 6 provide listings of the dominant (top 15) Rev-2A and Rev-2B dominant Human Failure Events (HFEs) based on Fussell-Vesely Importance to core damage, respectively. As noted above, these results represent conservatively high estimates of the CDF after power uprate due to the conservative treatment of operator action.

Attachment to 2CAN070105 Page 6 of 33

It should be noted that the cutsets in Tables 3 and 4 contain accident sequence flags (e.g., event  $TBF<sup>1</sup>$ . These flags link the cutset its associated accident sequence; since the flag has a value of one, it does not contribute to the core damage frequency. In addition, it should be noted that as part of the quantification process, whenever more than one operator failure event occurs in a single cutset, a single combined operator failure event is applied to the cutset to effectively replace the contributing operator failure events. The value of the combined operator failure event accounts for the dependence between the operator failure events and the values of the contributing operator failure events in the cutset are set to one. In this manner, the probability of cutsets remain conservative.

<b>Sequence</b>	<b>Rev-2A CDF</b>	<b>Rev-2B CDF</b>	<b>Change in CDF</b>	% Change in
	(Pre-Power	(Post-Power	(fr Rev-2A to Rev-	<b>Total CDF</b>
	<b>Uprate</b> )	<b>Uprate</b> )	2B)	
AU	6.63E-08	6.63E-08	$0.00E + 00$	$0.0\%$
AX	7.15E-08	7.15E-08	$0.00E + 00$	$0.0\%$
MU	1.93E-08	1.93E-08	$0.00E + 00$	$0.0\%$
<b>MX</b>	7.15E-08	7.15E-08	$0.00E + 00$	0.0%
<b>RBF</b>	1.44E-07	3.03E-07	1.59E-07	0.9%
<b>RBU</b>	5.00E-10	5.00E-10	$0.00E + 00$	$0.0\%$
<b>RBX</b>	3.29E-09	3.29E-09	$0.00E + 00$	$0.0\%$
<b>RU</b>	2.15E-09	2.15E-09	$0.00E + 00$	$0.0\%$
<b>RX</b>	1.64E-08	1.71E-08	7.00E-10	$0.0\%$
<b>SBF</b>	9.92E-10	1.29E-07	1.28E-07	0.8%
<b>SBU</b>	$0.00E + 00$	1.91E-10	1.91E-10	$0.0\%$
<b>SBX</b>	4.88E-14	1.92E-10	1.92E-10	$0.0\%$
<b>SU</b>	3.14E-07	3.14E-07	$0.00E + 00$	$0.0\%$
SX	1.01E-07	1.01E-07	$0.00E + 00$	$0.0\%$
<b>TBF</b>	1.23E-05	1.39E-05	1.60E-06	9.4%
TBX	1.30E-06	1.33E-06	3.00E-08	0.2%
<b>TQBF</b>	4.00E-07	1.17E-06	7.70E-07	4.5%
<b>TQBU</b>	1.82E-08	1.83E-08	1.00E-10	0.0%
<b>TQBX</b>	2.30E-08	2.37E-08	7.00E-10	0.0%
<b>TQU</b>	9.77E-07	9.78E-07	1.00E-09	0.0%
<b>TQX</b>	1.20E-06	1.20E-06	$0.00E + 00$	0.0%
Total	1.70E-05	1.97E-05	2.69E-06	15.8%

Table 2. Change in ANO-2 CDF due to Power Uprate by Accident Sequence

 $<sup>1</sup>$  The TBF sequence involves transient initiating events with a subsequent loss of reactor coolant system and core</sup> heat removal (i.e., main feedwater and emergency feedwater failures) and failure of once-through cooling (due to high pressure safety injection or emergency core cooling system and low temperature overpressure protection vent valve failures). This sequence leads to high reactor coolant system pressure early core melt.

#### Attachment to **2CAN070105**  Page 7 of 33

#### Table 3. Dominant ANO-2 Rev-2A Core Damage Cutset Contributors



### Attachment to **2CAN070105**  Page **8** of **33**



### Attachment to 2CAN070105 Page 9 of 33



#### Attachment to 2CAN070105 Page 10 of 33



#### Attachment to 2CAN070105 Page 11 of 33



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#### Attachment to 2CAN070105 Page 12 of 33



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### Attachment to 2CAN070105 Page 13 of 33



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#### Attachment to 2CAN070105 Page 14 of 33



### Attachment to 2CAN070105 Page 15 of 33



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### Attachment to 2CAN070105 Page 16 of 33



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### Attachment to 2CAN070105 Page 17 of 33



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### Attachment to 2CAN070105 Page 18 of 33

## Table 4. Dominant ANO-2 Rev-2B Core Damage Cutset Contributors



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### Attachment to 2CAN070105 Page 19 of 33



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### Attachment to 2CAN070105 Page 20 of 33



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#### Attachment to 2CAN070105 Page 21 of 33



### Attachment to 2CAN070105 Page 22 of 33



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### Attachment to 2CAN070105 Page 24 of 33



#### Attachment to 2CAN070105 Page 25 of 33



### Attachment to 2CAN070105 Page 26 of 33



### Attachment to 2CAN070105 Page 27 of 33



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#### Attachment to 2CAN070105 Page 28 of 33



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#### Attachment to 2CAN070105 Page 29 of 33



Table 5. Dominant Human Failure Events for ANO-2 Rev-2A Based on Fussell-Vesely Importance to Core Damage

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### Attachment to 2CAN070105 Page 30 of 33



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#### 2 LEVEL 2 **INTERNAL EVENTS**

Large-Early Release Fraction (LERF) as defined by the Electric Power Research Institute PSA Applications Guide (TR-105396) is:

- \* Unscrubbed Containment Bypass Pathway Occurring With Core Damage; or
- Unscrubbed Containment Failure Pathway of Sufficient Size to Release the Contents of the Containment (i.e., one volume change) Within One Hour, Which Occurs Before or Within 4 Hours of Vessel Breach.

The majority of Level-2 sequences do not fall in the large early release category either because they do not result in containment failure, they are vented or scrubbed, or they occur late, many hours into the accident. Those sequences that do fall in this category are called LERF sequences.

The Level 1 Internal Events Risk Impact Assessment of ANO-2 Power Uprate develops two Level 1 fault tree models. The Rev. 2A model represents the plant before the power uprate while the Rev. 2B model represents the plant after the power uprate. The LERF analysis calculates the LERF associated with each of the Level 1 PSA Models and compares them to determine the LERF impact of the power uprate.

From the pre-power uprate LERF results, the potential for a large early release was estimated to be 3.872E-07/rx-yr. This value is small, on the order of 2% of the total core damage frequency. As with other large, dry containment plants, large early releases for ANO-2 are dominated by bypass events (i.e., steam generator tube ruptures and interfacing system LOCAs) along with transients followed by failure of primary to secondary cooling and once through cooling. Table 7 presents the distribution of LERF sequences for both pre- and post-power uprate.

Following power uprate, the potential for a large early release was estimated to be 4.801E-07/rx-yr. The increase in LERF is roughly proportional to the increase in the core damage frequency, remaining at approximately 2% of the new core damage frequency. Virtually all of the changes in the LERF quantification are a result of the changes made to the Level 1 accident sequence analysis in the form of reduced time available for operator action in preventing core damage. The major contributors to large early releases remain the same as indicated in the pre-power uprate analysis and include steam generator tube ruptures and interfacing system LOCAs along with transients followed by failure of primary to secondary cooling and once through cooling.

In conclusion, the 7.5% Power Uprate increases LERF roughly proportionally to the increase in the core damage frequency. The LERF value remains at approximately 2% of the new core damage frequency. The major contributors to large early releases remain the same as indicated in the pre-power uprate analysis and include steam generator tube ruptures and interfacing system LOCAs along with transients followed by failure of primary to secondary cooling and once through cooling.

Attachment to 2CAN070105 Page 32 of 33

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Table 7. Change in ANO-2 LERF due to Power Uprate by Accident Sequence

Attachment to 2CAN070105 Page 33 of 33

#### **3** QUALITY OF THE MODEL

The ANO-2 Probabilistic Safety Analysis Level-1 Model (ANO-2 PSA Level-i Model) was initially developed in response to NRC Generic Letter 88-20. This model was developed by safety analysis design engineering personnel with support from other design engineering groups and from operations. A peer review of the IPE results was performed. Since its submittal to the Staff, this model has been updated several times to maintain it consistent with the as-built plant. Design Engineering calculations and reports document the development of all major elements of the initial and updated versions of the model. These calculations have been independently reviewed and are retained as quality records for the life of the plant.

The PSA evaluation performed in support of the ANO-2 power uprate has similar quality attributes. Engineering personnel who performed the original IPE submittal and others who are familiar with the model performed this evaluation. And, the calculations and reports which document this model were independently reviewed and are retained as quality records for the life of the plant.

#### 4 **CONCLUSION**

As a result of this assessment, the effect of the 7.5% Power Uprate from 2815 MWt to 3026 MWt was estimated to increase the internal events core damage frequency from 1.70E-05/reactor-year to 1.97E-05/reactor-year (an increase of about 16%).

The 7.5% Power Uprate increases the large early release fraction roughly proportionally to the increase in the core damage frequency. The large early release fraction value remains at approximately 2% of the new core damage frequency. The major contributors to large early releases remain the same as indicated in the pre-power uprate analysis and include steam generator tube ruptures and interfacing system loss of coolant accidents.