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SEP 11 1975

Docket No. 50-313

Arkansas Power and Light Company
ATTN: Mr. J. D. Phillips
Senior Vice President
Production, Transmission and Engineering
Sixth and Pine Streets
Pine Bluff, Arkansas 71601

Gentlemen:

The Commission has issued the enclosed Amendment No. 4 to Facility Licnese No. DPR-51 for Arkansas Nuclear One - Unit 1. This amendment includes Change No. 4 to the Technical Specifications and is in response to your request dated April 17, 1975.

This amendment incorporates: (1) provisions for the operability of the Steam Line Break Instrumentation and Control System (SLBIC) at a minimum acceptable main steam pressure and for an actuation setpoint of the SLBIC, (2) replacement of the main steam line instrument channel operation requirements with the SLBIC control and logic channel operation requirements, and (3) addition of surveillance requirements for the SLBIC and its safety related valves.

Copies of our Safety Evaluation and the Federal Register Notice relating to this action are enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 4 w/Change No. 4
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

OFFICE➤	RL:ORB #2	RL:ORB	OELD	RL:ORB #2		
SURNAME➤	FDAnderson:ah	RKINgram		DLZiemann		
DATE➤	7/25/75	7/25/75	9/10/75	9/11/75		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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Honorable Wayne Nordin
Acting County Judge of Pope County
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Health Services
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Mr. Clinton Spotts
U. S. Environmental Protection Agency
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1600 Patterson Street
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ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated April 17, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.c(2) of Facility License No. DPR-51 is hereby amended to read as follows:

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"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 4."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION
Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Attachment:
Change No. 4 to the
Technical Specifications

Date of Issuance:

SEP 11 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 4 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Delete pages 39a, 40, 41, 42, 43a, 44, 45b, 46, 72 and 73a from the Appendix A Technical Specifications and insert the attached replacement pages. The changed areas on the revised pages are shown by a marginal line.

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The penetration room ventilation system consists of two independent, full capacity, 100% redundant trains.[5] If one train is removed from operation, the other train must be operable.

REFERENCES

(1) FSAR, Section 14.2.5

(2) FSAR, Section 3.2

(3) FSAR, Section 9.5.2

(4) FSAR, Section 9.3.1

(5) FSAR, Section 6.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

3.4.1 The reactor shall not be heated, above $280^{\circ}F$ unless the following conditions are met:

1. Capability to remove a decay heat load of 5% full reactor power by at least one of the following means:

   a. A condensate pump and a main feedwater pump, using turbine by-pass valve.

   b. A condensate pump and the auxiliary feedwater pump using turbine by-pass valve.

2. Fourteen of the steam system safety valves are operable.

3. A minimum of 16.3 ft. (107,000 gallons) of water is available in the condensate storage tank.

4. Both emergency feedwater pumps are operable.

5. Both main steam block valves and both main feedwater isolation valves are operable.

6. The emergency feedwater valves associated with Specification 3.4.1.4 shall be operable.

3.4.2 The Steam Line Break Instrumentation and Control System (SLBIC) shall be operable when main steam pressure exceeds 700 psig and 4
shall be set to actuate at 600 ±25 psig.

3.4.3 Components required by Specification 3.4.1 and 3.4.2 to be operable shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification
3.4.1 and 3.4.2 within 24 hours the reactor shall be placed in the hot 4
shutdown condition within 12 hours. If the requirements of Specifi-
3.4.1 and 3.4.2 are not met within an additional 48 hours, the reactor 4
shall be placed in the cold shutdown condition within 24 hours.

Bases

The feedwater flow required to remove decay heat corresponding to 5% full power with saturated steam at 1065 psia (lowest setting of steam safety valve) as a function of feedwater temperature is:

Feedwater Temperature	Flow
60	758
90	777
120	799
140	814

The feedwater system and the turbine bypass system are normally used for decay heat removal and cooldown above 280 F. Feedwater makeup is supplied by operation of a condensate pump and either a main or the auxiliary feedwater pump.

In the incredible event of loss of all AC power, feedwater is supplied by the turbine driven emergency feedwater pump which takes suction from the condensate storage tank. Decay heat is removed from a steam generator by steam relief through the atmospheric dump valves or safety valves. Fourteen of the steam system safety valves will relieve the necessary amount of steam for rated reactor power.

The minimum amount of water in the condensate storage tank would be adequate for about 4.5 hours of operation. This is based on the estimate of the average emergency flow to a steam generator being 390 gpm. This operation time with the volume of water specified would not be reached, since the decay heat removal system would be brought into operation within 4 hours or less.

If the turbine driven emergency feedwater pump has not been verified to be operable within 3 months prior to heatup its operability will be verified upon reaching hot shutdown conditions.

The SLBIC System is designed to isolate the steam generators to assure that only one steam generator will experience uncontrolled blowdown following a steam line break. Normal steam line operating pressures are approximately
900 psig at all power levels, thus operability above 700 psig with actuation 4
at 600 ±25 psig are appropriate. The setpoint is based on severe transients in the main steam lines resulting in rapid pressure decays.

References

FSAR, Section 10

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 Startup and operation are not permitted unless the requirements of Table 3.5.1-1, columns 3 and 4 are met.

3.5.1.2 In the event the number of protection channels operable falls below the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel will be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in this untripped state at any one time. Only one channel bypass key shall be accessible for use in the control room.

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation.

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes following detection. The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of both the RPS and the ESAS enable complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state on-line repairs to the failed device, when practical will be made, and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

The Steam Line Break Instrumentation and Control System (SLBIC) is designed to automatically close the Main Steam Block valves and the Main Feedwater Isolation valves upon loss of pressure in either of the two main steam lines. | 4

The SLBIC is also designed to be reset from its trip position only when the system is shut down or the Main Steam line pressure is below 650 psig. | 4

REFERENCE

FSAR, Section 7.1

Table 3.5.1-1 Instrumentation Limiting Conditions for Operation
(Note 6)

REACTOR PROTECTION SYSTEM	1	2	3	4	5
Functional Unit	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action is conditions of column 3 or 4 cannot be met
1. Manual pushbutton	1	1	1	0	Note 1
2. Power range instrument channel	4	2	3 (Note 4)	1 (Note 4)	Note 1
3. Intermediate range instrument channels	2	Note 7	1	0	Notes 1, 2
4. Source range instrument channels	2	Note 7	1	0	Notes 1, 2, 3
5. Reactor coolant temperature instrument channels	4	2	2	1	Note 1
6. Pressure-temperature instrument channels	4	2	2	1	Note 1
7. Flux/imbalance/flow instrument channels	4	2	2	1	Note 1
8. Reactor coolant pressure					
a. High reactor coolant pressure instrument channels	4	2	2	1	Note 1
b. Low reactor coolant pressure instrument channels	4	2	2	1	Note 1
9. Power/number of pumps instrument channels	4	2	2	1	Note 1
10. High reactor building pressure channels	4	2	2	1	Note 1

Table 3.5.1-1 (Contd)

OTHER SAFETY RELATED SYSTEMS Functional unit	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
2. Steam line break instrumentation control system (SLBIC)					
a. SLBIC Control & Logic channels	2	1	2	1	Notes 9, 5 4

Notes:

1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition if the requirements of Columns 3 and 4 are not met within 12 hours.

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater that 10^{-10} amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of 4 hours, after which Note 1 applies.

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, Specification 3.3 shall apply.

7. These channels initiate control rod withdrawal inhibits not reactor trips at <10% rated power. Above 10% rated power these inhibits are bypassed.

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies. 4

3.5.2 Control Rod Group and Power Distribution Limits

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall be not less than 1% Δk/k with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

1. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.

2. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existance of 1% Δk/k available shutdown margin. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

3. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a 1% Δk/k available shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.

4. Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

5. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
30. Decay Heat Removal System Isolation Valve Automatic Closure And Interlock System	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) Shall Also Be Tested During Refueling Shutdown Prior to Re-pressurization at a pressure greater than 300 but less than 420 psig.
31. Turbine Overspeed Trip Mechanism	NA	R	NA	
32. Steam Line Break Instrumentation And Control System Logic Test & Control Circuits	W	Q	R	4
33. Diesel Generator Protective Relaying, Starting Interlocks And Circuitry	M	Q	NA	
34. Off-site Power Undervoltage And Protective Relaying Interlocks And Circuitry	W	R	R	
35. Borated Water Storage Tank Level Indicator	W	NA	R	
36. Boric Acid Mix Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	

Table 4.1-2 (Continued)
Minimum Equipment Test Frequency

Item	Test	Frequency
12. Flow Limiting Annulus on Main Feedwater Lines at Reactor Building Penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.
13. SLBIC Pressure Sensors	Calibrate	Each Refueling Period
14. Main Steam Isolation Valves	a. Excercise Through Approximately 10% Travel	a. Quarterly
	b. Cycle	b. Each Refueling Shut-down.
15. Main Feedwater Isolation Valves	a. Exercise Through Approximately 5% Travel	a. Quarterly
	b. Cycle	b. Each Refueling Shut-down.

4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 4 TO LICENSE NO. DPR-51

CHANGE NO. 4 TO TECHNICAL SPECIFICATIONS

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

INTRODUCTION

By letter dated April 17, 1975, the Arkansas Power & Light Company (the licensee) requested changes to the Technical Specifications appended to Facility Operating License No. DPR-51 for Arkansas Nuclear One - Unit 1. The changes involve setting and surveillance requirements for the Steam Line Break Instrumentation and Control System (SLBIC).

DISCUSSION

During the operating license review for the Arkansas Nuclear One - Unit 1 facility, it was recognized that a very large steamline break could cause an unacceptably high reactivity increase in the core if that break occurred late in core life when the moderator temperature coefficient has a substantial negative value. The SLBIC system was designed to limit the positive reactivity addition and attendant reactor power increase to acceptable levels in the event of any steam line break at any time in core life.

The plant was licensed while the final design and Technical Specifications for the SLBIC were under review since analysis submitted by the licensee showed that the moderator temperature coefficient does not reach sufficiently negative values to require the protection of the SLBIC until the core has accumulated 225 EFPD (effective full power days) of exposure. The licensee agreed to have the SLBIC system operable at this point in core life. The SLBIC system is a seismic Cateogry I system which consists of four steam pressure sensors and one logic cabinet per steam loop plus valve operators, power supplies, and associated wiring. It provides for prompt detection of the rapid decrease in steam pressure that would accompany a large steam line break and prompt isolation of the steam generator in the affected loop by closing that loop's main steam block valve and main feedwater block valve. This isolation eliminates the excessive removal of heat by that steam generator thereby limiting the decrease in water temperature in the affected primary coolant loop. This, in turn, limits the positive reactivity added to the core (due to the negative temperature coefficient) and the attendant power increase is reduced to acceptable levels. We have reviewed the design of the SLBIC and by letter of February 12, 1975, notified the licensee of its acceptability.

The Technical Specification changes involved in this amendment provide for the operation of and establish the safety settings for the SLBIC system. The changes proposed by the licensee include a change to Specification 3.4 to require that the SLBIC be operable before the plant's main steam pressure exceeds 700 psig. This specification also calls for the SLBIC to be set at 600 + 25 psig. The licensee also proposed to change Table 3.5.1-1 of the Technical Specifications to add the SLBIC to the instrumentation required for operation. Lastly, the licensee proposes changes in the surveillance section to require weekly check and quarterly test of the SLBIC as well as calibration at each refueling period.

EVALUATION

We have evaluated these proposed changes to the Technical Specifications and find them acceptable for the following reasons:

a. The steam pressure limit at which the SLBIC is required to be operable, 700 psig, is well below the normal steam pressure of 900-1000 psig assuring that the SLBIC will be available when a very large steam break excursion might be experienced. Any system blowdown occurring at initial pressure less than 700 psig will not yield consequences greater than those which have been found acceptable in the staff's safety evaluation of the Arkansas Nuclear One - Unit 1 Final Safety Analysis Report.

b. The SLBIC actuation pressure of 600 + 25 psig is far enough below the required operability pressure (700 psig) to provide a practical interval for arming the system. The setpoint provides a reasonable margin between the lowest pressure expected during an operating transient, 700 psig, and is high enough to provide prompt system actuation should a ~~main~~ major steam line break occur. The consequences of any system blowdown which commences at an initial steam pressure equal to or greater than the operability pressure and which is terminated by the SLBIC system actuated at a pressure of 600 + 25 psig is less than those consequences which have been found acceptable in the staff's safety evaluation of the Arkansas Nuclear One - Unit 1 Final Safety Analysis Report.

c. The surveillance requirements proposed by the licensee for the SLBIC are consistent with the surveillance requirements usually applied to systems of this type and are sufficient to cover all elements of the system. Item 14.a of Table 4.1-2 (Minimum Equipment Test Frequency)

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concerning exercising of the main steam isolation valves has been modified to allow exercising the valves through 10% of travel vice 20% of travel proposed by the licensee. Twenty percent of travel could interfere with steam flow at high power operation whereas 10% of travel sufficiently verifies valve operability, yet has a negligible effect on power operation. The licensee agrees with this change.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or conseqeunces of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: SEP 11 1975

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SURNAME ➤						
DATE ➤						

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-313

ARKANSAS POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 4 to Facility Operating License No. DPR-51 issued to Arkansas Power and Light Company which revised Technical Specifications for operation of the Arkansas Nuclear One - Unit 1, located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment: (1) incorporates provisions for the operability of the Steam Line Break Instrumentation and Control System (SLBIC) at a minimum acceptable main steam pressure and for an actuation setpoint of the SLBIC, (2) replaces the main steam line instrument channel operation requirements with the SLBIC control and logic channel operation requirements, and (3) adds surveillance requirements for the SLBIC and its safety related valves.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

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For further details with respect to this action, see (1) the application for amendment dated April 17, 1975, (2) Amendment No. 4 to License No. DPR-51, with Change No. 4, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Arkansas Polytechnic College, Russellville, Arkansas. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 11th day of September, 1975

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

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SURNAME ➤						
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