DCR 016

November 14, 1986

Docket No. 50-313

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Mr. Gene Campbell
Vice President, Nuclear
Operations
Arkansas Power and Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Campbell:

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your applications dated July 18, 1986, and July 31, 1986. Clarifying information concerning the July 31, 1986 application, was provided by letter dated October 17, 1986.

The amendment (1) increases the setpoint for reactor trip on high pressure from 2300 psig to 2355 psig, and (2) increases the anticipatory reactor trip on turbine trip arming threshold from 20% of full power to 45% of full power and (3) corrects several editorial errors.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Guy S. Vissing, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 104 to DPR-51
2. Safety Evaluation

cc w/enclosures:
See next page

PBD-6	PBD-6	PBD-6	PBD-6	PBD-6	PBD-6	OGC
RIngram	GVissing:eh	JRamsey	CMcCracken	GEdison	JFStolz	No Legal Objection
/ /86	11/7/86	11/7/86	11/17/86	11/7/86	11/7/86	11/10/86

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P PDR

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Arkansas Power & Light Company

Arkansas Nuclear One, Unit 1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-51

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

   A. The applications for amendments by Arkansas Power and Light Company (the licensee) dated July 18, 1986 and July 31, 1986, comply 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 applications, 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;

   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; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 104, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 14, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
10	10
13	13
14a	14a
15	15
42a	42a
43a	43a
45f	45f

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.(2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established.(2) The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%) (3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.(4)

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.[2]

The low pressure (1800 psig) and variable low pressure ($11.75T_{out}$ -5103) trip setpoints shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction.[2,3]

Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of ($11.75t_{out}$ -5143).

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620F.

E. Reactor Building Pressure

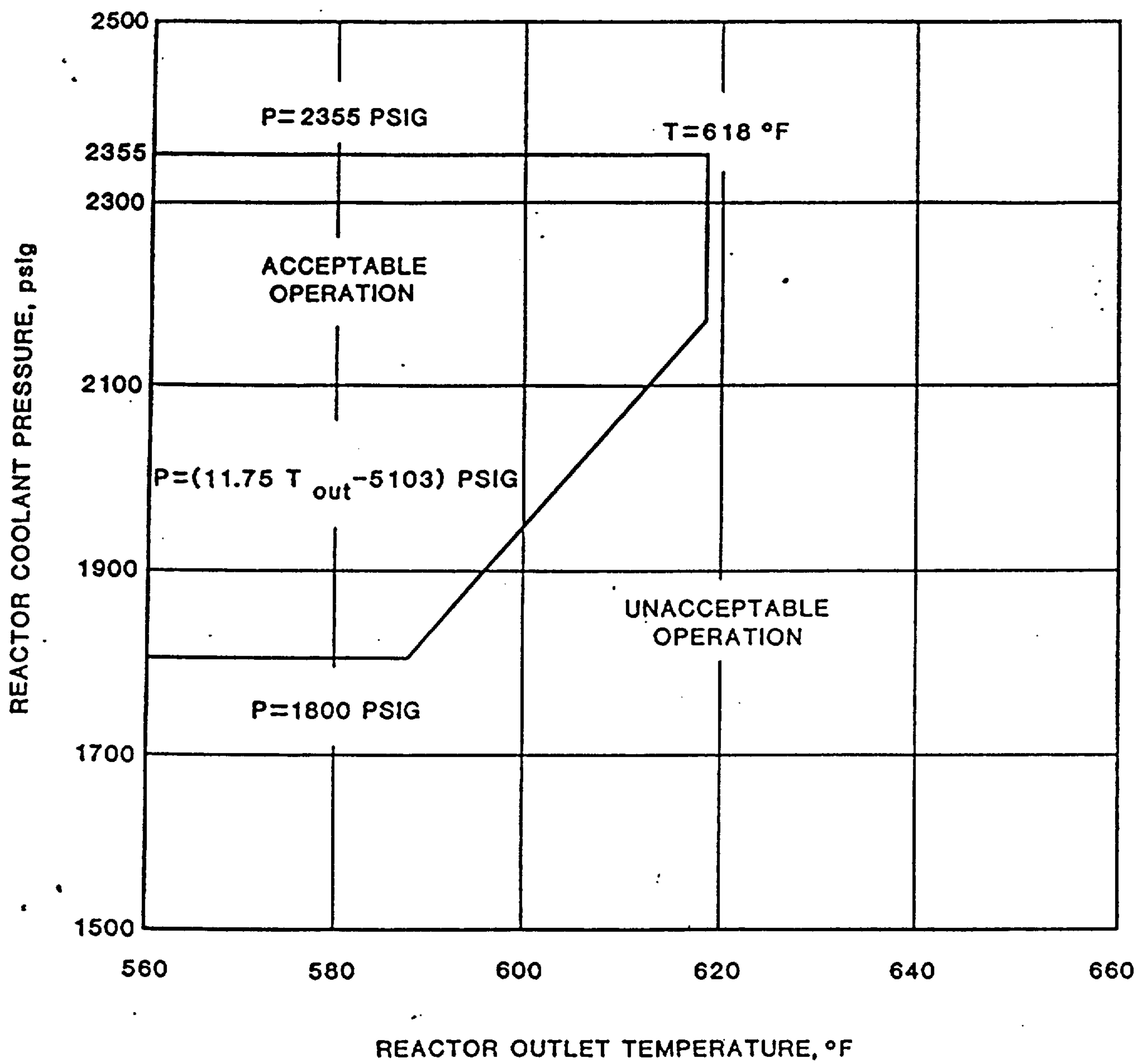
The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip setpoint of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINT

FIGURE 2.3-1

Table 2.3-1
Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[(a)]
Nuclear Power based on flow[b] and imbalance, % of rated, max	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	Bypassed
Nuclear Power based on pump monitors[c], % of rated, max	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig, min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	$11.75\ T_{out}-5103^{d}$	$11.75\ T_{out}-5103^{d}$	$11.75\ T_{out}-5103^{d}$	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

[a]Automatically set when other segments of the RPS (as specified) are bypassed.

[b]Reactor coolant system flow.

[c]The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

[d]T_{out} is given in degrees Fahrenheit (F).

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3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig.

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows:

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be >3115 VAC but <3177 VAC.

b. The 460 V emergency bus undervoltage relay setpoints shall be > 423 VAC but <431 VAC with a time delay setpoint of 8 seconds ±1 second.

3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated:

1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a and item 35 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)

2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% reactor power.)

3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.

3.5.1.10 The control room ventilation chlorine detection system instrumentation shall be operable and capable of actuating control room isolation and filtration systems, with alarm/trip setpoints adjusted to actuate at a chlorine concentration of ≤5ppm.

3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed.

3.5.1.12 The Containment High Range Radiation Monitoring instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr.

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 the RPS, the ESAS, and the EFIC enables 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 EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. The EFIC can be tested from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activation of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the component. The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

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.

TABLE 3.5.1-1 (Cont'd)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

15. This trip function may be bypassed at up to 10% reactor power.

16. This trip function may be bypassed at up to 45% reactor power.

17. With no channel operable, within 1 hour restore the inoperable channels to operable status, or initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.

20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, <u>or</u> (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.4 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable staus within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in hot shutdown within 12 hours.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

INTRODUCTION

By letters dated July 18 and 31, 1986, the Arkansas Power and Light Company (AP&L or the licensee) requested amendments to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). By letter dated October 17, 1986, the licensee provided clarifying information for the submittal dated July 31, 1986. The proposed amendments would (1) increase the setpoint for reactor trip on high pressure from 2300 psig to 2355 psig, (2) increase the anticipatory reactor trip (ART) on turbine trip arming threshold from 20% of full power to 45% of full power, and (3) correct several editorial errors. Specifically, the proposed amendments would change TS Figure 2.3-1, Table 2.3-1, 2.2 Bases, 2.3 Bases C, 3.5.1 Bases, TS 3.5.1.9.1, 3.5.1.9.2 and Table 3.5.1-1.

EVALUATION

ANO-1's TSs currently have a value of 2300 psig for the high pressure trip setpoint, and a value of 20% of full power for the ART arming threshold. These values were based on changes required by the NRC staff (Ref. 1), subsequent to the accident at Three Mile Island Unit 2, to reduce challenges to and opening of the power operated relief valve (PORV).

In April 1986, we completed our review of the following two Babcock & Wilcox (B&W) topical reports related to AP&L's request: (1) "Justification for Raising Setpoint For Reactor Trip on High Pressure," BAW-1890, September 1985 (Ref. 2); and (2) "Basis For Raising Arming Threshold For Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985 (Ref. 3). The results of our review of the topical reports are contained in two Safety Evaluation Reports (Refs. 4 and 5). In those Safety Evaluation Reports, we: (1) reviewed the basis for the proposed changes; (2) reviewed B&W's method of analysis of the effect of the proposed high pressure trip setpoint on PORV openings; (3) compared the results of Monte Carlo simulation for PORV openings with the NRC requirements contained in NUREG-0737 (Ref. 6); and (4) reviewed the results of B&W's analysis of the arming threshold for ART. The NRC requirements include: (1) the PORV will open in less than 5% of all anticipated overpressure transients (Ref. 6, Item II.K.3.7); and (2) the probability of a small-break loss of coolant accident caused by a stuck-open PORV will be less than 0.001 per reactor-year (Ref. 6, Item II.K.3.2). In the Safety Evaluation Reports, we concluded on a generic basis that the proposed changes met the NRC requirements, and should benefit plants by potentially reducing the reactor trip frequency.

In regard to ANO-1, the licensee states in their supplemental response dated October 17, 1986, that the accidents analyzed in Chapter 14 of the Final Safety Analysis Report (FSAR) were based on the high pressure trip setpoint of 2355 psig (not the current setpoint of 2300 psig) and no credit for ART. The increase in the high pressure trip setpoint and the increase in arming threshold for ART will not increase the frequency of challenges to the main steam safety valves as compared to the initial plant design. Consequently, the accidents analyzed in the FSAR for ANO-1 bound the proposed TS changes.

We have reviewed the licensee's submittals and on the basis of the above, have determined that the proposed changes meet the applicable regulatory guidance and requirements and are, therefore, acceptable.

We have also reviewed AP&L's proposed changes to TS 3.5.1.9.1 and 3.5.1.9.2. The purpose of these changes is to correct several editorial errors. The changes are appropriate and acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Dated: November 14, 1986

Principal Contributors:

E. Branagan
G. Vissing

REFERENCES

1. "Nuclear Incident at Three Mile Island - Supplement," IE Bulletin 79-05B, April 21, 1979.

2. "Justification for Raising Setpoint For Reactor Trip on High Pressure," BAW-1890, September 1985.

3. "Basis For Raising Arming Threshold For Anticipatory Reactor Trip on Turbine Trip," BAW-1893, October 1985.

4. Letter from D. M. Crutchfield, NRC to J. H. Taylor, Babcock and Wilcox Company, April 22, 1986.

5. Letter from D. M. Crutchfield, NRC to J. H. Taylor, Babcock and Wilcox Company, April 25, 1986.

6. "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.