PDR

July 22, 1986

Docket No.: 50-368

Mr. T. Gene Campbell
Vice President
  Nuclear Operations
Arkansas Power & Light Company
Post Office Box 551
Little Rock, Arkansas 72203

Dear Mr. Campbell:

Subject: Issuance of Amendment No. 77 to Facility Operating License NPF-6 - Arkansas Nuclear One, Unit No. 2

The Commission has issued the enclosed Amendment No. 77 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated September 16, 1985.

The amendment revises the Technical Specifications pertaining to the Core Protection Calculator (CPC) addressable constants and the reactor protection system surveillance requirements. During our review of the proposed changes, we found that certain modifications to your proposed Technical Specifications were necessary to meet our requirements. The changes were discussed with your staff.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/s/

Robert S. Lee, Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 77 to NPF-6
2. Safety Evaluation

cc: See next page

No legal objection
McGurren
concurrence as per changes to SE.

PBD7	PBD7	OELD	DIR/PBD7
RSLee/yt	JLee		GWKnighton
5/28/86	5/29/86	5/ /86	7/21/86

Mr. T. Gene Campbell
Arkansas Power & Light Company

Arkansas Nuclear One
Unit No. 2

cc:
Mr. J. Ted Enos, Manager, Nuclear
Engineering and Licensing
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Mr. James M. Levine, Director,
Site Nuclear Operations
Arkansas Nuclear One
P. O. Box 608
Russellville, Arkansas 72801

Nicholas S. Reynolds, Esq.
Bishop, Liberman, Cook,
Purcell & Reynolds
1200 Seventeenth Street, N.W.
Suite 700
Washington, D.C. 20036

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Office of Executive Director for
Operations
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 2090
Russellville, Arkansas 72801

Mr. Frank Wilson, Director
Division of Environmental Health
Protection
Arkansas Department of Health
4815 West Markam Street
Little Rock, Arkansas 72201

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220
7910 Woodmont Avenue
Bethesda, Maryland 20814

Mr. Charlie B. Brinkman, Manager
Washington Nuclear Operations
C-E Power Systems
7910 Woodmont Avenue
Suite 1310
Bethesda, Maryland 20814

The Honorable Ermil Grant
Acting County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. NPF-6

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

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated September 16, 1985, 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;

   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. NPF-6 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 77, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Director
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 22, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
III	III
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
B 2-7	B 2-7
3/4 3-7	3/4 3-7
3/4 3-9	3/4 3-9
6-13	6-13

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

THIS PAGE LEFT BLANK INTENTIONALLY

BASES

a.	RCS Cold Leg Temperature-Low	$\geq$ 465°F
b.	RCS Cold Leg Temperature-High	$\leq$ 605°F
c.	Axial Shape Index-Positive	Not more positive than +0.6
d.	Axial Shape Index-Negative	Not more negative than -0.6
e.	Pressurizer Pressure-Low	$\geq$ 1750 psia
f.	Pressurizer Pressure-High	$\leq$ 2400 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

TABLE 4.3-1

REACTOR PROTECTION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Linear Power Level - High	S	D(2,4), M(3,4), Q(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U (1)	1, 2, 3, 4, 5 and *
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2 and *
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2 and *
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6),	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2 and *
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *
14. Core Protection Calculators	S, W(9)	D(2,4) R(4,5)	M, R(6),	1, 2
15. CEA Calculators	S	R	M, R(6),	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is > 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power sub-channel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The correct values of addressable constants (See Table 2.2-2) shall be verified to be installed in each OPERABLE CPC.

THIS PAGE LEFT BLANK INTENTIONALLY

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President, Nuclear Operations and to the SRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Vice-President, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (PCP) Addressable Constants. These procedures should include provisions to assure that sufficient margin is maintained in CPC Type I addressable constants to avoid excessive operator interaction with the CPCs during reactor operation.

NOTE: Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P that has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

h. New and spent fuel storage.

i. ODCM and PCP implementation.

j. Postaccident sampling (includes sampling of reactor coolant, radioactive iodines and particulates in plant gaseous effluent, and the containment atmosphere).

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the ANO General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PSC and approved by the ANO General Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Administrator of the Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated September 16, 1985, Arkansas Power and Light Company (AP&L) submitted a Technical Specification change request for Arkansas Nuclear One, Unit 2 (ANO-2). The proposed change would remove the Core Protection Calculator (CPC) Type I and Type II Addressable Constants listings from the Technical Specifications. The addressable constants of the Combustion Engineering (CE) design CPCs provide a mechanism to incorporate reload dependent parameters and calibration constants to the CPC software so that the CPC Core model is maintained current with changing core configurations and operating characteristics. In addition, Notation (10) of Table 4.3-1 would be deleted based on the fact that the amendment which added the notation specified that the notation would be applicable only for the month of January, 1983.

There are two types of addressable constants. Type I addressable constants are the calibration constants, the sensor operability status flag and the pretrip alarm set points which are expected to change frequently during cycle operation. These constants were entered into the CPC via the CPC operator module. Type II addressable constants are related to measured physics test parameter, uncertainties, allowances and adjustments. Values are determined or confirmed during startup tests following each fuel loading and are not expected to change during cycle operation. These addressable constants are typically entered into the CPCs from diskettes.

2.0 SAFETY EVALUATION

We have reviewed the safety effects of removing the CPC addressable constants from the Technical Specifications. Based on our review, we have determined that the proposed change is administrative in nature and does not authorize any physical change to the plant's safety-related structures, systems or components. Any modifications to the addressable constants are accomplished through strict administrative procedures as required by Technical Specifications 6.8.1(g). CPC software changes involving addressable constants or software limit values are made and tested under NRC approved software change procedures and are available for NRC review. CPC modifications which result in an unreviewed safety question or a Technical Specification change including additions or changes to software limits on the addressable constants will require NRC approval prior to implementation. In addition, the CPC software are equipped with automatic acceptable input checks against range limits that are specified by the CPC functional design requirement and CPC surveillance requirements (Technical Specification Table 4.3-1) require that the correct values of addressable constants shall be verified to be installed in each operable CPC at least once a week.

Based on these requirements, and on the fact that the NRC has previously approved the deletion of CPC addressable constants from the Palo Verde Unit 2, Waterford Unit 3 and the San Onofre Unit 2 and 3 Technical Specifications, we find the proposed removal of the addressable constants and related administrative requirements from the ANO-2 Technical Specifications acceptable.

With respect to the deletion of Notation (10) from Table 4.3-1, we find the proposed deletion acceptable based on the fact that Amendment No. 39 to Facility Operating License No. NPF-6 for ANO-2 specified that the notation which allowed a 30-day waiver of certain reactor protective system functional tests would not be valid after January 31, 1983.

3.0 EVALUATION SUMMARY

The staff has reviewed the proposed revisions to ANO-2 Technical Specification 2.2.2, CPC Addressable Constants; Table 2.2-1, Type I and Type II CPC Addressable Constants; the associated Bases; Table 4.3-1 and Notations; and Administrative Control 6.8.1. Based on our evaluation given in the proceeding sections, we find these revisions acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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.

4.0 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal contributor to this SE was L. Kopp.

Dated: July 22, 1986