8005150292
P

DISTRIBUTION
H. Denton/E. Case
~~Docket~~ File 50-327
NRC PDR
Local PDR
LWR-4 File
Attorney, ELD - E. Reis
D. Ross
D. Vassallo
S. Varga
L. Rubenstein
C. Stahle
M. Service
F. Williams
H. Berkow/W. Russell
L. Cobb
IE (5)
MPA
M. Jinks, OA
R. Diggs, ADM
V. Moore
D. Muller
W. Kreger
M. Ernst
R. Denise
R. Mattson
J. Knight
S. Hanauer
B. Scharf, ADM
D. Skovholt
I. Dinitz
A. Toalston, AIG
S. Keblusek
M. Duncan
H. Bristow, NMSS
S. Duncan, NMSS
D. Eisenhut
B. Grimes
W. Pasciak
C. Miles, PA
C. Gaskin, NMSS
M. Virgilio

bcc: NSIC
TIC
ASLAB
ASLBP
ACRS (16)

API
2

6D

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

APR 22 1980

Docket No.: 50-327

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: ISSUANCE OF AMENDMENT NO. 1 TO LICENSE NO. DPR-77 - SEQUOYAH NUCLEAR PLANT, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 1 to License No. DPR-77.

This amendment approves several of your proposed changes to the Appendix A Technical Specifications for the Sequoyah Nuclear Plant, Unit 1, dealing with the auxiliary feedwater system, the rod position indicating system, and divider barrier seal surveillance. These changes were proposed in your letters dated March 28, 1980 and April 14, 1980. We have also corrected typographical errors to Table 4.8-1, page 6-21 and page 3/4 1-10 contained in Appendix A. We will continue our review of the remaining proposed changes included in your letter of March 28, 1980.

A copy of the related safety evaluation supporting Amendment No. 1 to License DPR-77 is enclosed. Also enclosed is a copy of the Federal Register Notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

L. S. Rubenstein

L. S. Rubenstein, Acting Chief
Light Water Reactors Branch No. 4
Division of Project Management

Enclosures:
1. Amendment No. 1 to
   License No. DPR-77
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

60

Tennessee Valley Authority

APR 2 2 1980

ccs:
Herbert S. Sanger, Jr., Esq.
General Counsel
Tennessee Valley Authority
400 Commerce Avenue
E 11B 33
Knoxville, Tennessee 37902

Mr. H. N. Culver
Tennessee Valley Authority
400 Commerce Avenue, 249A HBB
Knoxville, Tennessee 37902

Mr. Michael Harding
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Mr. David Lambert
Tennessee Valley Authority
400 Chestnut Street Tower II
Chattanooga, Tennessee 37401

Mr. J. F. Cox
Tennessee Valley Authority
400 Commerce Avenue, W10C131C
Knoxville, Tennessee 37902

Resident Inspector/Sequoyah NPS
c/o U. S. Nuclear Regulatory Commission
P. O. Box 699
Hixson, Tennessee 37343

Director
Office of Urban & Federal Affairs
108 Parkway Towers
404 James Robertson Way
Nashville, Tennessee 37219

The Honorable Don Moore, Jr.
County Judge
Hamilton County Courthouse
Chattanooga, Tennessee 37201

U. S. Environmental Protection Agency
ATTN: EIS Coordinator
Region IV Office
345 Courtland St., N. E.
Atlanta, Georgia 30308

Attorney General
Supreme Court Building
Nashville, Tennessee 37219

8005150297

P

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

LICENSE NO. DPR-77

AMENDMENT NO. 1

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

   A. The application for amendment to the Sequoyah Nuclear Plant, Unit 1 (the facility) license, DPR-77, filed by the Tennessee Valley Authority (licensee), dated March 28, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, 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 hereby amended by page changes to the Appendix A Technical Specifications as indicated in the attachment to this license amendment and revised pages 3/4 7-5, 3/4 7-6, 3/4 8-7, 3/4 10-5, 3/4 6-36, 3/4 6-37, 6-21, and 3/4 1-10.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 1, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

F. This amended license is effective as of the date of issuance.

For the Nuclear Regulatory Commission

L S Rubenstein

L. S. Rubenstein, Acting Chief
Light Water Reactors, Branch No. 4
Division of Project Management

Date of Issuance: April 22, 1980

Enclosure:
Appendix A Technical
Specification changes

ATTACHMENT TO LICENSE AMENDMENT NO. 1

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

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

Pages

3/4 1-10
3/4 6-36
3/4 6-37
3/4 7-5
3/4 7-6
3/4 8-7
3/4 10-5
6-21

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE shutdown board.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2400 psig when tested pursuant to Specification 4.0.5.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 2400 psig when tested pursuant to Specification 4.0.5.

REFUELING CANAL DRAINS

LIMITING CONDITION FOR OPERATION

3.6.5.8 The refueling canal drains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With a refueling canal drain inoperable, restore the drain to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.8 Each refueling canal drain shall be demonstrated OPERABLE:

a. Prior to increasing the Reactor Coolant System temperature above 200°F after each partial or complete filling of the canal with water by verifying that the blind flange is removed from the drain line and that the drain is not obstructed by debris, and

b. At least once per 92 days by verifying, through a visual inspection, that the blind flange is removed and there is no debris that could obstruct the drain.

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:

a. Removing and pressure testing the divider barrier seal test coupons in accordance with Table 3.6-3,

b. Visually inspecting at least 95 percent of the seal's entire length and:

   1. Verifying that the seal and seal mounting bolts are properly installed, and

   2. Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

TABLE 3.6-3

DIVIDER BARRIER SEAL
ACCEPTABLE PHYSICAL PROPERTIES

Material	Differential Pressure	Elongation
Presray Corp. EPDM Compound E603 (2 ply Dacron Coated EPDM)	15 psid after LOCA environment simulation*	NA

*The test sequence will be as follows: 2 coupons will be tested to 60 psid; with no failures, the results are acceptable. If a failure occurs at 60 psig, 4 coupons will be tested to 30 psid; with no failures, the results are acceptable. If a failure occurs at 30 psid, 5 coupons will be sent to the manufacture for LOCA environment simulation (radiation, humidity, temperature) and testing to 15 psid.

3/4.6.6 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.6.1 The primary containment vacuum relief valves shall be OPERABLE with an actuation set point of less than or equal to 0.1 PSID.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one primary containment vacuum relief valve inoperable, restore the valve to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate shutdown boards, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two capable of being powered from separate shutdown boards and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a differential pressure of greater than or equal to 1397 psid on recirculation flow.

2. Verifying that the steam turbine driven pump develops a differential pressure of greater than or equal to 1183 psid on recirculation flow when the secondary steam supply pressure is greater than 842 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

4. Verifying that each automatic control valve in the flow path is OPERABLE whenever the auxiliary feedwater system is placed in automatic control or when above 10% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

   2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 100 Valid Tests*	Test Frequency
≤ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
≥ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last 100 valid tests". Entry into this test schedule shall be made at the 31 day test frequency.

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator sets 1A-A and 2A-A or 1B-B and 2B-B each with:

1. Two diesels driving a common generator,

2. Two engine-mounted fuel tanks containing a minimum volume of 250 gallons of fuel per tank,

3. A fuel storage system containing a minimum volume of 62,000 gallons of fuel,

4. A fuel transfer pump, and

5. A separate 125-volt D.C. distribution panel, 125-volt D.C. battery bank and associated charger.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 (except for requirement 4.8.1.1.2.a.5), 4.8.1.1.3, and 4.8.1.1.4.

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the position indication system inoperable, or more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

* This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

   1. A description of the event and equipment involved.

   2. Cause(s) for the unplanned release.

   3. Actions taken to prevent recurrence.

   4. Consequences of the unplanned release.

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when averaged over any calendar quarter sampling period.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1 and 6.8.4.

8005150303
P

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 1 TO LICENSE DPR-77

TENNESSEE VALLEY AUTHORITY

Introduction

By letter dated March 28, 1980 the Tennessee Valley Authority (TVA) proposed changes to the technical specifications for Sequoyah Unit 2 dealing with the auxiliary feedwater system and the rod position indicating system. We have evaluated these changes.

Two changes were proposed which would provide operational flexibility during startup.

a. The steam driven auxiliary feedwater pump cannot be demonstrated operable before entering Mode 3 (HOT STANDBY) due to insufficient steam supply. This demonstration could be performed during Mode 3 operation.

b. The initial calibration of the rod position indication system could be performed during rod bank withdrawal provided the Keff is maintained less than or equal to 0.95 and only one bank is withdrawn from the fully inserted position at one time.

In addition to the above proposed changes (1) the licensee has agreed to modify the surveillance requirements on testing the auxiliary feedwater system to include each of the safety related auxiliary feedwater actuation signals and (2) we are correcting a typographical error to Table 4.8-1 and page 6-21.

On April 14, 1980, TVA requested a change to the surveillance requirements for the divider barrier seal. This barrier prevents the flow of steam during an accident from the lower compartment to the upper compartment.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types of total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Evaluation

The proposed change to the Sequoyah Technical Specifications for divider barrier seal surveillance involves use of a different method of testing for the divider barrier seal sample coupons to assure that seal integrity is maintained in the event of an accident. Rather than testing the tensile strength of a coupon as previously prescribed, the licensee proposes a new method which would test the ability of the coupon to withstand a differential pressure of at least 15 psid without rupturing or otherwise losing integrity. The maximum calculated differential pressure which the seal would be expected to experience during an accident is no more than 12 psid. In view of the conservatism in the calculational methods, the actual expected maximum differential pressure would be less than 12 psid. The coupons for the 15 psid test would also be subjected to a LOCA environment simulation (radiation, humidity, temperature) before the proposed testing to 15 psid. The acceptance level of 15 psid provides adequate margin for continued assurance of seal integrity during an accident, considering the Technical Specification test frequency (i.e., once per 18 months).

In addition, the licensee proposes to test coupons (before LOCA environment simulation) at a series of higher differential pressures before performing the 15 psid acceptance test. If tests at 60 psid produced no failures, the results would be acceptable and further testing would not be performed. A failure during the 60 psid test would indicate a need for testing at 30 psid, and no failures there would mean acceptance and a stop to further testing. A failure at 30 psid would then be followed by the 15 psid testing described above. We find that this "screening" test sequence is acceptable.

We therefore conclude that the proposed testing method, for determining the integrity of the divider barrier seal will assure the integrity of the seal in the event of an accident, and that the proposed change to the Technical Specifications is acceptable.

The other changes, above, add requirements or clarify requirements currently stated in the Standard Technical Specifications and are acceptable.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment 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: April 22, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-327

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT

LICENSE NO. DPR-77

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 1 to License No. DPR-77, issued to Tennessee Valley Authority (licensee) which revised Technical Specifications Appendix A issued with License No. DPR-77 for the Sequoyah Nuclear Plant, Unit 1. These changes deal with the auxiliary feedwater system, the rod position indicating system, divider barrier surveillance, and the correction of typographical errors. The amendment is effective as of its date of issuance.

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 was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, it has further been concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) Tennessee Valley Authority letter, dated March 29, 1980, (2) Amendment No. 1 to License No. DPR-77 with Appendix A Technical Specification page changes, 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. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland this 22d day of April, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

L. S. Rubenstein

L. S. Rubenstein, Acting Chief
Light Water Reactors, Branch No. 4
Division of Project Management

APR 22 1980

Docket No.: 50-327

Mr. H. G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: ISSUANCE OF AMENDMENT NO. 1 TO LICENSE NO. DPR-77 - SEQUOYAH NUCLEAR PLANT, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 1 to License No. DPR-77.

This amendment approves several of your proposed changes to the Appendix A Technical Specifications for the Sequoyah Nuclear Plant, Unit 1, dealing with the auxiliary feedwater system, the rod position indicating system, and divider barrier seal surveillance. These changes were proposed in your letters dated March 28, 1980 and April 14, 1980. We have also corrected typographical errors to Table 4.8-1, page 6-21 and page 3/4 1-10 contained in Appendix A. We will continue our review of the remaining proposed changes included in your letter of March 28, 1980.

A copy of the related safety evaluation supporting Amendment No. 1 to License DPR-77 is enclosed. Also enclosed is a copy of the Federal Register Notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/

L. S. Rubenstein, Acting Chief
Light Water Reactors Branch No. 4
Division of Project Management

Enclosures:
1. Amendment No. 1 to License No. DPR-77
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

OELD
CWoodhead
4/21/80

No legal objection to form of FR Notice and Proposed Amendment provided that SER has "Boiler Plate" No Sign Hazards finding. Foregoing to support statement in the Notice

OFFICE	DPM	DPM:LWR-4	DOR	DPM:LWR-4		
SURNAME	Rushbrook/Service (mec)(JT)	CStahle	JWetmore	LSRubenstein		
DATE	4/ /80	4/21/80	4/ /80	4/22/80		

NRC FORM 318 (9-76) NRCM 0240

☆U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

LICENSE NO. DPR-77

AMENDMENT NO. 1

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

   A. The application for amendment to the Sequoyah Nuclear Plant, Unit 1 (the facility) license, DPR-77, filed by the Tennessee Valley Authority (licensee), dated March 28, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, 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 hereby amended by page changes to the Appendix A Technical Specifications as indicated in the attachment to this license amendment and revised pages 3/4 7-5, 3/4 7-6, 3/4 8-7, 3/4 10-5, 3/4 6-36, 3/4 6-37, 6-21, and 3/4 1-10.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 1, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

F. This amended license is effective as of the date of issuance.

For the Nuclear Regulatory Commission

/s/

L. S. Rubenstein, Acting Chief
Light Water Reactors, Branch No. 4
Division of Project Management

Date of Issuance: April 22, 1980

Enclosure:
Appendix A Technical
Specification changes

OFFICE	DPM	DPM:LWR-4	DOR	OELD	DPM:LWR-4	
SURNAME	Rushbrook/Service	CStahle	JWetmore		LSRubenstein	
DATE	4/ /80	4/21/80	4/[illegible]/80	4/21/80	4/22/80	

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

LICENSE NO. DPR-77

AMENDMENT NO. 1

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

   A. The application for amendment to the Sequoyah Nuclear Plant, Unit 1 (the facility) license, DPR-77, filed by the Tennessee Valley Authority (licensee), dated March 28, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, as amended, 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 hereby amended by page changes to the Appendix A Technical Specifications as indicated in the attachment to this license amendment and revised pages 3/4 7-5, 3/4 7-6, 3/4 8-7, 3/4 10-5, 3/4 6-36, 3/4 6-37, 6-21, and 3/4 1-10.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 1, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

F. This amended license is effective as of the date of issuance.

For the Nuclear Regulatory Commission

L. S. Rubenstein

L. S. Rubenstein, Acting Chief
Light Water Reactors, Branch No. 4
Division of Project Management

Date of Issuance: April 22, 1980

Enclosure:
Appendix A Technical
Specification changes

ATTACHMENT TO LICENSE AMENDMENT NO. 1

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

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

Pages

3/4 1-10
3/4 6-36
3/4 6-37
3/4 7-5
3/4 7-6
3/4 8-7
3/4 10-5
6-21

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 2400 psig when tested pursuant to Specification 4.0.5.

CONTAINMENT SYSTEMS

DIVIDER BARRIER SEAL

LIMITING CONDITION FOR OPERATION

3.6.5.9 The divider barrier seal shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the divider barrier seal inoperable, restore the seal to OPERABLE status prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.5.9 The divider barrier seal shall be determined OPERABLE at least once per 18 months during shutdown by:

a. Removing and pressure testing the divider barrier seal test coupons in accordance with Table 3.6-3,

b. Visually inspecting at least 95 percent of the seal's entire length and:

   1. Verifying that the seal and seal mounting bolts are properly installed, and

   2. Verifying that the seal material shows no visual evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearances.

TABLE 3.6-3

DIVIDER BARRIER SEAL
ACCEPTABLE PHYSICAL PROPERTIES

Material	Differential Pressure	Elongation
Presray Corp. EPDM Compound E603 (2 ply Dacron Coated EPDM)	15 psid after LOCA environment simulation*	NA

*The test sequence will be as follows: 2 coupons will be tested to 60 psid; with no failures, the results are acceptable. If a failure occurs at 60 psig, 4 coupons will be tested to 30 psid; with no failures, the results are acceptable. If a failure occurs at 30 psid, 5 coupons will be sent to the manufacture for LOCA environment simulation (radiation, humidity, temperature) and testing to 15 psid.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate shutdown boards, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two capable of being powered from separate shutdown boards and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a differential pressure of greater than or equal to 1397 psid on recirculation flow.

2. Verifying that the steam turbine driven pump develops a differential pressure of greater than or equal to 1183 psid on recirculation flow when the secondary steam supply pressure is greater than 842 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

4. Verifying that each automatic control valve in the flow path is OPERABLE whenever the auxiliary feedwater system is placed in automatic control or when above 10% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 100 Valid Tests*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last 100 valid tests". Entry into this test schedule shall be made at the 31 day test frequency.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the position indication system inoperable, or more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

* This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 1 TO LICENSE DPR-77

TENNESSEE VALLEY AUTHORITY

Introduction

By letter dated March 28, 1980 the Tennessee Valley Authority (TVA) proposed changes to the technical specifications for Sequoyah Unit 2 dealing with the auxiliary feedwater system and the rod position indicating system. We have evaluated these changes.

Two changes were proposed which would provide operational flexibility during startup.

a. The steam driven auxiliary feedwater pump cannot be demonstrated operable before entering Mode 3 (HOT STANDBY) due to insufficient steam supply. This demonstration could be performed during Mode 3 operation.

b. The initial calibration of the rod position indication system could be performed during rod bank withdrawal provided the Keff is maintained less than or equal to 0.95 and only one bank is withdrawn from the fully inserted position at one time.

In addition to the above proposed changes (1) the licensee has agreed to modify the surveillance requirements on testing the auxiliary feedwater system to include each of the safety related auxiliary feedwater actuation signals and (2) we are correcting a typographical error to Table 4.8-1 and page 6-21.

On April 14, 1980, TVA requested a change to the surveillance requirements for the divider barrier seal. This barrier prevents the flow of steam during an accident from the lower compartment to the upper compartment.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types of total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Evaluation

The proposed change to the Sequoyah Technical Specifications for divider barrier seal surveillance involves use of a different method of testing for

OFFICE ▶						
SURNAME ▶						
DATE ▶						

the divider barrier seal sample coupons to assure that seal integrity is maintained in the event of an accident. Rather than testing the tensile strength of a coupon as previously prescribed, the licensee proposes a new method which would test the ability of the coupon to withstand a differential pressure of at least 15 psid without rupturing or otherwise losing integrity. The maximum calculated differential pressure which the seal would be expected to experience during an accident is no more than 12 psid. In view of the conservatism in the calculational methods, the actual expected maximum differential pressure would be less than 12 psid. The coupons for the 15 psid test would also be subjected to a LOCA environment simulation (radiation, humidity, temperature) before the proposed testing to 15 psid. The acceptance level of 15 psid provides adequate margin for continued assurance of seal integrity during an accident, considering the Technical Specification test frequency (i.e., once per 18 months).

In addition, the licensee proposes to test coupons (before LOCA environment simulation) at a series of higher differential pressures before performing the 15 psid acceptance test. If tests at 60 psid produced no failures, the results would be acceptable and further testing would not be performed. A failure during the 60 psid test would indicate a need for testing at 30 psid, and no failures there would mean acceptance and a stop to further testing. A failure at 30 psid would then be followed by the 15 psid testing described above. We find that this "screening" test sequence is acceptable.

We therefore conclude that the proposed testing method, for determining the integrity of the divider barrier seal will assure the integrity of the seal in the event of an accident, and that the proposed change to the Technical Specifications is acceptable.

The other changes, above, add requirements or clarify requirements currently stated in the Standard Technical Specifications and are acceptable.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment 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: April 22, 1980

OFFICE	DPM:LWR-4	DPM:LWR-4				
SURNAME	CStahle	LRubenstein				
DATE	4/22/80	4/22/80				

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 1 TO LICENSE DPR-77

TENNESSEE VALLEY AUTHORITY

Introduction

By letter dated March 28, 1980 the Tennessee Valley Authority (TVA) proposed changes to the technical specifications for Sequoyah Unit 2 dealing with the auxiliary feedwater system and the rod position indicating system. We have evaluated these changes.

Two changes were proposed which would provide operational flexibility during startup.

a. The steam driven auxiliary feedwater pump cannot be demonstrated operable before entering Mode 3 (HOT STANDBY) due to insufficient steam supply. This demonstration could be performed during Mode 3 operation.

b. The initial calibration of the rod position indication system could be performed during rod bank withdrawal provided the Keff is maintained less than or equal to 0.95 and only one bank is withdrawn from the fully inserted position at one time.

In addition to the above proposed changes (1) the licensee has agreed to modify the surveillance requirements on testing the auxiliary feedwater system to include each of the safety related auxiliary feedwater actuation signals and (2) we are correcting a typographical error to Table 4.8-1 and page 6-21.

On April 14, 1980, TVA requested a change to the surveillance requirements for the divider barrier seal. This barrier prevents the flow of steam during an accident from the lower compartment to the upper compartment.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types of total amounts nor an increase in power level and will not result in

OFFICE ▶						
SURNAME ▶						
DATE ▶						

any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

The proposed change to the Sequoyah Technical Specifications for divider barrier seal surveillance involves use of a different method of testing for the divider barrier seal sample coupons to assure that seal integrity is maintained in the event of an accident. Rather than testing the tensile strength of a coupon as previously prescribed, the licensee proposes a new method which would test the ability of the coupon to withstand a differential pressure of at least 15 psid without rupturing or otherwise losing integrity. The maximum calculated differential pressure which the seal would be expected to experience during an accident is no more than 12 psid. In view of the conservatism in the calculational methods, the actual expected maximum differential pressure would be less than 12 psid. The coupons for the 15 psid test would also be subjected to a LOCA environment simulation (radiation, humidity, temperature) before the proposed testing to 15 psid. The acceptance level of 15 psid provides adequate margin for continued assurance of seal integrity during an accident, considering the Technical Specification test frequency (i.e., once per 18 months).

In addition, the licensee proposes to test coupons (before LOCA environment simulation) at a series of higher differential pressures before performing the 15 psid acceptance test. If tests at 60 psid produced no failures, the results would be acceptable and further testing would not be performed. A failure during the 60 psid test would indicate a need for testing at 30 psid, and no failures there would mean acceptance and a stop to further testing. A failure at 30 psid would then be followed by the 15 psid testing described above. We find that this "screening" test sequence is acceptable.

We therefore conclude that the proposed testing method, for determining the integrity of the divider barrier seal will assure the integrity of the seal in the event of an accident, and that the proposed change to the Technical Specifications is acceptable.

The other changes, above, add requirements or clarify requirements currently stated in the Standard Technical Specifications and are acceptable.

OFFICE	DPM:LWR-4	DPM:LWR-4				
SURNAME	CStahle	LRubenstein				
DATE	4/ /80	4/ /80				

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-327

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENT

LICENSE NO. DPR-77

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 1 to License No. DPR-77, issued to Tennessee Valley Authority (licensee) which revised Technical Specifications Appendix A issued with License No. DPR-77 for the Sequoyah Nuclear Plant, Unit 1. These changes deal with the auxiliary feedwater system, the rod position indicating system, divider barrier surveillance, and the correction of typographical errors. The amendment is effective as of its date of issuance.

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 was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, it has further been concluded that the amendment involves an

OFFICE ▶						
SURNAME ▶						
DATE ▶						

action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) Tennessee Valley Authority letter, dated March 29, 1980, (2) Amendment No. 1 to License No. DPR-77 with Appendix A Technical Specification page changes, 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. 20555, Attention: Director, Division of Project Management.

Dated at Bethesda, Maryland this 22nd day of April, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

L. S. Rubenstein, Acting Chief
Light Water Reactors, Branch No. 4
Division of Project Management

OFFICE	DPM	DPM:LWR-4	OELD	DPM:LWR-4		
SURNAME	Rushbrook/Service	CStahle		LSRubenstein		
DATE	4/ /80	4/21/80	4/21/80	4/22/80		

NRC FORM 318 (9-76) NRCM 0240 ☆U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369