May 25, 1993

Docket No. 50-461

Mr. Frank A. Spangenberg
Licensing and Safety
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Dear Mr. Spangenberg:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M85816)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 75 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. The amendment is in response to your application dated February 11, 1993 (U-602085).

The amendment modifies the Clinton Power Station Technical Specifications by: (1) revising Specification 5.3.1, "Fuel Assemblies," to make the fuel design features more generic to allow use of other NRC-approved fuel designs, (2) revising Specification 5.3.2, "Control Rod Assemblies," to allow the use of NRC-approved control rod designs which contain hafnium metal in addition to boron carbide powder, and (3) revising Specification 3.3.1, "Reactor Protection System Instrumentation," and its Bases to transfer the specific value of the simulated thermal power time constant for the Average Power Range Neutron Monitors (APRMs) from the Technical Specifications to the Core Operating Limits Report (COLR).

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly <u>Federal Register</u> notice.

Sincerely,

Original signed by:

Douglas V. Pickett, Project Manager
Project Directorate III-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 75 to NPF-62
2. Safety Evaluation

cc w/enclosures:
see next page

LA:PD32:DRPW	PM:PD32:DRPW	D:PD32:DRPW	OGC-OWFN	SRXB:DSSA
CMoore	DPickett/ DVP	JDyer	J Hull	RJones
5/5/93	5/6/93	5/11/93	5/17/93	5/6/93

DOCUMENT NAME: CL85816.AMD

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Mr. Frank A. Spangenberg
Illinois Power Company

Clinton Power Station
Unit No. 1

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ILLINOIS POWER COMPANY, ET AL.

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. NPF-62

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

   A. The application for amendment by Illinois Power Company* (IP), and Soyland Power Cooperative, Inc. (the licensees) dated February 11, 1993, 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-62 is hereby amended to read as follows:

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*Illinois Power Company is authorized to act as agent for Soyland Power Cooperative, Inc. and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 75 , are hereby incorporated into this license. Illinois Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett for

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 25, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 75

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

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. The corresponding overleaf pages, as indicated by an asterisk, are provided to maintain document completeness.

Remove Pages	Insert Pages
B 2-7	B 2-7
* B 2-8	* B 2-8
3/4 3-7	3/4 3-7
* 3/4 3-8	* 3/4 3-8
* 3/4 3-9	* 3/4 3-9
3/4 3-10	3/4 3-10
5-5	5-5
* 5-6	* 5-6

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

The APRM trip system is calibrated using heat balance data taken during steady-state conditions. Fission chambers provide the basic input to the system and therefore, the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-High setpoint; i.e; for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow-Biased Simulated Thermal Power-High setpoint, a time constant specified in the COLR is introduced into the flow-biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1. In these flow biased equations, the variable W is the loop recirculation flow as a percentage of the total loop recirculation flow which produces a rated core flow of 84.5 million lbs/hr.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase during operation will also tend to increase the power of the reactor by compressing voids, thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure and turbine control valve fast closure trips are bypassed. For a turbine trip or load rejection under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

5. Reactor Vessel Water Level-High

A reactor scram from high reactor water level, approximately two feet above normal operating level, is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater. An excess of feedwater entering the vessel would be detected by the level increase in a timely manner. This scram feature is only effective when the reactor mode switch is in the Run position, because at THERMAL POWER levels below 10% to 15% of RATED THERMAL POWER, the approximate range of power level for changing to the Run position, the safety margins are more than adequate without a reactor scram.

6. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

7. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time, the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

8. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or loss of drywell cooling. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and to the primary containment. The trip setting was selected as low as possible to minimize heat loads of equipment located within the primary containment and to avoid spurious trips.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	≤ 0.09**
c. Neutron Flux - High	≤ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.33
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.03
5. Reactor Vessel Water Level - High, Level 8	≤ 1.03
6. Main Steam Line Isolation Valve - Closure	≤ 0.04
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	
a. Level Transmitter	NA
b. Float Switches	NA
10. Turbine Stop Valve - Closure	≤ 0.04
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤ 0.05#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including a simulated thermal power time constant specified in the COLR.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b) S	S/U(c), W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:(f)				
a. Neutron Flux - High, Setdown	S/U,S,(b) S	S/U(c), W W	SA SA	2 3, 4, 5
b. Flow-Biased Simulated Thermal Power - High	S	S/U(c), Q	W(d)(e), SA, R(i)	1
c. Neutron Flux - High	S	S/U(c), Q	W(d)(e), SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R(g)	1, 2(j)
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R(g)	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	Q	R(g)	1
6. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
7. Main Steam Line Radiation - High	S	Q	R	1, 2(j)
8. Drywell Pressure - High	S	Q	R(g)	1, 2(l)

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	Q	R[g]	1, 2, 5[k]
b. Float Switches	NA	Q	R	1, 2, 5[k]
10. Turbine Stop Valve - Closure	NA	Q[m]	R[m]	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q[m]	R[m]	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	Q	NA	1, 2, 3, 4, 5

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1 decade during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.

(e) This calibration shall consist of a setpoint verification of the Neutron Flux-High and the Flow Biased Simulated Thermal Power-High trip functions. The Flow Biased Simulated Thermal-High trip function is verified using a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Calibrate the analog trip module at least once per 92 days.

(h) Deleted.

(i) This calibration shall consist of verifying that the simulated thermal power time constant is within the limits specified in the COLR.

(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(k) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(l) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required to be OPERABLE per Special Test Exception 3.10.1.

(m) The CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Fuel Building, the ECCS pump rooms and the containment gas control boundary, including extension, and has a minimum free volume of 1,710,000 cubic feet.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 624 fuel assemblies. Each assembly shall consist of a matrix of zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material, and water rod(s). Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead-use assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 145 cruciform shaped control rod assemblies as approved by the NRC. The control material shall be boron carbide powder (B_4C) and/or hafnium metal.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.

   2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 16,000 cubic feet at a nominal steam dome saturation temperature of 549°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 6.4375 inch center-to-center distance between fuel assemblies placed in the storage racks in the Fuel Building storage pool. A nominal center-to-center spacing between rows of 12.25 inches and within the rows of 7.00 inches for fuel assemblies placed in the storage rack in the Upper Containment Fuel Pool.

5.6.1.1 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 754'0".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2522 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-62

ILLINOIS POWER COMPANY, ET AL.

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By letter dated February 11, 1993, the licensee stated that a different (but NRC-approved) fuel design (GE10) would be utilized during the fourth refueling outage currently scheduled to begin September 26, 1993. In order to support this outage, the following technical specification changes were proposed:

- Technical Specification 5.3.1, "Fuel Assemblies," provides a fuel design feature description which includes information of the number of fuel and water rods, cladding material, active fuel length and bundle enrichments. This description will change with new fuel designs. In addition, this description will change if fuel reconstitution is necessary or if lead-use assemblies are used. In order to make the fuel design description more generic to allow the use of NRC-approved designs, the NRC issued Supplement 1 to Generic Letter 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications." This correspondence provided generic wording to be used in the fuel design feature description so that NRC-approved fuel designs could be utilized without the burden of amending the technical specifications. The licensee has proposed incorporating the wording provided by the staff.

- Technical Specification 5.3.2, "Control Rod Assemblies," provides a description stating that the control rods will consist, in part, of boron carbide powder surrounded by a cruciform shaped stainless steel sheath. Such wording precludes the use of any other neutron absorbing material in control rod design. Subsequent to licensing the Clinton Power Station (CPS), the NRC has reviewed and approved the use of hafnium control rods at BWR facilities. In order to use the new or upgraded control rod designs being offered by nuclear vendors, the licensee has proposed revising Specification 5.3.2 to permit the use of hafnium control rods.

- Technical Specification 3.3.1, "Reactor Protection System Instrumentation," specifies the value of the simulated thermal power time constant for the Average Power Range Neutron Monitors. This value is dependent upon the fuel pellet diameter and will change with different fuel designs incorporating 9x9 or 10x10 arrays of fuel pins. (The CPS currently has 8x8 fuel pin arrays.) The NRC issued Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from the Technical Specifications," to permit core reloads using NRC-approved methods without the burden of amending the technical specifications. This correspondence permitted transferring cycle-specific parameters from the technical specifications to the Core Operating Limits Report (COLR). By changing fuel designs, the simulated thermal power time constant will change. Therefore, consistent with the guidance provided in the generic letter, the licensee has proposed transferring the simulated thermal power time constant from the technical specifications to the COLR.

2.0 EVALUATION

Technical Specification 5.3.1, "Fuel Assemblies"

The fuel assembly description found in technical specification 5.3.1 is somewhat detailed and is limited to fuel assemblies incorporating an 8x8 fuel pin array. Descriptive information on cladding material, active fuel length and bundle enrichments limits the licensee's ability to change fuel designs. In order to take advantage of new and improved fuel assembly designs incorporating a 9x9 or 10x10 fuel pin array, a technical specification change would be necessary. For this reason, the licensee proposes to revise the fuel design requirements to be more generic but still require that these designs be developed and analyzed using NRC-approved codes and methods.

On July 31, 1992, the staff issued Supplement 1 to Generic Letter 90-02 as a line-item improvement to accommodate limited fuel reconstitution based on NRC-approved generic topical reports. The generic letter proposed wording to be used by licensees to describe fuel assemblies that incorporate the use of NRC-approved applications of fuel rod configurations. In the licensee's letter of February 11, 1993, the licensee adopted the descriptive wording as suggested by the staff. The wording is less restrictive than that currently found in the CPS technical specifications and will permit the licensee to utilize new and improved fuel designs. Therefore, since the licensee's proposal is consistent with the staff's position, the proposed change is acceptable.

Technical Specification 5.3.2, "Control Rod Assemblies"

The control rod assembly description found in technical specification 5.3.2 limits the licensee to control rods that consist of stainless steel tubes containing 143.70 inches of boron carbide powder surrounded by a cruciform shaped stainless steel sheath. The lack of a reference to other neutron-absorbing materials precludes the use of control rods having designs utilizing a different composition of materials such as hafnium. Subsequent to the licensing of the CPS, the NRC has reviewed and approved control rod designs for BWR facilities using hafnium. Hafnium control rods have certain advantages over boron carbide including reduced stress corrosion cracking due to irradiation effects and a longer reactivity life. Overall, this leads to a longer control rod blade life.

In their letter of February 11, 1993, the licensee proposed rewording Specification 5.3.2 to permit the use of boron carbide powder and/or hafnium metal in the control rod assemblies. In addition, the licensee committed to use control rod assemblies that had been approved by the NRC for use in BWR facilities. Since the staff has previously found hafnium to be an acceptable neutron-absorbing material for control rod assemblies in BWR facilities and the licensee has committed to use a design approved by the staff, the proposed change is acceptable.

Technical Specification 3.3.1, "Reactor Protection System Instrumentation"

Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," discussed the merits of processing license amendments required for each fuel cycle to update the values of cycle-specific parameter limits found in technical specifications. The staff stated that the processing of changes to technical specifications that are developed using an NRC-approved methodology is an unnecessary burden on licensee and NRC resources. The generic letter provided guidance for licensees to remove all cycle-specific parameters from the technical specifications and transfer them to a Core Operating Limits Report (COLR). The Clinton licensee incorporated the staff's guidance of Generic Letter 88-16 and removed the cycle-specific parameters.

As described in the licensee's letter of February 11, 1993, the simulated thermal power time constant is dependent, in part, on the fuel pellet diameter. The value decreases with decreased fuel pellet diameter as it takes less time for the heat to travel through the fuel to the outer surface. CPS Technical Specification Table 3.3.1-2, "Reactor Protection System Response Times," and Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," reference a simulated thermal power time constant based on a 8x8 fuel pin array. By incorporating new fuel designs having 9x9 or 10x10 fuel pin arrays, the fuel pellet diameter will decrease resulting in a revised simulated thermal power time constant.

Consistent with the staff's guidance of Generic Letter 88-16, the licensee has proposed removing the simulated thermal power time constant from the technical specifications and transferring it to the COLR. The licensee has also proposed modifying the Bases section accordingly. These changes will preclude both the licensee and NRC staff from processing license amendments associated with changes in fuel designs. Since fuel designs will be limited to those approved by the staff and the proposal is consistent with previous NRC guidance, the staff finds the licensee's proposal acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 16862). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The staff has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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: Douglas V. Pickett, NRR

Date: May 25, 1993