January 31, 1990

Docket No.: 50-461

Mr. Dale L. Holtzscher
Acting Manager - Licensing and Safety
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, Illinois 61727

[CLINTON AMENDMENT1]
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Dear Mr. Holtzscher:

SUBJECT: TECHNICAL SPECIFICATION CHANGE REQUEST TO DELETE A REACTOR PROTECTION SYSTEM SURVEILLANCE THAT COMPARES THE MEASURED CORE FLOW TO THE EXPECTED CORE FLOW AT THE LOOP FLOW CONTROL SETTINGS FOR CLINTON POWER STATION, UNIT NO. 1 (TAC NO. 73806)

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 5, 1988.

This amendment revises Technical Specification Table 4.3.1.1-1 to delete note (h). Note (h) required a verification that measured core flow was greater than or equal to established core flow at the existing loop flow control.

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

Sincerely,

/s/

John B. Hickman, Project Manager
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects
Office of Reactor Regulation

Enclosures:
1. Amendment No. 30 to License No. NPF-62
2. Safety Evaluation

cc w/enclosures:
See next page

| PDIII-2:LA | PDIII-2:PM | PDIII-2:PD | OGC |
|---|---|---|---|
| LLuther | JHickman:ta | JCraig | RBachmann |
| 12/7/89 | 12/6/89 | 12/22/89 | 1/11/90 |

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

January 31, 1990

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Acting Manager - Licensing and Safety
Clinton Power Station
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Clinton, Illinois 61727

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2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
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 5, 1988, 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:

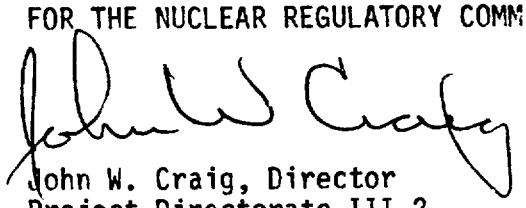
*Illinois Power Company is authorized to act as agent for Soyland Power Cooperative, Inc. and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 30 , 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



John W. Craig, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 31, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 30

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.

| Remove | Insert |
|---|---|
| 3/4 3-8 | 3/4 3-8 |
| 3/4 3-10 | 3/4 3-10 |

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)<br>S | S/U[c], W<br>W | R<br>R | 2<br>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)<br>S | S/U[c], W<br>W | SA<br>SA | 2<br>3, 4, 5 |
| b. Flow-Biased Simulated Thermal Power - High | S | S/U[c], W | W[d][e], SA, R[i] | 1 |
| c. Neutron Flux - High | S | S/U[c], W | W[d][e], SA | 1 |
| d. Inoperative | NA | W | NA | 1, 2, 3, 4, 5 |
| 3. Reactor Vessel Steam Dome Pressure - High | S | M | R[g] | 1, 2[j] |
| 4. Reactor Vessel Water Level - Low, Level 3 | S | M | R[g] | 1, 2 |
| 5. Reactor Vessel Water Level - High, Level 8 | S | M | R[g] | 1 |
| 6. Main Steam Line Isolation Valve - Closure | NA | M | R | 1 |
| 7. Main Steam Line Radiation - High | S | M | R | 1, 2[j] |
| 8. Drywell Pressure - High | S | M | R[g] | 1, 2[l] |

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 31 days.

(h) Deleted.

(i) This calibration shall consist of verifying the 6±0.6 second simulated thermal power time constant.

(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.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. NPF-62

## CLINTON POWER STATION, UNIT NO. 1

## ILLINOIS POWER COMPANY, ET AL.

## DOCKET NO. 50-461

## 1.0 INTRODUCTION

By letter dated February 5, 1988, the Illinois Power Company (IP), et al. (the licensees) requested an amendment to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit 1. The proposed amendment would revise Technical Specification Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements," to delete the Daily Channel Check requirements of note (h) for the Average Power Range Monitor Flow-Biased Simulated Thermal Power - High. Note (h) requires a verification that measured core (total core flow) flow is greater than or equal to established core flow at the existing loop flow control (APRM % flow).

The licensee has conducted discussions with the NRC and General Electric to determine the specific intent of note (h) and has noted and investigated differences in the wording of this item with other comparable Boiling Water Reactors (BWR).

## 2.0 EVALUATION

A review of the various versions of BWR Technical Specifications shows that there are two general versions of the footnote. Neither of the two versions exactly matches the wording appearing in the draft BWR-6 Standard Technical Specifications. (The last official version of the BWR STS for the BWR-5 does not contain the footnote at all.) The version in the Clinton Technical Specifications generally requires verifying that measured total core flow (total jet pump flow) for a given indicated reactor recirculation loop flow (as sensed by the APRMs) is greater than or equal to a previously established total core flow for that particular reactor recirculation loop drive flow.

A number of concerns and/or events may have been considered when the note was incorporated in the Technical Specifications. These are:

1) Flow control valve crudding;

2) Jet pump beam cracking;

3) Jet pump blockage;

4) Core crudding; and

5) Jet pump instrumentation problems.

The flow control valve (FCV) crudding problem does not apply to Clinton since the version of the surveillance at Clinton is not applicable to the drive-flow/FCV-position relationship. The surveillance only considers changes in the core-flow/drive-flow relationship. A check of the drive-flow/FCV-position relationship is provided for in the first surveillance in Technical Specification 3/4.4.1.2 (Jet Pump Operability) which requires verification that the indicated recirculation loop flow does not differ by more than 10% from established FCV-position/loop-flow characteristics.

Jet pump beam cracking or jet pump blockage is already addressed by the requirements of Technical Specification 3/4.4.1.2 (Jet Pump Operability). The surveillance requirements for this Technical Specification (4.4.1.2) are as follows:

> "Each of the above required jet pumps in an operating loop shall be demonstrated OPERABLE at least once per 24 hours when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur:
>
> a. The indicated recirculation loop flow differs by more that 10% from the established flow control valve position-loop flow characteristics.
>
> b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.
>
> c. The indicated jet pump diffuser-to-lower plenum differential pressure (or jet pump flow) of any individual jet pump differs from established patterns by more than 20% (10% for flow)."

If jet pump beam cracking or jet pump blockage were to occur, the problem would be recognized by this surveillance. General Electric Service Information Letter No. 330 identified surveillance 4.4.1.2.c. as an acceptable method for identifying jet pump beam cracking. Failure to meet the acceptance criteria would then require a plant shutdown because the corresponging ACTION under 3.4.1.2 states, "With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours." Therefore, additonal ACTION under the Reactor Protection System (RPS) instrumentation Technical Specification should not be required.

With respect to core crudding, General Electric has indicated that the change in m-ratio (core flow/recirculation loop drive flow) that might occur from beginning-of-cycle to end-of-cycle due to core crudding is so slight that this

phenomenon is not considered to be a significant concern and that the resultant change in the m-ratio would have negligible impact on the Average Power Range Monitor Flow-Biased Simulated Thermal Power trip setpoint.

Finally, with respect to jet pump instrumentation problems, if any of the surveillances under 4.4.1.2 yield unacceptable results, a jet pump instrumentation problem would be suspected. Cross checks against other related instruments associated with the required jet pump surveillances would be performed to determine if it is indeed just an instrument problem. If an instrument problem is identified, then the necessary actions would be performed to restore the instrumentation to operable status. No concern with respect to the Average Power Range Monitor Flow-Biased Simulated Thermal Power trip exists (assuming the Average Power Range Monitor Flow-Biased Simulated Thermal Power instrumentation is operable as verified by the performance of its associated surveillances) because a jet pump instrument problem does not involve an actual change in the m-ratio.

The five concerns are adequately addressed by the RPS instrumentation surveillances, the recirculation flow unit surveillances, and the jet pump surveillance. Jet pump beam cracking or jet pump blockage, which could cause a gross change in m-ration are already covered by specific surveillance requirements. Changes to the m-ration due to core crudding would be expected to be minimal over the course of the cycle. The surveillance requirements for RPS and recirculation flow unit instrumentation provide assurance that the concerns associated with core crudding are adequately addressed. A requirement like Note (h), therefore, should not be included in the RPS instrumentation Technical Specification because the concerns described above do not require it.

## 3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in surveillance requirements for the facility. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

## 4.0 CONCLUSION

The proposed change to delete the Daily Channel Check for the Average Power Range Monitor Flow-Biased Simulated Thermal Power-High scram function and the associated Note (h) from Table 4.3.1.1-1 is acceptable. Adequate steps are taken without Note (h) to detect and take appropriate action for degraduation in the amount of core flow resulting from a given recirculation loop flow.

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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and the security nor to the health and safety of the public.

Prinicipal Contributor: John B. Hickman, NRR/PDIII-2

Dated: January 31, 1990