July 16, 1990

Docket No. 50-461

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Mr. Frank A. Spangenberg
Licensing and Safety
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, Illinois 61727

Dear Mr. Spangenberg:

SUBJECT: AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NO. NPF-62 (TAC NO. 66563)

The Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This amendment is in response to your application dated October 30, 1987.

This amendment modifies Technical Specification Surveillance Requirement 4.4.3.2.1. to indicate that the drywell floor and equipment drain sump leak detection system instrumentation does not include direct quantitative indication of sump level and that the drywell atmospheric radioactivity leak detection system instrumentation does not quantify leakage.

A copy of the Safety Evaluation is also enclosed. The notice of issuance is being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by M. D. Lynch for/

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 39 to License No. NPF-62
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

DOCUMENT NAME: 66563 AMD

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Surname:	PKreutzer	JHickman/lb	JHannon	E Holler
Date:	6/27/90	6/28/90	6/29/90	7/5/90

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OP-1

Mr. Frank A. Spangenberg
Illinois Power Company

Clinton Power Station
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ILLINOIS POWER COMPANY, ET AL.

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
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 October 30, 1987, 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.

(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. 39 , 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 N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: July 16, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 39

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. Corresponding overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 4-13	3/4 4-13
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm IDENTIFIED LEAKAGE (averaged over any 24-hour period).

d. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1, at rated reactor pressure.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed manual or deactivated automatic valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate and gaseous radioactivity at least once per 12 hours (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment drain sump flow rate at least once per 12 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

SURVEILLANCE REQUIREMENTS (Continued)

4.4.3.2.1 (Continued)

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months.*

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve or its associated actuator.

c. As outlined in ASME Code, Section XI, paragraph IWV-3427(b).

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

*The requirements of this specification for valves 1E12F023, 1E51F066, and 1E51F013 will not be completed until prior to startup following the first refueling outage.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

The recirculation flow control valves provide regulation of individual recirculation loop drive flows; which, in turn, will vary the flow rate of coolant through the reactor core over a range consistent with the rod pattern and recirculation pump speed. The recirculation flow control system consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated flow control valves. Solid state control logic will generate a flow control valve "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the flow control valve fails "as is." This design feature insures that the flow control valves do not respond to potentially erroneous control signals.

Electronic limiters exist in the position control loop of each flow control valve to limit the flow control valve stroking rate to 10±1% per second in opening and closing directions on a control signal failure. The analysis of the recirculation flow control failures on increasing and decreasing flow are presented in Sections 15.3 and 15.4 of the FSAR respectively.

The required surveillance interval is adequate to ensure that the flow control valves remain OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 11 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 5 SRVs operating in the relief mode and 6 SRVs operating in the safety mode is acceptable.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems meet the intent of Regulatory Guide 1.45, 3/4.4.3.1 LEAKAGE DETECTION SYSTEMS (Continued)

"Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973 and are consistent with the recommendations of ANSI S67.03, "Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection," 1982. They provide the ability to detect and/or measure leakage from fluid systems in the drywell.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel.

The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 39 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 October 30, 1987, Illinois Power, et al. (the licensees), requested several changes to the Technical Specifications for the Clinton Power Station. Package number 9 of that submittal requested changes to Surveillance Requirements 4.4.3.2.1 to clarify the information that is provided by two of the reactor coolant system leakage detection systems. The first change would add a note to 4.4.3.2.1.a. to indicate that the drywell atmospheric particulate and gaseous radioactivity leak detection system is "not a means of quantifying leakage." The second change would revise 4.4.3.2.1.b. to delete the reference to monitoring the drywell floor and equipment drain sump level.

2.0 EVALUATION

Surveillance Requirement 4.4.3.2.1 of the Clinton Technical Specifications states in part:

> The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:
>
> a. Monitoring the drywell atmospheric particulate and gaseous radioactivity at least once per 12 hours,
>
> b. Monitoring the drywell floor and equipment drain sump level and sump flow rate at least once per 12 hours, ...

Part a. of the surveillance implies that the drywell atmospheric particulate and gaseous radioactivity monitor may be used to quantify leakage. This was consistent with the requirements of NRC Regulatory Guide 1.45. However, industry experience since that regulatory guide was issued has indicated that the uncertainties associated with the calculation of a relationship between drywell leakage and counts per minute by the radioactivity monitor make such a relationship unusable. The NRC has recognized the difficulties in quantifying leakage with a radiological monitoring instrument. By letter dated February 16, 1987, the licensee documented discussions with the NRC and committed to compliance with Regulatory Guide 1.45 except that the drywell atmospheric particulate and gaseous radioactivity monitoring system will not measure leakage as required. The airborne radioactivity monitoring systems will be considered a secondary detection method, along with the monitoring of pressure and temperature to detect gross unidentified leakage. The proposed amendment to this surveillance will add a note to part a. stating that the monitoring of drywell atmospheric particulate and gaseous radioactivity is "not a means of quantifying leakage." This clarification is consistent with staff policy on the use of radiological monitoring for leak detection and consistent with Technical Specification changes approved for other facilities. Therefore, the proposed change is acceptable to the staff.

Part b. of the surveillance addresses the use of the drywell floor and equipment drain sump to quantify leakage. As currently worded the surveillance directs monitoring of the sump level and flow rate. However, the drywell floor drain sump and equipment drain sump leak detection systems use a system of times for pump out to determine average leakage rates. No direct quantitative indication of sump level (other than an alarm) is provided. Although Regulatory Guide 1.45 states that one detection method should be "sump level and flow monitoring" the use of only one method of sump monitoring is consistent with staff policy and has been approved for other facilities. Therefore, the proposed change to delete the reference to monitoring sump level is considered acceptable by the staff.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an evironmental assessment and finding of no significant impact has been prepared and published in the Federal Register on July 11, 1990 (55 FR 28470). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 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; and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Hickman, NRR

Dated: July 16, 1990

UNITED STATES NUCLEAR REGULATORY COMMISSION

ILLINOIS POWER COMPANY ET AL.

DOCKET NO. 50-461

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 39 to Facility Operating License No. NPF-62 issued to the Illinois Power Company (IP), and Soyland Power Cooperative, Inc, (the licensee) for operation of the Clinton Power Station, Unit 1, located in DeWitt County, Illinois. The amendment was effective as of the date of issuance.

The amendment changed the Technical Specifications related to Surveillance Requirement 4.4.3.2.1 for Reactor Coolant System leakage detection. The change indicates that the drywell floor and equipment drain sump leak detection system instrumentation does not include direct quantitative indication of sump level and that the drywell atmospheric radioactivity leak detection system instrumentation does not quantify leakage.

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.

Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on February 18, 1988 (53 FR 4917). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact related to this action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendments dated October 30, 1987; (2) Amendment No. to License No. NPF-62; and (3) the Environmental Assessment and Finding of No Significant Impact dated July 2, 1990 (55 FR 28470). All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street N. W., Washington, D. C., and at the Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland this 16th day of July 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation