April 24, 1992

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. 62 TO FACILITY OPERATING LICENSE NO. NPF-62 (TAC NO. M 82643)

The Commission has issued the enclosed Amendment No. 62 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This action amends the license and revises the Technical Specifications (TS) in response to your application dated December 23, 1991.

The amendment revises TS 4.6.1.2.d, Primary Containment Leakage, by granting an exemption from Type C (Local Leak Rate) testing requirements of Appendix J to 10 CFR Part 50 as they apply to the packing and body-to-bonnet seal of test boundary valve IE51-F374, and by modifying operating license condition 2.D, Exemptions.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

original signed by

C. E. Carpenter, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

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

cc w/enclosures:
See next page

LA:PD33:DRPW	PM:PD33:DRPW	PM:PD33:DRPW	D:PD33:DRPW	OGC
PKreutzer	JLombardo	CCarpenter/bj	JHannon	CPW
4/22/92	4/22/92	4/22/92 CEC	4/24/92	4/23/92

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Illinois Power Company

Clinton Power Station
Unit No. 1

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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. 62
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 December 23, 1991, 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, Facility Operating License No. NPF-62 is hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.D 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. 62 , are hereby incorporated into this license. Illinois Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include: (a) an exemption from the requirements of 10 CFR 70.24 for the criticality alarm monitors around the fuel storate area; (b) an exemption from the requirements of Appendix A to 10 CFR Part 50, General Design Criterion 61 to permit a schedular deferral of completion of preoperational testing of a portion of the Fuel Handling System until prior to offloading fuel from the reactor vessel (Section 14, SSER 8); (c) an exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, substituting the seal leakage test at Pa of paragraph III.D.2(b)(iii) for the entire airlock test at Pa of paragraph III.D.2(b)(ii) of Appendix J when no maintenance has been performed in the airlock that could affect its sealing capability (Section 6.2.6 of SSER 6); (d) an exemption from the requirement of paragraph III.C.3 of Appendix J, exempting the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for the local leak rate tests (Section 6.2.6 of SSER 6); and (e) an exemption from the requirements of paragraph III.B.3 of Appendix J, exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests. The special circumstances regarding each exemption, except for Items (a) and (e) above, are identified in the referenced section of the safety evaluation report and the supplements thereto.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1886, issued November 27, 1985, and relieved IP from the requirement of having a criticality alarm system. IP is hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

The special circumstances regarding the exemption identified in Item (e) above are identified in the safety evaluation accompanying Amendment No. 62 to this license.

These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items (b), (c) and (d) above are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with application, as amended, the provision of the Act, and the rules and regulations of the Commission.

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
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: April 24, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 62

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following pages of the licensee and 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 are provided to maintain document completeness.

Remove	Insert
5 (license)	5 (license)
6 (license)	6 (license)
3/4 6-4	3/4 6-4

(8) <u>Post-Fuel Loading Initial Test Program (Section 14, SER, SSER 5 and SSER 6)</u>

Any changes to the initial test program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(9) <u>Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737, Section 7.5.3.1, SSER 5 and SSER 8, and Section 18, SER, SSER 5 and Safety Evaluation Dated April 17, 1987)</u>

a. IP in accordance with the commitment contained in a letter dated December 11, 1986, shall install and have operational separate power sources for each of the fuel zone level channels as provided for in Regulatory Guide 1.97 prior to startup following the first refueling outage.

b. IP shall submit a detailed control room design final supplemental summary report within 90 days of issuance of the full power license that completes all the remaining items identified in Section 18.3 of the Safety Evaluation dated April 17, 1987.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include: (a) an exemption from the requirements of 10 CFR 70.24 for the criticality alarm monitors around the fuel storage area; (b) an exemption from the requirements of Appendix A to 10 CFR Part 50, General Design Criterion 61 to permit a schedular deferral of completion of preoperational testing of a portion of the Fuel Handling System until prior to offloading fuel from the reactor vessel (Section 14, SSER 8); (c) an exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J, substituting the seal leakage test at Pa of paragraph III.D.2(b)(iii) for the entire airlock test at Pa of paragraph III.D.2(b)(ii) of Appendix J when no maintenance has been performed in the airlock that could affect its sealing capability (Section 6.2.6 of SSER 6); (d) an exemption from the requirement of paragraph III.C.3 of Appendix J, exempting the measured leakage rates from the main steam isolation valves from inclusion in the combined leak rate for the local leak rate tests (Section 6.2.6 of SSER 6); and (e) an exemption from the requirements of paragraph III.B.3 of Appendix J, exempting leakage from the valve packing and the body-to-bonnet seal of valve 1E51-F374 associated with containment penetration 1MC-44 from inclusion in the combined leakage rate for penetrations and valves subject to Type B and C tests. The special circumstances regarding each exemption, except for Items (a) and (e) above, are identified in the referenced section of the safety evaluation report and the supplements thereto.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1886, issued November 27, 1985, and relieved IP from the requirement of having a criticality alarm system. IP is hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

The special circumstances regarding the exemption identified in Item (e) above are identified in the safety evaluation accompanying Amendment No. 62 to this license.

These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items (b), (c) and (d) above are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensees shall fully implement and maintain in effect all provisions of the Commission-approved physical security plan, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Clinton Power Station Physical Security Plan," with revisions submitted through November 30, 1987; "Clinton Power Station Training and Qualification Plan," with revisions submitted through October 1, 1987; and "Clinton Power Station Safeguards Contingency Plan," with revisions submitted through October 1, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. IP shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report as amended, for the Clinton Power Station, Unit No. 1, and as approved in the Safety Evaluation Report (NUREG-0853) dated February 1982 and Supplement Nos. 1 thru 8 thereto subject to the following provision:

> IP may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, IP shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

LIMITING CONDITION FOR OPERATION (Continued)

3.6.1.2 ACTION (Continued):

a. The overall integrated leakage rate(s) to less than or equal to 0.75 La, and

b. The combined leakage rate for all penetrations and all valves subject to Type B and C tests to less than or equal to 0.60 La, and

c. The leakage rate to less than 28 scf per hour for any one main steam line through the isolation valves, and

d. The combined leakage rate for all penetrations shown in Table 3.6.4-1 as secondary containment bypass leakage paths to less than or equal to 0.08 La, and

e. The combined leakage rate for all containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972 and BN-TOP-1 and verifying the result by the Mass Point Methodology described in ANSI/ANS N56.8-1981.

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at Pa, 9.0 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 La the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 La a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 La. The formula to be used is : $[Lo + Lam - 0.25\ La] \leq Lc \leq [Lo + Lam + 0.25\ La]$ where Lc = supplemental test result, Lo = superimposed leakage and Lam = measured Type A leakage.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.2 (Continued)

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the primary containment or bled from the primary containment during the supplemental test to be between 0.75 La and 1.25 La.

d. Type B and C tests shall be conducted*** with gas at Pa, 9.0 psig*, at intervals no greater than 24 months except for tests involving:**

1. Air locks,

2. Main steam line isolation valves,

3. Penetrations using continuous leakage monitoring systems,

4. All containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment, and

5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at Pa, 9.0 psig, at every other reactor shutdown for refueling, but in no case at intervals no greater than once per 3 years.

h. All containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment shall be leak tested at least once per 18 months.

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.8.3.

j. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.d, and 4.6.1.2.g.

*Unless a hydrostatic test is required per Table 3.6.4-1.

**The requirements of this specification for valves 1E12-F023, 1E51-F034, 1E51-F035, 1E51-F390, 1E51-F391, 1E12-F061, 1E12-F062, and 1E51-F013 will not be completed until prior to startup following the first refueling outage.

***Except as provided in NRC-approved exemption to Appendix J to 10 CFR 50 for containment penetration 1MC-44.

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

SAFETY EVUALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 62 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 December 23, 1991, the licensee requested a permanent exemption from the local leak rate testing requirements of Appendix J to 10 CFR Part 50 as they apply to the Reactor Core Isolation Cooling (RCIC) vacuum breaker line associated with containment penetration 1MC-44 and to the packing and body-to-bonnet seal of test boundary valve 1E51-F374. The licensee also proposed an amendment to the facility operating license and changes to the Technical Specifications (TS) to implement the exemption. The subject valve is not a containment isolation valve and is expected to remain open during an accident; therefore, leakage through the valve (past the valve seat and disc) is not a safety concern and there is no requirement to measure it. However, due to the valve's position in the piping relative to the containment isolation valves (described in detail in section 2.0 below), the valve's body is part of the containment boundary. Because of this, leakage out of the valve past the stem packing or body-to-bonnet seal would be leakage out of the containment and must be measured as local leakage, in accordance with the requirements of Appendix J. Unfortunately, this requirement was not well-understood when the plant was designed and built, and the available testing arrangements (i.e., block valves and test, vent, and drain lines) are insufficient to make the required testing possible. The licensee has developed a make-shift method to perform the test, but it is awkward and has considerable attendant costs in terms of time, resources, and radiation exposure.

The licensee submits that the safety benefit to be derived from performing the required testing does not justify the costs. The licensee has therefore requested an exemption from the Appendix J local leak rate testing requirement and proposes an alternate test as described below.

2.0 EVALUATION

Valve 1E51-F374 is associated with containment penetration 1MC-44, the Reactor Core Isolation Cooling (RCIC) vacuum breaker line. The containment isolation valves for this penetration are outside of containment; there are no valves in the line inside containment, where the line simply ends, open to the containment atmosphere. Valve 1E51-F374 is located in the line outside containment, between the containment wall and the first containment isolation valve. It is a block valve which is closed during the local leak rate testing of the adjacent containment isolation valve, allowing that valve to be tested in the "forward" direction; that is, with pressure applied in the same direction as that which would exist if the valve were required to perform its safety function (outward from containment). The position of valve 1E51-F374, outside containment but before the first containment isolation valve, makes the valve's body part of the containment boundary, and leakage through it to the environment (such as through the packing or body-to-bonnet seal) is containment leakage that must be measured and maintained within limits.

Valve 1E51-F374 is a gate valve. Because this valve is normally in the open position, the valve's packing and body-to-bonnet seal are normally exposed to the containment atmosphere. These potential leakage pathways are therefore required to be included in the local leak rate test boundary per Appendix J. However, because of the gate valve design, it cannot be confirmed that the valve's packing and body-to-bonnet seal are exposed to the test pressure when the valve is in the closed position (i.e., during the performance of local leak rate tests). As a result, the requirements of Appendix J would require this valve to be in the open (i.e., post-accident) position during local leak rate testing.

As identified in LER 90-018, several alternatives were evaluated to correct this testing deficiency. One alternative consisted of identifying alternate testing configurations. Another alternative consisted of modifying the valve to allow the body-to-bonnet seal and valve packing to be pressurized during local leak rate testing. Modification of the valve was determined by the licensee to be inappropriate as such a modification would degrade the valve's sealing capability (valve-to-seat), making it more difficult to successfully pass the Type C tests on the adjacent isolation valves. Further, performance of such a modification would result in radiation exposure during implementation (the valve is located in the Residual Heat Removal heat exchanger room).

Alternate testing configurations that were evaluated consisted of installing a plug inside containment in the end of this line and/or connecting the leak rate testing rig to the pipe end. As this line terminates over and approximately 10 feet above the suppression pool, a temporary scaffold would have to be erected to gain access to the pipe end. The licensee estimates that erecting and disassembling a temporary scaffold in this area would take approximately 80 man-hours and result in approximately 100 mrem radiation exposure each refueling outage. (It should be noted that this estimate is based on current plant conditions with no known leaking fuel and no significant safety/relief valve leakage. As a result, background radiation levels for performing these activities would likely increase over plant life). In addition, erecting a temporary scaffold would create additional radioactive waste and would increase the potential for foreign objects to be introduced into the suppression pool.

The licensee has evaluated each of these alternatives and determined that the additional radiation exposure and resource expenses far outweigh the benefits to be gained by including the valve packing and body-to-bonnet seal of valve 1E51-F374 in the local leak rate test boundary. This valve is located in a nominal 3-inch line and is exercised each refueling outage solely for the performance of the Type C test for this containment penetration's associated isolation valves. This line normally contains air at containment pressure and temperature. As a result, the valve packing and body-to-bonnet seal are not subjected to degradation due to large thermal or hydraulic transients. Further, any air leakage through these pathways would be filtered by the standby gas treatment system prior to release to the environment. For these reasons, the licensee believes that leakage through these potential leakage pathways would not be significant, and therefore, inclusion of these pathways in the local leak rate test boundary is not necessary. In addition, these potential leakage pathways are included in the Integrated Leak Rate Test (ILRT) boundary, and thus, any leakage through these pathways will be included in the total leakage rate measured during an ILRT. To provide added assurance that these pathways do not constitute a significant leakage source and to provide additional indication when repairs are necessary, the body-to-bonnet seal and valve packing of valve 1E51-F374 will be leak tested with a soap solution during each ILRT.

The staff finds that the additional assurance of leak-tight integrity of the subject leakage pathways provided by local leak rate testing, when compared to the proposed alternate soap solution test during each ILRT, is not great enough to justify the costs associated with local leak rate testing, described above. The small size and mild environment of the valve makes it unlikely that the packing or body-to-bonnet seals will degrade quickly and experience a leak that would add significantly to the radiological consequences of a LOCA, considering also the action of the standby gas treatment system. The local leak rate test, performed at every refueling outage (but at least every 2 years), would be replaced by the roughly equivalent ILRT-with-soap-solution test performed approximately every 3-1/3 years (typically every other refueling outage). This increase in test interval is acceptable, considering the likely stable nature of the leakage pathways, as discussed above.

Based on the above evaluation, the staff finds the proposed exemption from the local leak rate testing requirements of Appendix J for the packing and body-to-bonnet seal of valve 1E51-F374, and the associated facility operating license and TS changes, to be acceptable, providing the purposed alternate testing (soap solution test during each ILRT) is performed.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the appropriate 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 or changes a surveillance requirement. 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 9445). 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 Contributors: J. Pulsipher
J. Lombardo

Date: April 24, 1992