August 21, 1992

Docket No. 50-461

Mr. Frank A. Spangenberg
Clinton Power Station
P. O. Box 678C
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Clinton, Illinois 61727

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Dear Mr. Spangenberg:

SUBJECT: AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-62 (TAC NO. M77182)

The Commission has issued the enclosed Amendment No. 65 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This amendment revises the Technical Specifications (TS) in response to your application dated July 11, 1990 and supplemented May 7, 1991.

The amendment revises the TS in response to the NRC Generic Letter (GL) 88-01 by changing Sections 4.0.5, Surveillance Requirements for Inservice Inspection and Testing; 3/4.4.3.1, Reactor Coolant System Leakage Detection Systems and Bases; and, 3/4.4.3.2, Operational Leakage and Bases. The revisions add a commitment to perform the Inservice Inspection Program for piping as identified in GL-88-01, change the actions for out of service leakage detection systems, and add Limiting Conditions for Operation and Action Statements for sudden increases in Reactor Coolant System leakage. Administrative changes are also included.

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

Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

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

cc w/enclosures:
See next page

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DOCUMENT NAME: a:cli77182.amd

AT 8/21/92

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

August 21, 1992

Docket No. 50-461

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

Dear Mr. Spangenberg:

SUBJECT: AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. NPF-62 (TAC NO. M77182)

The Commission has issued the enclosed Amendment No. 65 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This amendment revises the Technical Specifications (TS) in response to your application dated July 11, 1990 as supplemented May 7, 1991.

The amendment revises the TS in response to the NRC Generic Letter (GL) 88-01 by changing Section 4.0.5, Surveillance Requirements for Inservice Inspection and Testing, Section 3/4.4.3.1, Reactor Coolant System Leakage Detection Systems; and Section 3/4.4.3.2, Operational Leakage and Bases. The revisions add a commitment to perform the Inservice Inspection Program for piping as identified in GL 88-01, change the actions for out of service leakage detection systems, and add Limiting Conditions for Operation and Action Statements for sudden increases in Reactor Coolant System leakage. Administrative changes are also included.

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Sincerely,

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Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 65 to License No. NPF-62
2. Safety Evaluation

cc w/enclosures:
See next page

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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. 65
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 July 11, 1990, as supplemented May 7, 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, 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. 65, 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 to be implemented within 60 days of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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: August 21, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 65

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 are provided to maintain document completeness.

Remove	Insert
3/4 0-3	3/4 0-3
3/4 4-12	3/4 4-12
-------	3/4 4-12a
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4
--------	B 3/4 4-4a

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC Staff positions on schedule, methods and personnel, and sample expansion included in the generic letter.

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings*:

Valve No.	Low-Low Set Function Setpoint* (psig) ± 15 psi Open	Close
F051D	1033	926
F051C	1073	936
F047F	1113	946
F051B	1113	946
F051G	1113	946

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. With either low-low set function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 The low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 31 days.

b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST at least once per 18 months. Each of the two trip systems or divisions of the low-low set function actuation logic associated with the Nuclear System Protection System shall be manually tested independent of the SELF TEST SYSTEM during separate refueling outages such that both divisions and all channel trips are tested at least once every four fuel cycles.†

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*One channel may be placed in an inoperable status for up to 2 hours for the purpose of performing surveillance testing in accordance with Specification 4.4.2.2.

**The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

†Manual testing for the purpose of satisfying Specification 4.4.2.2.b. is not required until after shutdown during the first regularly scheduled refueling outage.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell atmosphere particulate radioactivity monitoring system,

b. The drywell sump flow monitoring system, and

c. Either the drywell atmosphere gaseous radioactivity monitoring system or the drywell air coolers condensate flow rate monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With only two of the above required leakage detection systems OPERABLE,

a. operation may continue for up to 30 days when the drywell atmosphere particulate radioactivity monitoring system is inoperable provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours.

b. operations may continue:

   1. with the drywell equipment drain sump flow monitoring subsystem inoperable provided the drywell equipment drain sump flow rate is monitored and determined by alternate means at least once per 12 hours,

   2. for up to 30 days with the drywell floor drain sump flow monitoring subsystem inoperable provided the drywell floor drain sump flow rate is monitored and determined by alternate means at least once per 8 hours,

c. operation may continue for up to 30 days when the drywell atmosphere gaseous radioactivity monitoring system and the drywell air coolers condensate flow rate monitoring system are inoperable provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Drywell sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Drywell air cooler condensate flow rate monitoring system performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

d. Flow testing the drywell floor drain sump inlet piping for blockage at least once every 18 months during shutdown.

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.

e. No greater than a 2 gpm increase in UNIDENTIFIED LEAKAGE within a 24-hour period or less during OPERATIONAL CONDITION 1.

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.

d. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 24-hour period or less (during OPERATIONAL CONDITION 1), within 4 hours from the time of discovery isolate the source of increased leakage or verify that the source of increased leakage is not associated with service sensitive Type 304 or 316 austenitic stainless steel; otherwise 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 drain sump flow rate at least once per 8 hours,

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

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

e. 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.

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 USAR 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. With certain exceptions as noted in the Clinton Power Station Updated Safety Analysis Report, these detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. Except for the drywell particulate and gaseous radioactivity monitors, the systems provide the ability to measure leakage from fluid systems in the drywell. The drywell sump flow monitoring system consists of the drywell floor drain sump flow monitoring subsystem and the drywell equipment drain sump flow monitoring subsystem. OPERABILITY of each of these subsystems requires that the applicable portion of the monitoring subsystem associated with the v-notched weir box be OPERABLE. Other portions of the subsystem, including the sump pump control circuit and the associated timer, cycle counter and level switches, may be utilized as appropriate to provide an alternate means of monitoring and determining UNIDENTIFIED or IDENTIFIED leakage under the provisions of the associated ACTION statements for the respective subsystem.

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. With respect to IGSCC-related cracks in service sensitive austenitic stainless steel piping however, an additional limit on the allowed increase in UNIDENTIFIED LEAKAGE (within a 24-hour period or less) is imposed in accordance with Generic Letter 88-01, "NRC Postion on IGSCC in BWR Austenitic Stainless Steel Piping," since an abrupt increase in the UNIDENTIFIED LEAKAGE could be indicative of leakage from such a source. 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 reactor will also be shut down if an increase in UNIDENTIFIED LEAKAGE exceeds the specified limit and the source of increased leakage cannot be isolated or it cannot be determined within a short period of time that the source of increased leakage is not associated with austenitic stainless steel.

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. 65 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 July 11, 1990, and supplemented May 7, 1991, Illinois Power Company (IP) and Soyland Power Cooperative, Inc. (the licensee) requested an amendment to Facility Operating License No. NPF-62 for the Clinton Power Station. The proposed amendment would revise the Technical Specifications (TS) by changing Section 4.0.5, Surveillance Requirements for Inservice Inspection and Testing; Section 3/4.4.3.1, Reactor Coolant System Leakage Detection Systems and Bases; and Section 3/4.4.3.2, Operational Leakage and Bases. The amendment would conform to the NRC staff position on Inservice Inspection (ISI) and monitoring of unidentified leakage, as delineated in Generic Letter (GL) 88-01 entitled "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) problems in Boiling Water Reactor (BWR) Austenitic Stainless Steel Piping."

GL 88-01, which was issued January 25, 1988, provides guidance regarding IGSCC problems in BWR piping made of austenitic stainless steel that is 4 inches or larger in nominal diameter. The GL only addresses problems where the piping contains reactor coolant above 200 degrees F during reactor power operation, regardless of ASME Code classification. Information was requested of affected licensees regarding conformance with the NRC positions stated in the GL. Two specific requests were: (1) a TS change to include a statement in the TS section on ISI certifying that the ISI program for piping covered by GL 88-01 will conform with the NRC positions on schedule, methods and personnel, and sample expansion; and (2) confirmation that the licensee's TSs related to leakage detection will be amended as appropriate to conform with the NRC positions included in the GL. These two requests are addressed by the proposed amendment.

Two other revisions requested by the July 11, 1990 letter, to TS Section 3/4.8.3.1, Onsite Power Distribution Systems - Operating, and Section 3/4.8.1.1, AC Sources - Operating, were issued by Amendment No. 45, dated September 17, 1990, and by Amendment No. 49, dated September 27, 1990, respectively.

2.0 EVALUATION

The NRC staff previously completed an evaluation of the licensee's programs to meet the 13 staff positions and other guidance provided in GL 88-01. By letter dated August 24, 1990, the licensee was informed that its programs were acceptable and satisfied the requirements of GL 88-01, with two exceptions. The first exception concerned the categorization of 50 welds. The staff has reviewed the licensee's April 12, 1991, response to this issue, and has found that its classification of welds is acceptable. The licensee was notified of this finding by letter dated November 15, 1991. The second exception involved the licensee's proposed TS changes as submitted by letter dated July 11, 1990. The licensee addressed the TS exception in supplemental correspondence dated May 7, 1991. The staff's evaluation of the proposed amendment, as supplemented, follows.

The licensee has proposed the following changes to the TSs which conform to the guidance provided in GL 88-01:

(1) Add new Surveillance Requirement 4.0.5.f to read "The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods and personnel, and sample expansion included in the generic letter."

(2) Modify the Action Statement in TS 3.4.3.1 to read "With only two of the above required leakage detection systems OPERABLE,

"a. operation may continue for up to 30 days when the drywell atmosphere particulate radioactivity monitoring system is inoperable provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours,

"b. operations may continue:

"1. with the drywell equipment drain sump flow monitoring subsystem inoperable provided the drywell equipment drain sump flow rate is monitored and determined by alternate means at least once per 12 hours.

"2. for up to 30 days with the drywell floor drain sump flow monitoring subsystem inoperable provided the drywell floor drain sump flow rate is monitored and determined by alternate means at least once per 8 hours.

"c. operation may continue for up to 30 days when the drywell atmosphere gaseous radioactivity monitoring system and the drywell air coolers condensate flow rate monitoring system are inoperable provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours.

"Otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."

(3) Add new Limiting Condition for Operation 3.4.3.2.e to read "No greater than a 2 gpm increase in UNIDENTIFIED LEAKAGE within a 24-hour period or less during OPERATIONAL CONDITION 1." In addition, add associated Action Statement 3.4.3.2.d to read "With any reactor coolant system

UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 24-hour period or less (during OPERATIONAL CONDITION 1), within 4 hours from the time of discovery isolate the source of increased leakage or verify that the source of increased leakage is not associated with service sensitive Type 304 or 316 austenitic stainless steel; otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."

(4) Renumber Surveillance Requirements 4.4.3.2.1.b, 4.4.3.2.1.c, and 4.4.3.2.1.d to become 4.4.3.2.1.c, 4.4.3.2.1.d, and 4.4.3.2.1.e, respectively; and, add Surveillance Requirement 4.4.3.2.1.b to read "Monitoring the drywell floor drain sump flow rate at least once per 8 hours." Delete from the renumbered 4.4.3.2.1.c the redundant words "...floor and...". The drywell floor drain sump flow rate surveillance is covered by the new 4.4.3.2.1.b.

(5) Make administrative change to TS Section 4.4.3.2.2.a removing a footnote that was applicable only until startup following the first refueling outage.

(6) Revise Bases Section 3/4.4.3.1 on Leakage Detection Systems to include the statement "With certain exceptions as noted in the Clinton Power Station Updated Safety Analysis Report, these detection systems are consistent with the recommendation of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. Except for the drywell particulate and gaseous radioactivity monitors, the systems provide the ability to measure leakage from fluid systems in the drywell. The drywell sump flow monitoring system consists of the drywell floor drain sump flow monitoring subsystem and the drywell equipment drain sump flow monitoring subsystem. OPERABILITY of each of these subsystems requires that the applicable portion of the monitoring subsystem associated with the v-notched weir box be OPERABLE. Other portions of the subsystem, including the sump pump control circuit and the associated timer, cycle counter and level switches, may be utilized as appropriate to provide an alternate means of monitoring and determining UNIDENTIFIED or IDENTIFIED leakage under the provisions of the associated ACTION statements for the respective subsystem."

(7) Revise Bases Section 3/4.4.3.2 on Operational Leakage to include the statements "With respect to IGSCC-related cracks in service sensitive austenitic stainless steel piping however, an additional limit on the allowed increase in UNIDENTIFIED LEAKAGE (within a 24-hour period or less) is imposed in accordance with Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," since an abrupt increase in the UNIDENTIFIED LEAKAGE could be indicative of leakage from such a source." and "The reactor will also be shut down if an increase in UNIDENTIFIED LEAKAGE exceeds the specified limit and the source of increased leakage cannot be isolated or it cannot be determined within a short period of time that the source of increased leakage is not associated with austenitic stainless steel."

These proposed changes are consistent with the model TSs provided in the GL 88-01 and are adequate for inservice inspection, monitoring, testing and surveillance of system leakage. The staff has, therefore, concluded that the TS changes satisfy the intent and objectives of GL 88-01 and are acceptable.

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 (55 FR 36345 and (57 FR 4488). 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.

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Date: August 21, 1992