March 14, 19[illegible]

Docket No. 50-461

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Mr. Richard F. Phares
Director - Licensing
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, Illinois 61727

Dear Mr. Phares:

SUBJECT: ISSUANCE OF AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. NPF-62 - CLINTON POWER STATION, UNIT 1 (TAC NO. M88869)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 89 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. The amendment is in response to your application dated February 25, 1994 (U-602257), as supplemented by letter dated March 11, 1994 (U-602265).

The amendment, issued pursuant to 10 CFR 50.91(a)(5), changes Technical Specification 3/4.4.3.1, "Reactor Coolant System Leakage - Leakage Detection Systems," to permit continued plant operation with inoperable drywell floor drain sump flow rate monitoring instrumentation. Continued plant operation is permitted until the first time the plant is required to be brought to COLD SHUTDOWN after March 15, 1994.

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 Douglas V. Pickett

Douglas V. Pickett, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 89 to NPF-62
2. Safety Evaluation

cc w/enclosures:
see next page

*SEE PREVIOUS CONCURRENCE

			BC:SPLB*	BC:EMCB*
			CMcCracken	JStrosnider
			03/07/94	03/07/94
LA:PD33:DRPW	PM:PD33:DRPW	D:PD33:DRPW*	OGC-OWFN*	AD:ADIII
MRushbrook	DPickett/dy	JHannon	EHoller	JZwolinski
3/11/94	3/11/94	3/11/94	03/10/94	3/11/94

DOCUMENT NAME: TAC88869.AMD

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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. 20555-0001

ILLINOIS POWER COMPANY, ET AL.

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
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 25, 1994, as supplemented by letter dated March 11, 1994, 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. 89 , 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 A. Zwolinski, Assistant Director
for Region III Reactors
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 14, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. NPF-62

DOCKET NO. 50-461

Replace the following page of the Appendix "A" Technical Specifications with the attached page. The revised page is identified by amendment number and contains vertical lines indicating the area of change. The corresponding overleaf page, indicated by an asterisk, is provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 4-12	3/4 4-12
3/4 4-12a*	3/4 4-12a*

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.

*Operation may continue after March 15, 1994, until the next COLD SHUTDOWN, provided the drywell floor drain sump flow rate is monitored and determined by alternate means at least once per 8 hours. Additionally, the drywell atmosphere particulate and gaseous radioactivity monitoring systems may be periodically taken out-of-service to perform scheduled preventive maintenance, surveillances and testing without entering the shutdown requirements of the ACTION statement.

REACTOR COOLANT SYSTEM

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 89 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

Technical Specification 3/4.4.3.1, "Reactor Coolant System Leakage - Leakage Detection Systems," requires that selected systems capable of monitoring and determining reactor coolant system leakage remain operable. Reactor coolant leakage is collected and classified as either identified or unidentified leakage. The Clinton Power Station has two drywell sump monitoring systems. The drywell equipment drain sump monitors identified leakage whereas the drywell floor drain sump monitors unidentified leakage.

Reactor coolant system leakage that falls on the drywell floors is channeled through the floor drains and enters the drywell floor drain sump. Prior to entering the floor drain sump, water passes through the drywell floor drain sump flow monitoring instrumentation where the instantaneous flow rates and total integrated flow are measured. The flow monitoring instrumentation consists of a V-notch weir box containing a capacitance probe. Water flows through a V-notch water level which is directly proportional to the flow through the weir box. Thus, flow through the V-notch is equal to the sump inlet flow rate. The capacitance probe is calibrated to correspond to the incoming flow rate and provides a continuous control room indication of the unidentified reactor coolant system leakage rate. An alarm is generated when the technical specification limit of 5 gpm of unidentified leakage occurs. The V-notch weir box instrumentation meets the accuracy and sensitivity requirements of Regulatory Guide 1.45 for drywell floor drain sump flow monitoring.

Once water enters the drywell floor drain sump, a system of pumps, pump-out timers, cycle counters and level switches monitors and records unidentified reactor coolant system leakage. Sump pump performance is monitored to provide control room indication if excessive leakage occurs. The sump pumps automatically start and stop at pre-determined levels. Pump running time is monitored and provides an alarm if run times exceed a given value which would be indicative of excessive leakage. In addition, the time between automatic pump startup between cycles is monitored. Frequent cycling of the sump pumps would also be indicative of excessive leakage thus generating an alarm. Finally, a high-high sump level alarm would be generated indicative that sump pump operation was not maintaining proper level. By knowing the sump volume, pump curve, pump running time, and the cycling time between automatic pump startup and shutdown, an alternative means can be used to verify overall leakage into the sump.

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Technical specifications limit the amount of unidentified reactor coolant system leakage to a total of 5 gpm. Technical specifications also limit any increase of unidentified leakage to 2 gpm within any 24-hour period. This latter value is in accordance with NRC Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping," since an abrupt increase in unidentified leakage rate could be indicative of a leak before break in stainless steel piping.

In early February of 1994, control room operators observed fluctuating leakage rates sensed by the V-notch weir box measuring unidentified reactor coolant system leakage. Using the alternative means described above to verify the unidentified leakage rate, control room operators were able to verify that actual leakage increases had not occurred. Subsequently on February 13, 1994, the drywell floor drain sump flow monitoring instrumentation was declared inoperable. Technical Specification 3/4.4.3.1 permits continued plant operation for 30 days provided an alternative means is used to monitor and determine unidentified leakage rates once every 8 hours, and the remaining leakage detection systems are operable (i.e., the drywell atmosphere particulate monitor and either the drywell atmosphere gaseous monitor or the drywell air cooler condensate flow rate monitoring system).

Efforts by the licensee to restore the drywell floor drain sump monitoring instrumentation have been unsuccessful. The instrument loop has been recalibrated and equipment external to the drywell has been verified to be operating properly. In addition, the licensee has attempted to "backflush" the V-notch weir box by temporarily suspending sump pump operation to permit water to back up and dislodge any foreign material that may be blocking the V-notch. Having exhausted all efforts to trouble-shoot from outside the drywell, the remaining alternative is to make a drywell entry to make a physical examination of the V-notch weir box and the capacitance probe. However, the location of the V-notch weir box is within the biological shield wall and directly below the reactor vessel. Due to high radiation and temperature concerns, a plant shutdown would be required to permit personnel entry.

By letter dated February 25, 1994, Illinois Power Company requested an emergency technical specification change pursuant to 10 CFR 50.91(a)(5). The change would permit continued plant operation with the inoperable drywell floor drain sump monitoring instrumentation provided an alternate means was being used to monitor, and determine unidentified reactor coolant leakage rates, once every 8 hours. Operation in this mode is being requested until the first time the plant is required to be brought to COLD SHUTDOWN.

2.0 EVALUATION

Technical Specification 3/4.4.3.1 requires that multiple reactor coolant system leakage detection systems remain operable. Item 3.4.3.1.a requires operation of a drywell atmosphere particulate radioactivity monitoring system. This would provide early indication of fission product release. Item 3.4.3.1.b requires the drywell sump flow monitoring system to remain operable. The drywell sump flow monitoring system consists of both the drywell floor drain sump flow monitoring instrumentation previously discussed and the physically identical drywell equipment drain sump flow monitoring instrumentation used to monitor identified reactor coolant system leakage. Finally, item 3.4.3.1.c requires operation of either the drywell atmosphere gaseous radioactivity monitoring system or the drywell air coolers condensate flow rate monitoring system. These latter items would provide early indication of fission gas release or abnormal steam conditions in the drywell.

The licensee proposes to monitor and determine the unidentified reactor coolant system leakage rate using an alternate means once every 8 hours as currently required in the technical specification action statement. The alternate means would be through verification of sump pump performance and confirming that the integrated sump pump flow rates would not exceed technical specification limits. Current plant conditions are showing that the sump pumps are cycling approximately once every five hours. The pump run times of one to one and a half minutes correspond to a calculated unidentified leakage rate of 0.2 to 0.4 gallons per minute.

The licensee states that the accident analysis is unaffected by the proposed changes. The design basis accident involving leakage into the drywell is a guillotine break of the recirculation system suction piping. Safety systems accident mitigation is automatically initiated in response to high drywell pressure or low reactor vessel water level. Regarding small break loss-of-coolant accidents, the Updated Safety Analysis Report section 7.7.1.24.1 states that no credit is taken in the safety analysis for operation of or operator reliance upon the leakage detection monitoring instrumentation associated with the drywell sump. As previously discussed, control room operators will monitor and determine unidentified reactor coolant system leakage once every 8 hours. In addition, alternate indications via the drywell particulate radioactivity monitoring system, and either the drywell atmosphere gaseous radioactivity monitoring system or the drywell air coolers condensate flow rate monitoring system would be available to inform control room operators of abnormal conditions. The staff concurs that an adequate alternative means exists to monitor and determine unidentified reactor coolant system leakage.

The licensee's letter proposes that continued plant operation be permitted until the first time that the reactor is brought to COLD SHUTDOWN after March 15, 1994. The March 15, 1994 date, is the end of the current 30 day action statement of the technical specifications, and the staff concurs that this is the appropriate start date. The licensee's basis for choosing COLD SHUTDOWN as opposed to HOT SHUTDOWN is due to the extreme high radiation and temperature levels in the drywell. While the staff agrees that high temperatures are a major factor, the radiation concerns would not be expected to change from HOT SHUTDOWN to COLD SHUTDOWN and thus, should not be a factor.

The V-notch weir box is located in a keyway beneath the reactor vessel. Not only would this represent a high radiation level during plant operations, but it is also a highly contaminated area. Personnel entering this area will need to wear respirators and double plastic anti-contamination suits. Compounding the difficulty in working under such conditions are the anticipated temperatures. Normal drywell ventilation systems are not particularly effective in this location. The licensee anticipates that the temperatures in this region during HOT SHUTDOWN conditions would approach 140 °F. Personnel entering under these conditions would be required to wear ice packs, would need to be monitored for heat stress, and would be limited to approximately 20 minute stay times. Cooling the plant to COLD SHUTDOWN would result in primary coolant system temperatures of less than 200 °F. While the licensee did not provide any quantitative assessment on the amount of additional cooling that COLD SHUTDOWN conditions would provide to the drywell region, the staff agrees that conditions would be more tolerable to personnel entry. Viewed from a personnel safety aspect, the licensee considers COLD SHUTDOWN a more appropriate entry condition. The staff concurs with the licensee that repairs should not be required until the first time that the facility is brought to COLD SHUTDOWN.

The footnote as originally proposed by the licensee would be placed in the LIMITING CONDITION FOR OPERATION as opposed to the ACTION STATEMENT. The proposed footnote states:

> "In lieu of the requirement for the associated V-notched weir box to be OPERABLE, the drywell floor drain sump flow monitoring system may be considered OPERABLE provided the drywell floor drain sump flow is monitored and determined by alternate means at least once per 8 hours. This provision is applicable until the first time the reactor is brought to COLD SHUTDOWN after March 15, 1994."

The licensee will effectively be performing the action statement requirement for inoperable sump flow monitoring instrumentation, but will not be tracking an action statement. As stated in technical specification 3.4.3.1, if more than one of the reactor coolant system leakage detection systems is inoperable, the facility must be brought to HOT SHUTDOWN within 12 hours. By not tracking an action statement associated with the sump monitoring instrumentation, the licensee will avoid an immediate shutdown requirement if problems were experienced with the particulate monitor, or either the gaseous monitor, or the drywell air cooler condensate flow rate monitoring system.

The staff has considered this proposal and observes that an inoperable piece of equipment will be treated as if it were, in fact, operable. However, the licensee has made this proposal to avoid periodic entries into the 12-hour shutdown statement of technical specification 3.4.3.1. Discussions with the licensee indicate that both radiation monitors are periodically removed from service. The particulate monitor needs to be taken out of service, approximately once every other week to change the filter paper. In addition, both the particulate and gaseous monitors are taken out of service monthly for channel functional testing. While each of these activities typically requires less than an hour to perform, they would result in the facility being placed in an immediate shutdown condition. The staff does not believe that entering a plant shutdown to perform routine surveillances and testing would be consistent with the safety significance. This is supported by the technical specification action statement which permits continued plant operation for 30 days with any one of these components inoperable.

The staff did not agree with the licensee's proposal which would effectively consider a piece of inoperable equipment to be operable provided that the compensatory measures associated with the <u>ACTION</u> statement were being complied with. Discussions with the licensee developed an alternative wording that better acknowledged the inoperable condition of the instrumentation but would still permit continued operations as sought by the licensee. The revised footnote would read as follows:

> "Operation may continue after March 15, 1994, until the next COLD SHUTDOWN, provided the drywell floor drain sump flow rate is monitored and determined by alternate means at least once per 8 hours. Additionally, the drywell atmosphere particulate and gaseous radioactivity monitoring systems may be periodically taken out-of-service to perform scheduled preventive maintenance, surveillances and testing without entering the shutdown requirements of the <u>ACTION</u> statement."

By letter dated March 11, 1994, the licensee supplemented its application to agree with this revised wording of the footnote. In addition, the licensee agreed to place this footnote in the <u>ACTION</u> statement of the technical specification as opposed to the <u>LIMITING CONDITION FOR OPERATION</u>. The staff believes that this is the preferred location for the footnote.

The staff has reviewed the licensee's proposal for an emergency technical specification change to permit continued plant operation until the first time that the reactor is brought to COLD SHUTDOWN after March 15, 1994. Considering the alternate means of monitoring and determining unidentified reactor coolant system leakage available to the licensee, the relatively low safety significance of operating in this condition, and the desire to avoid any unnecessary plant shutdown and resultant risks, the staff finds the licensee's proposal acceptable.

3.0 <u>EMERGENCY CIRCUMSTANCES</u>

The licensee declared the drywell floor drain sump monitoring instrumentation inoperable on February 13, 1994. With the remaining reactor coolant system leakage detection systems operable, Technical Specification 3.4.3.1 permits 30 days of continuous plant operation provided the drywell floor drain sump flow rate is monitored and determined by alternative means at least once every 8 hours.

All efforts by the licensee to restore the drywell sump inlet flow monitoring instrumentation to operable status have been unsuccessful. The instrument loop has been recalibrated and equipment external to the drywell has been verified to be operating properly. The only option remaining for the licensee is to enter the drywell in order to examine the V-notch weir box and associated capacitance probe. However, the V-notch weir box is located in a keyway beneath the reactor vessel and inside the biological shield wall. Due to the high radiation and temperatures in this location, a plant shutdown would be required before personnel would be able to reach the instrumentation.

The staff does not consider loss of this instrumentation, by itself, to be safety significant. The 30 day action statement found in the technical specifications further supports this view. While the control room will not be capable of continuously monitoring unidentified leakage flow rates, the alternative means performed at least once every 8 hours should be sufficient to provide ample warning of any unanticipated crack in primary system piping.

In a letter dated February 25, 1994, the licensee requested that this amendment application be treated as an emergency because unless approved, the technical specifications would require a plant shutdown. The licensee stated that such action would be necessary to preclude an unnecessary plant transient and related plant risk associated with a plant shutdown. Due to time constraints, sufficient time was not available to permit the customary public notices in advance of this action.

Accordingly, pursuant to 10 CFR 50.91(a)(5), the Commission has determined that an emergency situation exists in that failure to act in a timely way will result in a plant shutdown. Further, the Commission has determined that the emergency is not due to the failure of the licensee to act in a timely manner.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. The Commission has made a final determination that the amendment does not involve a significant hazards consideration because:

1. The proposed change would permit continued plant operation with inoperable drywell floor drain sump monitoring instrumentation. This instrumentation does not provide any accident mitigation function nor is it relied on for operator action. The instrumentation is only one of several means of providing indication to control room operators of unidentified reactor coolant system leakage rates.

Control room operators will monitor and determine unidentified reactor coolant system leakage rates using an alternative means of monitoring the drywell floor drains sump pumps. By monitoring sump pump operating times, frequency of pump cycling, and level switches, operators will verify that unidentified reactor coolant system leakage rates remain within acceptable levels. In addition, the availability of particulate and gaseous radioactivity monitors and observation of the condensate discharge line flow rates from the drywell air coolers, will provide operators with indirect indication of any unanticipated increase in unidentified leakage.

Permitting continued plant operation until the first COLD SHUTDOWN after March 15, 1994, will avoid an unnecessary plant shutdown and resultant risk. Since the instrumentation is only used to provide indication and no credit is taken in the safety analysis for operation of or operator reliance on this instrumentation, the staff concludes that the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. The proposed change does not involve a change in the operation of the plant, nor does it introduce any new failure modes. This instrumentation does not provide any accident mitigation function nor is it relied on for operator action. Control room operators will use alternate means to periodically verify unidentified reactor coolant system leakage rates and will possess indirect means of observing increases in leakage rates via the particulate and gaseous monitors and observation of the condensate discharge line flow rates from the drywell air coolers. Therefore, the staff concludes that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The margin of safety associated with this proposed change relates to the limits on unidentified reactor coolant system leakage. As discussed in the Bases for Technical Specification 3/4.4.3.2, the allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for the unidentified leakage limits, the probability is small that the imperfection or crack associated with such leakage would grow rapidly.

   The V-notch weir box normally provides continuous control room indication of the unidentified leakage rate of the reactor coolant system. With this instrumentation inoperable, the licensee has proposed to monitor and determine the leakage rate through an alternative means once every 8 hours. Since the probability of a small imperfection or crack to grow rapidly is small, verification of leakage once every 8 hours should be sufficient.

NRC Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping," implemented more stringent limits of unidentified leakage. The generic letter imposed a limit of a 2 gpm increase in any 24-hour period since an abrupt increase could be indicative of a crack in service sensitive austenitic stainless steel piping. The proposed change does not alter any previously set limits on unidentified leakage.

Based on the above, the staff concludes that the proposed change does not involve a significant reduction in a margin of safety.

5.0 STATE CONSULTATION

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

6.0 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 NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The staff has made a final determination that this amendment involves no significant hazards consideration. 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.

7.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Douglas V. Pickett

Date: March 14, 1994