July 15, 1993

Docket No. 50-461

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R. Laufer
R. Stransky
E. Baker

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 (TAC NO. M86268)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 81 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. The amendment is in response to your application dated April 16, 1993 (U-602115)

The amendment changes Clinton Power Station Technical Specifications (TS) 3/4.1.3.2, "Control Rod Maximum Scram Insertion Times," 3/4.4.2.1, "Safety/Relief Valves," and 3/4.5.1, "Emergency Core Cooling Systems." The changes reduce, or mitigate, certain time restrictions associated with surveillance testing required during plant startup. As discussed with your staff, we have modified proposed TS 3/4.1.3.3 to permit alternate control rod testing at a reactor coolant system pressure "greater than or equal to" 950 psig as opposed to simply "greater than". In addition, we have modified the footnotes to proposed TS 3/4.4.2.1.1 and 3/4.5.1 to include both "reactor steam pressure and flow" as opposed to "reactor steam conditions" to better define the surveillance interval.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly <u>Federal Register</u> notice.

Sincerely,

Original Signed By:
Douglas V. Pickett, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 81 to NPF-62
2. Safety Evaluation

cc w/enclosures:
see next page

OFC	LA:PDIII-2	PE:PDIII-2	PM:PDIII-2	BC:OTSB	BC:SRXB	D:PDIII-2	OGC
NAME	CMOORE	RLAUFER	DPICKETT	CGRIMES	RJONES	JDYER	
DATE	6/23/93	/ /93	6/23/93	7/1/93	6/23/93	7/15/93	7/8/93

DF01

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

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Mr. Richard F. Phares
Illinois Power Company

Clinton Power Station
Unit No. 1

cc:

Mr. J. S. Perry
Senior Vice President
Clinton Power Station
Post Office Box 678
Clinton, Illinois 61727

Mr. J. A. Miller
Manager Nuclear Station
Engineering Department
Clinton Power Station
Post Office Box 678
Clinton, Illinois 61727

Sheldon Zabel, Esquire
Schiff, Hardin & Waite
7200 Sears Tower
233 Wacker Drive
Chicago, Illinois 60606

Resident Inspector
U.S. Nuclear Regulatory Commission
RR#3, Box 229 A
Clinton, Illinois 61727

Ms. K. K. Berry
Licensing Services Manager
General Electric Company
175 Curtner Avenue, M/C 382
San Jose, California 95125

Regional Administrator, Region III
799 Roosevelt Road, Building 4
Glen Ellyn, Illinois 60137

Chairman of DeWitt County
c/o County Clerk's Office
DeWitt County Courthouse
Clinton, Illinois 61727

Mr. Robert Neumann
Office of Public Counsel
State of Illinois Center
100 W. Randolph, Suite 11-300
Chicago, Illinois 60601

Illinois Department
of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Mr. Donald Schopfer
Project Manager
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

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

ILLINOIS POWER COMPANY

SOYLAND POWER COOPERATIVE, INC.

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 81
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 April 16, 1993, 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. 81 , 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 15, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 81

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, identified by an asterisk, are provided to maintain document completeness.

Remove Pages	Insert Pages
*3/4 1-7	*3/4 1-7
3/4 1-8	3/4 1-8
*3/4 4-9	*3/4 4-9
3/4 4-10	3/4 4-10
3/4 5-4	3/4 5-4
3/4 5-5	3/4 5-5
3/4 5-6	3/4 5-6
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4
* B 3/4 5-1	* B 3/4 5-1
B 3/4 5-2	B 3/4 5-2
B 3/4 5-3	B 3/4 5-3

LIMITING CONDITION FOR OPERATION (Continued)

3.1.3.2 ACTION (Continued):

3. The sum of "fast" control rods with individual scram insertion times in excess of the limits of ACTION a.2 and of "slow" control rods does not exceed 5.

4. No "slow" control rod, "fast" control rod with individual scram insertion time in excess of the limits of ACTION a.2, or otherwise inoperable control rod occupy adjacent locations in any direction, including the diagonal, to another such control rod.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. With a "slow" control rod(s) not satisfying ACTION a.1, above:

1. Declare the "slow" control rod(s) inoperable and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more "slow" control rods declared inoperable.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

c. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.c, operation may continue provided that:

1. "Slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, do not make up more than 20% of the 10% sample of control rods tested.

2. Each of these "slow" control rods satisfies the limits of ACTION a.1.

3. The eight adjacent control rods surrounding each "slow" control rod are:

   a) Demonstrated through measurement within 12 hours to satisfy the maximum scram insertion time limits of Specification 3.1.3.2 and

   b) OPERABLE.

4. The total number of "slow" control rods as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3 as determined by Specification 4.1.3.2.a and b, does not exceed 5.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

4.1.3.3 The maximum scram insertion time for specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig.* Alternatively, those specific control rods may be determined OPERABLE with reactor coolant pressure less than 950 psig by demonstrating an acceptable scram insertion time to notch position 13. The scram time acceptance criteria for this alternate test shall be determined by linear interpolation between 0.95 seconds at a reactor coolant pressure of 0 psig and 1.40 seconds at 950 psig. If this alternate test is utilized, the individual scram time shall also be measured with reactor coolant pressure greater than or equal to 950 psig prior to exceeding 40% of RATED THERMAL POWER. For each of the above single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators.

*The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided this surveillance requirement is completed prior to entry into OPERATIONAL CONDITION 1.

3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least six of the following valves and the relief valve function of at least five additional valves, other than those satisfying the safety valve function requirement, shall be OPERABLE with the specified lift settings; and the acoustic monitor for each OPERABLE valve shall be OPERABLE.*

Number of Valves	Function	Setpoint** (psig)
7	Safety	1165 ± 11.6 psi
5	Safety	1180 ± 11.8 psi
4	Safety	1190 ± 11.9 psi
1	Relief	1103 ± 15.0 psi
8	Relief	1113 ± 15.0 psi
7	Relief	1123 ± 15.0 psi

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety and/or relief valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 110°F, close the stuck open safety/relief valve(s); if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitor(s) inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With either relief valve 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.

---

* One relief valve pressure actuation channel and/or one acoustic monitor channel may be placed in an inoperable status for up to 6 hours for the purpose of performing surveillance testing in accordance with Specifications 4.4.2.1.1 and 4.4.2.1.2.

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

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months.*

4.4.2.1.2 The relief valve 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 92 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 relief valve 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.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow are adequate to perform the test.

SURVEILLANCE REQUIREMENTS (Continued)

4.5.1 (Continued)

2. Verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

b. Verifying that when tested pursuant to Specification 4.0.5 each:

1. LPCS pump develops a flow of at least 5010 gpm with a pump differential pressure greater than or equal to 276 psid.

2. LPCI pump develops a flow of at least 5050 gpm with a pump differential pressure greater than or equal to 113 psid.

3. HPCS pump develops a flow of at least 5010 gpm with a pump differential pressure greater than or equal to 363 psid.

c. For the LPCS, LPCI, and HPCS systems, at least once per 18 months performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

d. For the HPCS system, at least once per 18 months, verifying that the suction is automatically transferred from the RCIC storage tank to the suppression pool on a RCIC storage tank low water level signal and on a suppression pool high water level signal.

e. For the ADS by:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator low pressure alarm system.

2. At least once per 18 months, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

3. At least once per 18 months, manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig** and observing that:

a. The control valve or bypass valve position responds accordingly, or

---

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow are adequate to perform the test.

LIMITING CONDITION FOR OPERATION

b. There is a corresponding change in the measured steam flow, or

c. The acoustic tail-pipe monitor alarms.

4. At least once per 18 months, performing a CHANNEL CALIBRATION of the accumulator low pressure alarm system and verifying an alarm setpoint of ≥ 140 psig on decreasing pressure.

3.5.2 At least two of the following shall be OPERABLE and capable of being powered from a diesel generator of Specification 3.8.1.2.b.

a. The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.

b. Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

c. Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

d. Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

1. From the suppression pool, or

2. When the suppression pool level is less than the limit or is drained, from the RCIC storage tank containing at least 125,000 available gallons of water.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel. The provisions of Specification 3.0.4 are not applicable.

* The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the reactor cavity to steam dryer pool gate is open and water level in these upper containment pools is maintained within the limits of Specification 3.9.8 and 3.9.9.

LIMITING CONDITION FOR OPERATION

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish PRIMARY CONTAINMENT INTEGRITY within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.

4.5.2.2 The HPCS system shall be determined OPERABLE at least once per 12 hours by verifying the RCIC storage tank required volume when the RCIC storage tank is required to be OPERABLE per Specification 3.5.2.e.

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 surveillance requirement for performing a CHANNEL CALIBRATION of the acoustic monitor(s) includes an exception to the provisions of Specification 4.0.4. This exception allows reactor steam conditions to be established which are adequate to open the SRVs without resulting in unnecessary wear on the valves and to ensure that proper reactor pressure control can be maintained while opening and reclosing the valves. Reactor steam conditions which are considered adequate to perform the test thus include the establishment of sufficient reactor pressure as well as sufficient steam flow to ensure that the steam relieved by the SRVs can be compensated by the reactor pressure control system.

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 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. 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.

3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 AND 3/4.5.2 ECCS - OPERATING AND SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "1". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "2".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled folowing a loss-of-coolant accident and, together with the LPCI system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The LPCI system, together with the LPCS system, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1177 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING AND SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 467/1400/5010 gpm at differential pressures of 1177/1147/200 psid. Initially, water from the reactor core isolation cooling (RCIC) tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the RCIC tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days, as specified in the corresponding ACTION statement, is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety-relief valves although the safety analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

The surveillance requirements for the ADS include a requirement to manually open each ADS valve. This requirement includes an exception to the provisions of Specification 4.0.4. This exception allows reactor steam conditions to be established which are adequate to open the ADS valves without resulting in unnecessary wear on the valves and to ensure that proper reactor pressure control can be maintained while opening and reclosing the valves. Reactor steam conditions which are considered adequate to perform the test thus include the establishment of sufficient reactor pressure as well as sufficient steam flow to ensure that the steam relieved by the ADS valves can be compensated by the reactor pressure control system.

3/4.5.3 SUPPRESSION POOL

The suppression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITIONS 4 and 5 the suppression pool minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume and vortex prevention plus a safety margin of 2'4" for conservatism.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-62

ILLINOIS POWER COMPANY

SOYLAND POWER COOPERATIVE, INC.

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1.0 INTRODUCTION

By letter dated April 16, 1993, the Illinois Power Company (IP, the licensee), requested an amendment to Facility Operating License No. NPF-62 for the Clinton Power Station (CPS). The proposed amendment would change Clinton Power Station Technical Specifications (TS) 3/4.1.3.2, "Control Rod Maximum Scram Insertion Times," 3/4.4.2.1, "Safety/Relief Valves," and 3/4.5.1, "Emergency Core Cooling Systems." The proposed changes would reduce, or mitigate, certain time restrictions associated with surveillance testing required during plant startup.

2.0 EVALUATION

Control Rod Scram Time Testing

Surveillance Requirement (SR) 4.1.3.2.b currently requires that the maximum scram insertion time of specifically affected individual control rods be demonstrated, through measurement, following maintenance on, or modification to, the control rod or control rod drive system which could affect the scram insertion time of those specified rods. SR 4.1.3.2.b further requires that these scram insertion time tests be conducted with reactor coolant pressure greater than or equal to 950 psig and that during single control rod scram time tests the control rod drive pumps be isolated from the accumulators. SR 4.1.3.2.b also contains a footnote "*" stating, "The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided this surveillance requirement is completed prior to entry into OPERATIONAL CONDITION 1."

In their submittal the licensee points out the drawbacks of performing scram time testing during this time frame. The testing can not be performed until reactor pressure reaches 950 psig which also is the criteria for starting the 12-hour time clock for testing the Automatic Depressurization System (ADS) Safety/Relief Valves (SRV) and performing SRV acoustic monitor calibrations. Each scram time test takes approximately 30 minutes. If there are a large number of affected control rods to be tested, the reactor operators are under considerable pressure to complete the scram time tests in time to establish the necessary plant conditions for performing the SRV testing and to subsequently complete the SRV testing within the time limits. In addition, if a scram time test should fail, eight additional control rods may be required to be tested for each control rod that fails, creating additional time constraints.

In order to avoid the drawbacks discussed above, the licensee's proposal gives more flexibility to the control rod scram time testing requirements. The current exception to Specification 4.0.4 is an option the licensee chose to retain in the TS for cases which involve a small number of affected control rods to be tested. For a larger number of affected control rods to be tested, however, the licensee added an option of performing the scram time tests outside the 12-hour time limit during plant startup.

Specifically, the licensee's proposal deletes the current SR 4.1.3.2.b, reletters the current SR 4.1.3.2.c to become SR 4.1.3.2.b and creates a new SR 4.1.3.3. The new SR 4.1.3.3 incorporates the current requirements of SR 4.1.3.2.b and adds an alternative that "those specific control rods may be determined OPERABLE with reactor coolant pressure less than 950 psig by demonstrating an acceptable scram insertion time to notch position 13. The scram time acceptance criteria for this alternate test shall be determined by linear interpolation between 0.95 seconds at a reactor coolant pressure of 0 psig and 1.40 seconds at 950 psig. If this alternate test is utilized, the individual scram time shall also be measured with reactor coolant pressure greater than or equal to 950 psig prior to exceeding 40% of RATED THERMAL POWER." (Based on the licensee's request, the staff inserted the words "or equal to" into the last sentence.)

The value of 0.95 seconds for the scram time test acceptance criteria at 0 psig for notch position 13 was derived from the startup test program as reflected in CPS Updated Safety Analysis Report (USAR) Section 14.2.12.2.5, "Control Rod Drive System," Figure 14.2.8. The value of 1.40 seconds for the scram time acceptance criteria at 950 psig for notch position 13 is the most conservative maximum scram insertion limit provided in the current TS 3.1.3.2 (i.e., the scram insertion time limits provided for "fast" control rods in ACTION a.). Consequently, linear interpolation between these values provides adequate assurance that the control rod scram time has not been adversely affected by performance of the subject maintenance or modification.

Performing the test while the reactor vessel is at 0 psig will provide assurance of satisfactory control rod scram insertion capability prior to restarting the plant. This option of testing the affected control rods will allow the plant to enter Operational Conditions 1 and 2 with the affected control rods considered operable. Under this condition, the affected control rods will also be required to be tested again prior to 40% power along with the remaining control rods per TS 4.1.3.2.a. Although the proposed option will thus require the scram time test(s) to be performed twice for the affected control rod(s), this option will alleviate some of the pressure on the reactor operators to perform the testing under such unnecessarily restrictive time limits.

The proposed revisions to SR 4.1.3.2 and the addition of SR 4.1.3.3 are consistent with changes that have been previously reviewed and approved by the staff. The requirements proposed by the licensee are similar to that found acceptable to the staff in the Improved Standard Technical Specifications for BWR/6 facilities. Therefore, the staff finds the proposed changes acceptable.

Safety Relief Valve and Acoustic Monitor Testing

Surveillance Requirement 4.4.2.1.1.b requires, "The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE by performance of a CHANNEL CALIBRATION at least once per 18 months*." Surveillance Requirement 4.5.1.e requires in part, "At least once per 18 months, manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig**..." The footnote for both of these SRs states, "The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test."

The reactor steam pressure "adequate to perform the test" is specified by the SRV manufacturer to be 950 psig. Thus, the allotted 12-hour time period for completing the required testing begins when reactor pressure reaches 950 psig.

In addition to reactor pressure, steam flow is an important parameter for maintaining reactor pressure control during stroking of the SRVs. Because opening SRVs can result in a reactor pressure transient which initially introduces negative reactivity, and subsequent SRV closure can result in the introduction of positive reactivity (through steam void collapse), two conditions must first be met before a stroke test of the SRVs is performed: (1) adequate reactor pressure must exist to protect the SRV from damage when stroking, and (2) sufficient steam flow must exist to open at least 1-1/2 main turbine steam bypass valves (which occurs at approximately 16% power). The latter is required because when an SRV is opened with less that 1-1/2 bypass valves open, a slow depressurization of the reactor vessel will occur after the bypass valves fully shut in an attempt to control pressure. This slow cooldown combined with the subsequent SRV closure can result in a pressure and corresponding reactor power spike and reactor scram.

As noted previously, per the current TS, the 12-hour time clock for performing stroke testing of the SRVs starts once adequate reactor pressure (950 psig) is reached. However, as explained above, the tests on the SRVs can not be performed until the required steam flow conditions are also achieved. A period of time is required after reactor pressure reaches 950 psig to establish adequate steam flow. Part of the allotted 12-hour test period is, thus, expended to establish the appropriate test conditions before stroke testing can actually begin.

The SRVs must also be stroked to complete the channel calibrations for the associated acoustic monitors. Specifically, acoustic monitor gain adjustments are performed under steam-flow conditions for each SRV. Performance of the SRV acoustic monitor gain adjustment requires each of the SRVs to be lifted at least twice, once to verify actuation and initial response of the monitor associated with each SRV and at least once to check the effect of the gain adjustment and verify that monitors on adjacent SRVs do not actuate. This procedure consumes much of the test time allotted by the current Specification 4.0.4 exception.

To alleviate unnecessary stress and time pressures on the reactor operators during reactor startups, CPS plans to revise the existing exceptions to Specification 4.0.4 within TS 4.5.1 and TS 4.4.2.1.1. The licensee proposed to revise the current wording from "within 12 hours after reactor steam pressure is adequate to perform the test" to "within 12 hours after reactor steam conditions are adequate to perform the test." However, after discussions with the the staff, the licensee agreed to modify the above wording to "within 12 hours after reactor steam pressure and flow are adequate to perform the test." This would better define the start of the surveillance interval. The licensee believes the proposed change is consistent with the original intent of the TSs to permit testing to be completed within a reasonable time once adequate test conditions are achieved after entering the applicable operating conditions.

Under the proposed change, the point in time during plant startup when SRV/acoustic monitor testing can begin would remain unchanged. That is, the current practice at CPS of ensuring that adequate steam pressure and flow both exist prior to commencing such testing would be maintained. The testing itself would continue to be performed within the required 12-hour time period. The intent of the proposed change is simply to change or clarify the point when the 12-hour clock must begin.

The staff concurs with the licensee in that the safety impact of the proposed change is negligible. As discussed above, the proposed change in the TS merely reflects current practice at the Clinton Power Station and does not change the time when testing is performed. The safety (spring) mode of operation of the SRVs is not affected by the proposed change as this function is independent of the relief mode or ADS function. In addition, the relief mode of the SRVs also would not be impacted as the SRV pressure relief instrumentation and ADS instrumentation is fully tested prior to reactor startup. As a result, the ADS valves would be expected to remain operable for reactor pressure control. With respect to the impact of the proposed change on the operability and testing of the acoustic monitors, the acoustic monitors merely provide indication that an SRV is open and do not provide any actuation signals. In addition, there would be other indications that an SRV is open such as reactor water level, reactor pressure, and main turbine bypass valve position.

The licensee has submitted these proposed changes in order to remove unnecessary time constraints during reactor startups and, thus, reduce stress on control room operators. The proposed changes will still meet the requirement to perform the SRV/acoustic monitoring and ADS valve testing within a 12-hour time span during plant startup. Therefore, based on the staff's review, the proposed changes are found acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes surveillance requirements. 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 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 (58 FR 30196). 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 Contributor: R. Laufer

Date: July 15, 1993