February 29, 1996

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Mr. Michael W. Lyon
Director - Licensing
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, IL 61727

SUBJECT: ISSUANCE OF AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. NPF-62 - CLINTON POWER STATION, UNIT 1 (TAC NO. M93992)

Dear Mr. Lyon:

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 102 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. The amendment is in response to your application dated October 27, 1995 (U-602498).

The amendment revises Technical Specification (TS) 3.1.3, "Control Rod OPERABILITY," to include the 25% surveillance overrun allowed by Limiting Condition for Operation (LCO) 3.0.2 into the allowances of the surveillance Notes for control rod "notch" testing per Surveillance Requirement (SR) 3.1.3.2 and SR 3.1.3.3. The amendment also includes a clarification to the description of TS Table 3.3.3.1-1, "Post Accident Monitoring Instrumentation," Function 7, to indicate that the Function's requirements apply to the position indication for only automatic primary containment isolation valves, rather than all primary containment isolation valves. Finally, the amendment includes changes to correct a number of editorial and typographical errors inadvertently contained in TS 3.3.4.1, "End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation," TS 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," TS 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring," and TS 3.6.5.2, "Drywell Air Lock."

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

Docket No. 50-461

Enclosures: 1. Amendment No. 102 to NPF-62
2. Safety Evaluation

*See Previously Concurrence

cc w/encls: See next page
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 29, 1996

Mr. Michael W. Lyon
Director - Licensing
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, IL 61727

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Sincerely,

Douglas V Pickett

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

Docket No. 50-461

Enclosures: 1. Amendment No. 102 to NPF-62
2. Safety Evaluation

cc w/encls: See next page

Mr. Michael W. Lyon
Illinois Power Company

Clinton Power Station
Unit No. 1

cc:

Mr. Wilfred Connell
Vice President
Clinton Power Station
Post Office Box 678
Clinton, Illinois 61727

Mr. Daniel P. Thompson
Manager Nuclear Station
Engineering Department
Clinton Power Station
Post Office Box 678
Clinton, Illinois 61727

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

Mr. R. T. Hill
Licensing Services Manager
General Electric Company
175 Curtner Avenue, M/C 481
San Jose, California 95125

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

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

Mr. J. W. Blattner
Project Manager
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

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

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. 102
License No. NPF-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Illinois Power Company* (IP), and Soyland Power Cooperative, Inc. (the licensees) dated October 27, 1995, 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. 102 , 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

Douglas V Pickett

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

Attachment: Changes to the Technical Specifications

Date of Issuance: February 29, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 102

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.

Remove Pages	Insert Pages
3.1-9	3.1-9
3.1-10	3.1-10
3.3-22	3.3-22
3.3-27	3.3-27
3.3-55	3.3-55
3.3-56	3.3-56
3.3-57	3.3-57
3.3-58	3.3-58
3.3-59	3.3-59
3.3-60	3.3-60
3.3-81	3.3-81
3.6-56	3.6-56

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	-------------------NOTE------------------- Not required to be performed until 8 days 18 hours after the control rod is fully withdrawn and THERMAL POWER is greater than the LPSP of the RPCS. -- Insert each fully withdrawn control rod at least one notch.	 7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.3	-------------------NOTE------------------- Not required to be performed until 38 days 18 hours after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RPCS. -- Insert each partially withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 13 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4
SR 3.1.3.5	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position AND Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1
1. Reactor Steam Dome Pressure	2	E
2. Reactor Vessel Water Level	2	E
3. Suppression Pool Water Level		
a. High Range	2	E
b. Low Range	2	E
4. Drywell Pressure	2	E
5. Primary Containment Area Radiation	2	F
6. Drywell Area Radiation	2	F
7. Penetration Flow Path, Automatic PCIV Position	2 per penetration flow path [a][b]	E
8. Drywell and Containment H_2 & O_2 Analyzer	2	E
9. Primary Containment Pressure		
a. High Range	2	E
b. Low Range	2	E
10. Suppression Pool Quadrant Water Temperature	2[c]	E

(a) Not required for isolation valves whose associated penetration flow path is isolated.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) Monitoring each quadrant.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.4.1.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. TSV Closure: ≤ 7% closed; and b. TCV Fast Closure, Trip Oil Pressure—Low: ≥ 465 psig.	18 months
SR 3.3.4.1.3	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	18 months
SR 3.3.4.1.4	Verify TSV Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 40% RTP.	18 months
SR 3.3.4.1.5	------------------NOTES------------------ 1. Breaker interruption time may be assumed from the most recent performance of SR 3.3.4.1.6. 2. The STAGGERED TEST BASIS Frequency shall be determined on a per Function basis. --- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS
SR 3.3.4.1.6	Determine RPT breaker interruption time.	60 months

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level — Low Low Low, Level 1	1,2,3	4	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -147.7 inches
b. Main Steam Line Pressure — Low	1	4	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 837 psig
c. Main Steam Line Flow — High	1,2,3	4	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 178 psid
d. Condenser Vacuum — Low	1,2[a], 3[a]	4	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7.6 inches Hg vacuum
e. Main Steam Tunnel Temperature — High	1,2,3	4	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Main Steam Line Turbine Building Temperature — High	1,2,3	4	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	Modules 1-4 ≤ 142°F, Module 5 ≤ 150°F
g. Manual Initiation	1,2,3	4	J	SR 3.3.6.1.6	NA

(continued)

(a) With any turbine stop valve not closed.

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	4(b)	K	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ - 47.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ - 47.7 inches
b. Drywell Pressure – High	1,2,3	4(b)	K	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
c. Reactor Vessel Water Level – Low Low, Level 2 (ECCS Divisions 1 and 2)	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
d. Drywell Pressure – High (ECCS Divisions 1 and 2)	1,2,3	4(b)	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
e. Reactor Vessel Water Level – Low Low, Level 2 (HPCS NSPS Div 3 and 4)	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
f. Drywell Pressure – High (HPCS NSPS Div 3 and 4)	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig

(continued)

(b) Also required to initiate the associated drywell isolation function.

(c) During operations with a potential for draining the reactor vessel.

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation (continued)					
g. Containment Building Fuel Transfer Pool Ventilation Plenum Radiation – High	(c),(d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 500 mR/hr
h. Containment Building Exhaust Radiation – High	1,2,3	$4^{(b)}$	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c),(d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
i. Containment Building Continuous Containment Purge (CCP) Exhaust Radiation – High	1,2,3	$4^{(b)}$	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
	(c),(d)	4	N	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 400 mR/hr
j. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	$4^{(b)}$	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -147.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -147.7 inches
k. Containment Pressure-High	(e)	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 3.0 psid
l. Manual Initiation	1,2,3	$2^{(b)}$	J	SR 3.3.6.1.6	NA
	(c),(d)	2	N	SR 3.3.6.1.6	NA

(continued)

(b) Also required to initiate the associated drywell isolation function.

(c) During operations with a potential for draining the reactor vessel.

(d) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the primary or secondary containment.

(e) MODES 1, 2, and 3 with the associated PCIVs not closed.

Table 3.3.6.1-1 (page 4 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 118.5 inches water
b. RCIC Steam Line Flow – High, Time Delay	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 13 seconds
c. RCIC Steam Supply Line Pressure – Low	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 52 psig
d. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig
e. RCIC Equipment Room Ambient Temperature – High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 207°F
f. Main Steam Line Tunnel Ambient Temperature – High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
g. Main Steam Line Tunnel Temperature Timer	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 28 minutes
h. RHR Heat Exchanger Ambient Temperature – High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 160°F
i. RCIC/RHR Steam Line Flow - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 188 inches water

(continued)

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
j. Drywell Pressure - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
k. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow - High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 66.1 gpm
b. Differential Flow-Timer	1,2,3	2	I	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 47 seconds
c. RWCU Heat Exchanger Equipment Room Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 205°F
d. RWCU Pump Rooms Temperature-High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 202°F
e. Main Steam Line Tunnel Ambient Temperature-High	1,2,3	2	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 171°F
f. Reactor Vessel Water Level — Low Low, Level 2	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
	(c)	4	O	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -47.7 inches
g. Standby Liquid Control System Initiation	1,2	2	L	SR 3.3.6.1.6	NA
h. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA
	(c),(d)	2	N	SR 3.3.6.1.6	NA

(continued)

(c) During operations with a potential for draining the reactor vessel.

(d) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the primary or secondary containment.

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION F.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RHR System Isolation					
a. RHR Heat Exchanger Ambient Temperature – High	1,2,3	2 per room	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 160°F
b. Reactor Vessel Water Level - Low, Level 3	1,2,3[f]	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 8.3 inches
c. Reactor Vessel Water Level - Low, Level 3	3[g],4,5	4[h]	M	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 8.3 inches
d. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ -147.7 inches
e. Reactor Vessel Pressure – High	1,2,3	4	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 150 psig
f. Drywell Pressure – High	1,2,3	8	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 1.88 psig
g. Manual Initiation	1,2,3	2	J	SR 3.3.6.1.6	NA

(f) With reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure.

(g) With reactor steam dome pressure less than the RHR cut in permissive pressure.

(h) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

LCO 3.3.8.2 One RPS electric power monitoring assembly shall be OPERABLE for each inservice RPS special solenoid power supply or alternate power supply.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both inservice power supplies with the electric power monitoring assembly inoperable.	A.1	Remove associated inservice power supply(s) from service.	1 hour
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 4 or 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies.	C.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.5.2 Drywell Air Lock

LCO 3.6.5.2 The drywell air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTES-----

1. Entry and exit is permissible to perform repairs of the affected air lock components.
2. Enter applicable Conditions and Required Actions of LCO 3.6.5.1, "Drywell," when air lock leakage results in exceeding overall drywell bypass leakage rate acceptance criteria.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102 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

Amendment No. 95 to the Clinton Power Station Technical Specifications, issued on December 2, 1994, approved implementation of the Improved Technical Specifications (ITS). The ITS were based on NUREG-1434, Rev 0, "Standard Technical Specifications, General Electric Plants, BWR/6." The Clinton Power Station was the lead plant for the four BWR/6 facilities and was the first to receive approval by the staff. During subsequent reviews of the remaining BWR/6 facilities, improvements to the ITS were identified and incorporated.

By letter dated October 27, 1995, Illinois Power requested a number of changes to their technical specifications (TSs). These changes would be consistent with those approved for other BWR/6 facilities. In addition, the licensee proposed to correct a number of administrative typographical errors resulting from Amendment No. 95.

2.0 EVALUATION

The licensee requested six separate changes to their TSs. The proposed changes, along with the staff's evaluation follows:

1. LCO 3.1.3, "Control Rod OPERABILITY"

Prior to Amendment No. 95, TS Surveillance Requirement (SR) 4.1.3.1.2 required "moving each control rod at least one notch at least once per 7 days." The ITS modified this SR and differentiated between control rods that are fully withdrawn as opposed to control rods that are only partially withdrawn. New SR 3.1.3.2 requires that each fully withdrawn control rod be inserted at least one notch once every 7 days whereas SR 3.1.3.3 requires that each partially withdrawn control rod be inserted at least one notch once every 31 days. As indicated, the ITS differs from the previous TSs in that it requires control rods to be "inserted" rather than just "moved."

Control rod positioning must be continually adjusted during the fuel cycle and individual rods will periodically change from the "partially withdrawn" to the "fully withdrawn" position or vice versa. When making this transition, the individual rods fall under different SRs. A partially withdrawn rod, which must be inserted at least one notch at least once per 31 days per SR 3.1.3.3, falls under SR 3.1.3.2 (which requires that the rod be inserted at least one notch at least once per 7 days) when it becomes fully withdrawn. Maintaining a tracking system for all 145 individual control rods at the Clinton Power Station would become an administrative burden. Therefore, the licensee has chosen to track control rods by SR as opposed to tracking them individually.

During the development of the ITS, it was recognized that the requirement for control rods to be "inserted" as opposed to just "moved" had the potential to impose unnecessary testing. As an example, if a partially withdrawn rod (which must be tested at least once per 31 days) is fully withdrawn (which must be tested at least once per 7 days), the rod may have to be initially inserted just to be current with the SR. The ITS attempted to resolve this situation by inserting Notes into SRs 3.1.3.2 and 3.1.3.3. The Note in SR 3.1.3.2 states that the surveillance need not be performed within 7 days. Similarly, the Note in SR 3.1.3.3 states that the surveillance need not be performed within 31 days. These Notes were intended to prevent unnecessary testing and would further assist the licensee in tracking control rods by SR as opposed to individually.

Subsequent to the Clinton Power Station implementation of the ITS, it was recognized that the Notes inserted into SRs 3.1.3.2 and 3.1.3.3 may not always provide the intended flexibility. SR 3.0.2 of the ITS allows up to 25% extension to surveillance frequencies. Therefore, the 7 day surveillance frequency of SR 3.1.3.2 could be extended upwards to 8 days 18 hours whereas the 31 day surveillance frequency of SR 3.1.3.3 could be extended upwards to 38 days 18 hours. However, during the review of the remaining BWR/6 facilities, it was recognized that the 25% extension was only applicable to surveillance frequencies and not to times identified in Notes. Thus, the potential still exists for unnecessary testing of control rods whose test frequency has changed.

In response to this concern, the staff modified the Notes to SRs 3.1.3.2 and 3.1.3.3 for both the Grand Gulf and River Bend facilities to include the 25% extension. Thus, for these two facilities, SR 3.1.3.2 states that the surveillance need not be performed within 8 days 18 hours and SR 3.1.3.3 similarly states that the surveillance need not be performed within 38 days 18 hours. Illinois Power has requested to make these identical changes and the staff finds their proposal acceptable. These changes will provide the scheduling flexibility originally intended for these Notes. In addition, the licensee proposed to insert the word "fully" into SR 3.1.3.2 to clarify that the time allowance of the Note begins when the control rod is fully withdrawn rather than upon its initial (partial) withdrawal. This change was previously included for both the Grand Gulf and River Bend facilities and the staff also finds it acceptable for the Clinton Power Station.

2. LCO 3.3.1.1, "Post Accident Monitoring (PAM) Instrumentation"

Technical Specification Table 3.3.3.1-1, "Post Accident Monitoring Instrumentation," lists the PAM instrumentation that must remain operable. Function 7 of this table lists "Penetration Flow Path, PCIV Position." As described in the Bases of the technical specifications, primary containment isolation valve (PCIV) position is required for verification of containment integrity. The Bases clearly states that valve position indication is only necessary for each automatic PCIV in a containment penetration flow path and that position indication is not needed for valves in an isolated penetration. The Bases specifically excludes penetrations having a closed and deactivated automatic valve, a closed manual valve, blind flange, or check valve with flow through the valve secured.

The licensee has proposed inserting the word "automatic" such that Function 7 of Table 3.3.3.1-1 will read "Penetration Flow Path, Automatic PCIV Position." This change will clarify that only PCIVs with automatic actuation need to have PAM position indication. This change reflects the staff's original intention for this TS and was previously included for the Grand Gulf facility. Therefore, the staff also finds it acceptable for the Clinton Power Station.

3. <u>LCO 3.3.4.1, "End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation"</u>

Due to an oversight during the conversion of the CPS technical specifications to the ITS, SR 3.3.4.1.5 contains an incorrect reference. SR 3.3.4.1.5 requires periodic verification of the EOC-RPT response time and makes reference to the RPT breaker interruption time of SR 3.3.4.1.7. However, as a result of Amendment No. 94 which was issued on November 3, 1994, the surveillance associated with the RPT breaker interruption time was renumbered as 3.3.4.1.6 and SR 3.3.4.1.7 was deleted.

The licensee's letter proposes to delete reference to SR 3.3.4.1.7 (which no longer exists) and insert the correct reference of SR 3.3.4.1.6. This modification is administrative in nature and eliminates an incorrect reference. Therefore, the staff finds this proposed change acceptable.

4. <u>LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"</u>

Due to an oversight during the conversion of the CPS technical specifications to the ITS, the header to Table 3.3.6.1-1, "Primary Containment and Drywell Isolation Instrumentation," incorrectly states that the table consists of seven pages. Table 3.3.6.1-1 only consists of six pages.

The licensee has proposed to revise the header of each page of Table 3.3.6.1-1 to correctly state that the table consists of six pages. This modification is administrative in nature and eliminates incorrect information. Therefore, the staff finds this proposed change acceptable.

5. <u>LCO 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring"</u>

Due to an oversight during the conversion of the CPS technical specifications to the ITS, a formatting error was included under the APPLICABILITY section. The licensee has proposed to insert an indentation in order to maintain uniformity within the technical specifications. This formatting change is editorial in nature and is acceptable to the staff.

6. LCO 3.6.5.2, "Drywell Air Lock"

Due to an oversight during the conversion of the CPS technical specifications to the ITS, an editorial error was included under the ACTIONS section. The Notes section title block is currently identified as "NOTE" when, in fact, the section contains two Notes. The licensee has proposed revising the word "NOTE" to the plural form of "NOTES". This change is editorial in nature and is acceptable to the staff.

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 a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and 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 (60 FR 65680). 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: Douglas V. Pickett

Date: February 29, 1996