October 18, 1990

Docket No. 50-461

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

Dear Mr. Spangenberg:

SUBJECT: ISSUANCE OF AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-62 (TAC NO. 76821)

The Commission has issued the enclosed Amendment No. 51 to Facility Operating License No. NPF-62 for the Clinton Power Station, Unit No. 1. This amendment is in response to your application dated April 27, 1990.

This amendment revises the pressure-temperature limits in the Clinton Technical Specifications in response to recommendations in NRC Generic Letter 88-11.

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

Sincerely,

original signed by

John B. Hickman, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 51 to License No. NPF-62
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: 76821 AMD

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Surname:	PKreutzer	JHickman	JHannon	BmB
Date:	9/18/90	9/28/90	9/28/90	10/2/90

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

October 18, 1990

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Licensing and Safety
Clinton Power Station
P. O. Box 678
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See next page

Mr. Frank A. Spangenberg
Illinois Power Company

Clinton Power Station
Unit 1

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

ILLINOIS POWER COMPANY, ET AL.

DOCKET NO. 50-461

CLINTON POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
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 27, 1990 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. 51 , 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 N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 18, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 51

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	Insert
xviii	xviii
3/4 4-22	3/4 4-22
3/4 4-23	3/4 4-23
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
B 3/4 4-6	B 3/4 4-6
B 3/4 4-8	B 3/4 4-8
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11

BASES

SECTION PAGE

INSTRUMENTATION (Continued)

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor vessel pressure and metal temperature shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. The maximum rate of change of reactor vessel steam space coolant temperature during normal heatup or cooldown shall be limited to 100°F in any 1 hour.

b. A maximum metal temperature change of $\leq$ 20°F in any 1 hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

c. The reactor vessel flange and head flange metal temperature shall be $\geq$ 70°F when reactor vessel head bolting studs are under full tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor vessel pressure and metal temperature of the reactor vessel flange surfaces, bottom head outside surface and bottom head inside surface, as measured by the bottom head drain temperature, shall be determined to be within the operating limits defined by Figure 3.4.6.1-1 at least once per 30 minutes.

4.4.6.1.2 The reactor steam space coolant temperature shall be determined to be within the heatup and cooldown limits of 100°F in any 1 hour at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.3 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.4 The reactor vessel material specimens shall be removed and examined to determine changes in reactor pressure vessel material properties as a function of time and THERMAL POWER as required by 10 CFR 50, Appendix H, in accordance with the schedule in Table 4.4.6.1-1. The results of these examinations shall be used to adjust the curves of Figure 3.4.6.1-1.

4.4.6.1.5 DELETED.

4.4.6.1.6 The reactor vessel flange and head flange temperature shall be verified to be $\geq$ 70°F when vessel head bolting studs are under full tension:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 90°F, at least once per 12 hours.
   2. $\leq$ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs except 10 percent of the bolting studs may be fully tensioned at $\geq$ 10°F but $\leq$ 70°F.

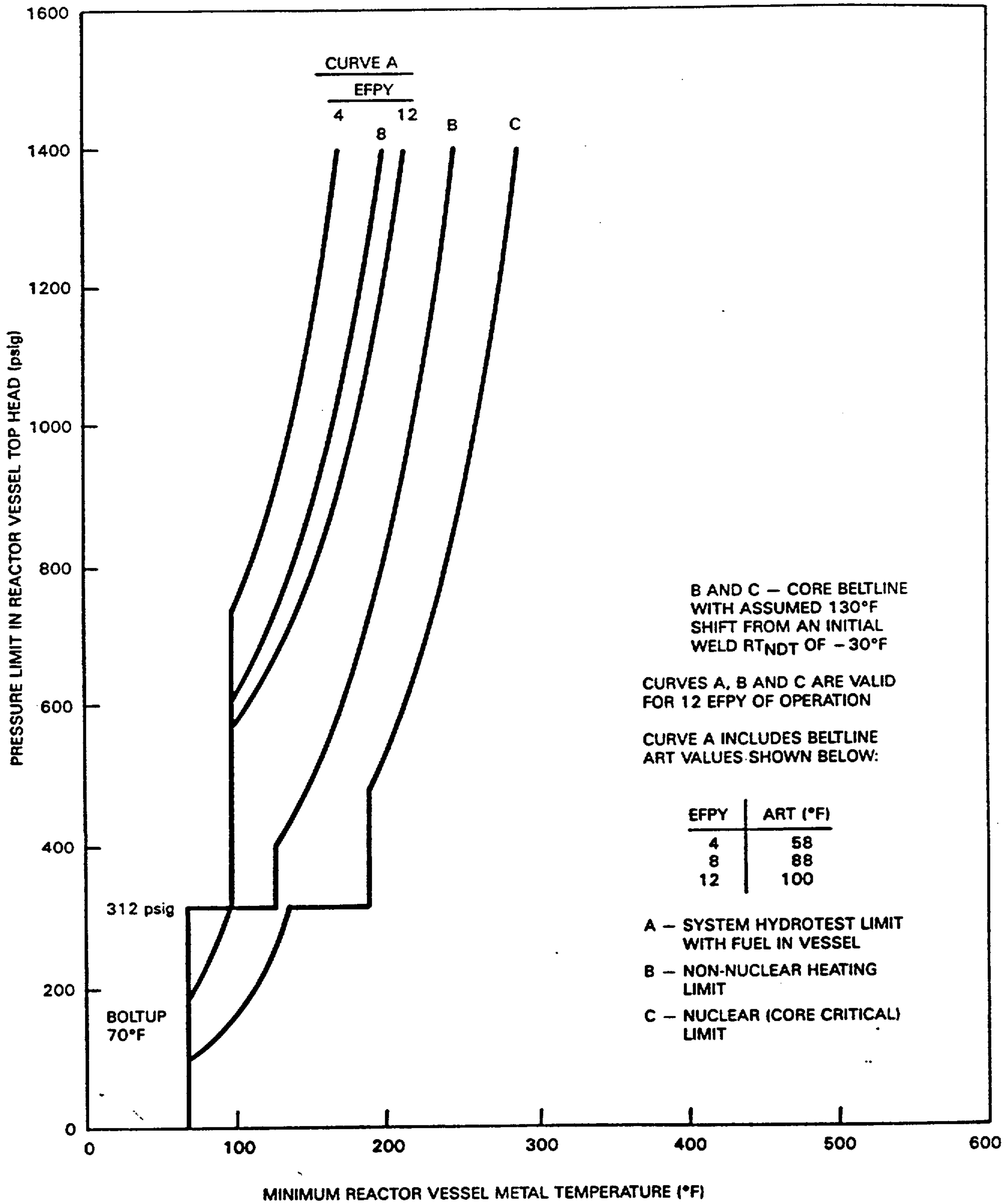


Figure 3.4.6.1-1. Reactor Vessel Pressure Versus
Minimum Reactor Vessel Metal Temperature

TABLE 4.4.6.1-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR at I.D.	WITHDRAWAL TIME (EFPY)
1. Capsule 1	3°	0.67	10
2. Capsule 2	177°	0.67	20
3. Capsule 3	183°	0.67	Spare

3/4.4.6 PRESSURE/TEMPERATURE LIMITS (Continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT} of the core beltline region. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A, B, and C, includes an assumed shift in RT_{NDT} for the conditions at 12 Effective Full Power Years. The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used to predict reactor vessel material transition temperature shift. Flux wires which were removed after the first fuel cycle and will be removed at later intervals with the surveillance specimens are analyzed and provide an improved neutron fluence estimate for the reactor vessel. This data is then used to modify Bases Figure B 3/4.4.6-1 and predictions of reactor vessel material transition temperature shift per Regulatory Guide 1.99, Revision 2. The operating limit curves of Figure 3.4.6.1-1 have been and will be adjusted, as required, on the basis of the specimen data and the recommendations of Regulatory Guide 1.99, Revision 2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C and A for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1-1 to assure compliance with the requirements of Appendix H to 10 CFR 50.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment; however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

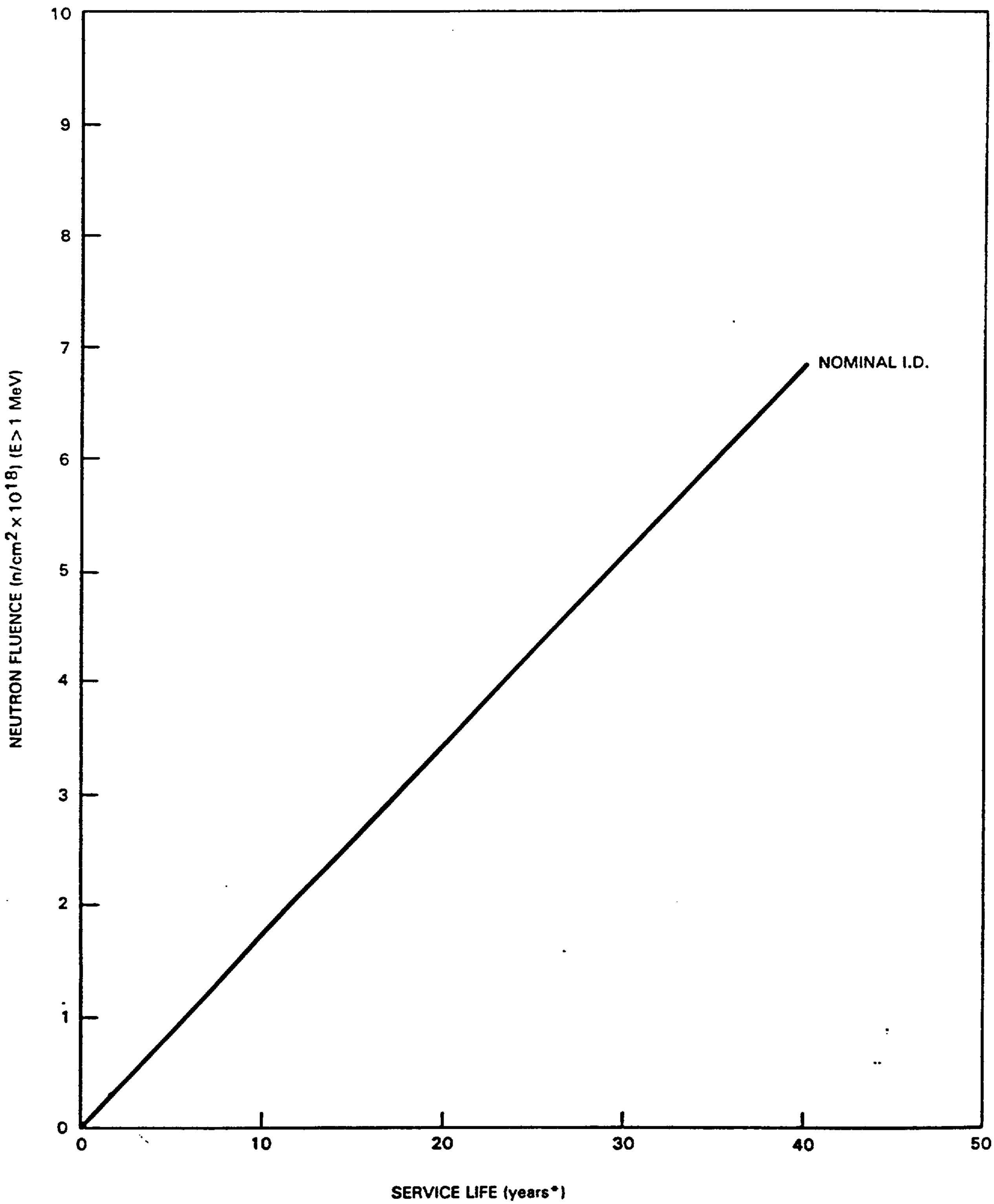
BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS VALUES

I.	LIMITING BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT#-SLAB# OR HEAT#/LOT#	CU(%)	Ni(%)	STARTING RT_{NDT} (°F)	ΔRT_{NDT} (°F)*	MIN. EOL UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT} (°F)
	PLATE	SA-533 GR.B, CL.1	C 4380-2	0.07	0.063	-20	69	85	49
	WELD	N/A	76492/ L430B27AE	0.10	1.08	-30	164	76	134

NOTE: * These values are given only for the benefit of calculating the end-of-life (EOL)(32EFPY) RT_{NDT}.

I.	NON-BELTLINE COMPONENT	MT'L TYPE OR WELD STEAM I.D.	HEAT # - SLAB # OR HEAT #/LOT #	HEAT # - SLAB # HIGHEST STARTING RT_{NDT} (°F)
	SHELL RING	SA-533 GR.B CL.1	C4240-2 A2758-1	-10
	BOTTOM HEAD DOME	"	A2757-1	-10
	BOTTOM HEAD TORUS	"	C4027-1	+10
	TOP HEAD DOME	"	C4374-3	-40
	TOP HEAD TORUS	"	A2879-2	-10
	TOP HEAD FLANGE	SA-508, CL. 2	CCZ 41-5478 SER 915	-40
	VESSEL FLANGE	"	CWS 51-5218 SER 878	0
	FEEDWATER NOZZLE	"	Q2AL10W	-20



Bases Figure B 3/4.4.6-1. Fast Neutron Fluence (E> 1 MeV) at I.D. Surface as a Function of Service Life

*At 90% Rated Thermal Power and 90% Availability

Bases Figure B 3/4.4.6-2 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 51 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

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the Illinois Power Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Clinton Technical Specifications, Section 3.4. The request was documented in letters from the licensee dated December 6, 1988 and April 27, 1990. This revision changes the P/T limits from 20 to 12 effective full power years (EFPY). The proposed P/T limits were based on Regulatory Guide 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Clinton reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 12 EFPY is weld 76492 with 0.1% copper (Cu), 1.08% nickel (Ni), and an initial RT_{ndt} of -30°F.

The licensee has not removed any surveillance capsules from the Clinton reactor vessel because the removal date for the first capsule has not been reached. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld 76492, the staff calculated the ART to be 100.3°F at 1/4T (T = reactor vessel beltline thickness) and 79.6°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 1.85E18 n/cm^2 at 1/4T and 9.4E17 n/cm^2 at 3/4T. The ART was determined per Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Clinton reactor vessel.

The licensee calculated an almost identical ART of 100°F for the same weld 76492 using RG 1.99, Rev. 2. The staff considers the difference of 0.3°F insignificant (100.3 vs 100°F). Substituting the ART of 100.3°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests.

Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 0°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest unirradiated Charpy impact upper shelf energy is plate A2758-1 with 67 ft-lb. Using Figure 2 of RG 1.99, Rev. 2, the staff calculated that the EOL USE would be 55.3 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 12 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Clinton Technical Specifications.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 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; and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Tsao

Dated: October 18, 1990