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

CONNECTICUT YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-213

HADDAM NECK PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. DPR-61

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

   A. The application by Connecticut Yankee Atomic Power Company (the licensee) dated September 29, 1978, as supported by the analyses submitted by letters dated September 22 and October 20, 1978, 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 license amendment, 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, Facility Operating License No. DPR-61 is hereby amended by deleting paragraph C.(3) in its entirety, by changing the Technical Specifications as indicated in the attachment to this license amendment and by amending paragraph 2.C.(2) to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 29 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 24, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY LICENSE NO. DPR-61

DOCKET NO. 50-213

Revise the Appendix A Technical Specifications by deleting the following pages and inserting the enclosed pages. The revised pages contain the captioned amendment number and vertical lines reflecting the area of change.

Remove Pages	Insert Pages
1-6	1-6
2-5	2-5
2-7	2-7
3-10b	3-10b
3-16	3-16
3-17	3-17
-	3-17a

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥0.99	> 5%	≥ 350°F
2. STARTUP	> .97	≤ 5%	≥ 350°F
3. HOT STANDBY	≤ 0.97	0	≥ 350°F
4. HOT SHUTDOWN	≤ 0.97	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	≤ 0.97	0	≤ 200°F
6. REFUELING**	≤0.92	0	≤ 140°F

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

2.4 MAXIMUM SAFETY SETTINGS - PROTECTIVE INSTRUMENTATION

Applicability: Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, and flow.

Objective: To provide for protective action in the event that the principle process variables approach a safety limit.

Specification: Protective instrumentation trip settings shall be as follows:

	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating
(1) Pressurizer Pressure	$\leq$ 2300 psig	$\leq$ 2300 psig
(2) Pressurizer Level*	$\leq$ 86% of range	$\leq$ 86% of range
(3) Variable Low Pressure***	$\geq 17.4(T_{avg}+1.38\Delta T)-8850$	$\geq 17.4(T_{avg}+1.38\Delta T)-8850$
(4) Nuclear Overpower**	$\leq$ 109% of rated power	$\leq$ 84% of rated power
(5) Low Coolant Flow***	$\geq$ 90% of normal loop flow	$\geq$ 90% of normal loop flow
(6) Reactor Coolant Loop Valve-Tempoerature Interlock	$\leq$ 20°F	$\leq$ 20°F
(7) High Steam Flow	110% of full load steam flow	110% of full load steam flow

* May be bypassed when the reactor is at least 1.5%Δk subcritical.

** The nuclear overpower trip is based upon a symmetrical core power distribution. If any asymmetric power distribution should occur, resulting in the power in any quadrant being 10% greater than the average core power as indicated by the neutron detectors and loop ΔT measurements, the nuclear overpower trip on all channels shall be reduced one percent for each percent deviation greater than the above 10%. If it is determined that asymmetric power distributions exceed 30%, the reactor will be shut down. When the reactor power is <10% the overpower trip setpoint is reduced to 25% of rated power.

*** May be bypassed below 10% of rated power.

Basis: The reactor protective system is designed and constructed such that no single failure in any of the instrument systems will prevent the desired safety action if an applicable parameter exceed a safety set point.

and shutdown. It is safe to block this trip below 10% power since the protection afforded by this trip is not required at this low level. Removal of unnecessary trip signals will reduce the number of spurious trips.

(4) Nuclear Overpower. As explained above, the nuclear overpower reactor trip, in conjunction with the variable low pressure reactor trip, provides overpower, overtemperature protection. The nuclear overpower trip channels will respond first to rapid reactivity insertion rates, detected by the increase in flux, before there are any significant changes in the system process variables. A maximum error of 9% of full power due to set point, instrumentation, and calorimetric determination (see Section 4.3.6 of the FDSA) is considered in establishing the set point. In order to reduce the time to trip for certain accidents occurring at low power, the overpower setpoint is lowered to 25% when reactor power is below 10%. This low overpower trip would terminate the postulated large steamline break accident from the hot zero power condition. The lower setting for three loop operation provides protection at the reduced power level equivalent to that provided by the setting for four loop operation at full power. The reduction in setting in the event of an asymmetric power distribution provides protection for the more adverse hot channel factors. The asymmetry is detected primarily by observation of changes in loop ΔT and neutron detector ion chamber current readings, each of which is displayed at the control board. The loop ΔT is the difference between the hot leg and cold leg temperature of the reactor coolant, as measured at the steam generator. Experience with operating reactors indicates that power distribution asymmetrics from $\pm$ 3 to 6% of nominal power can be detected by either of the above methods. If any of the neutron detectors or loop ΔT measurements differs from the average by more than 6%, a critical review of core power distributions will be performed to evaluate the significance of the measurement from the standpoint of power distribution.

(5) Low Coolant Flow. The low coolant flow reactor trip protects the core against an increase in coolant temperature resulting from a reduction in coolant flow while the reactor is at substantial power[(3)]. This trip will prevent DNB in any loss-of-flow incident, which eliminates the possibility of clad damage. Flow detection in each reactor coolant loop is from a measurement of pressure drop from inlet to outlet of each steam generator. The 90% low flow signal is high enough to activate a trip in time to prevent DNB, and low enough to reflect that a loss-of-flow condition truly exists. A maximum instrument and set point error of 5% full flow is considered in determining the set point. Loss-of-flow protection is also provided by reactor coolant pump breaker and from undervoltage

BASIS

This specification assures that adequate emergency core cooling capacity is available whenever the reactor is critical. Based on the loss of coolant analysis, melting of the cladding is prevented with only one high pressure safety injection pump and one low pressure safety injection (core deluge) pump in operation. Each of the two trains of emergency core cooling equipment includes these two pumps. With the pumps associated with both trains of emergency core cooling equipment operable, substantial margin exists whenever normal power supplies or both diesel generators are available. With only one diesel generator operating and the pumps associated with that diesel operable as required in Item (2) of Specification 3.12, the high pressure safety injection pump and the low pressure safety injection pump would be started automatically. When the safety injection pumps are operating on off-site power, the charging pump would be started automatically. The RHR pump would be available for manual start for long-term recirculation cooling.

(2) FDSA Section 5.2.8

(3) D. C. Switzer (CYAPCO) letter to D. L. Ziemann (NRC) dated May 22, 1978.

(4) D. C. Switzer (CYAPCO) letter to D. L. Ziemann (NRC) dated May 24, 1978.

3.10 REACTIVITY CONTROL

Applicability: Applies to control group position during power operation and shutdown margin during subcritical operation (except refueling).

Objective: To define control group insertion limits which insure: (1) An acceptable core power distribution during power operation, (2) A conservative limit on potential reactivity insertion for a hypothetical control rod ejection, and (3) Adequate shutdown margins after a reactor trip. To insure that a least 3% δK shutdown margin is available during subcritical operation.

Specification:

A. Except for low power physics test at or below 10 percent of full power or determination of "just critical" rod positions, operation of the control group banks shall be maintained above the limits shown in Figure 3.10-1.

B. If it is determined that a rod has been dropped, an evaluation of the effect of the dropped rod shall be made to establish permissible power levels for continued operation.

C. No more than one dropped or one stuck rod shall be permitted while the reactor is critical nor shall that rod remain in such condition for more than one full power month of reactor operation.

D. The maximum worth of any individual control rod in the core at rated power shall not exceed 0.17% ΔK, as measured at the beginning of core life.

E. The maximum worth of any individual control rod in the core with the reactor just critical shall not exceed 0.83% ΔK, as measured at the beginning of core life.

F. Except for physics testing, a 3% δK shutdown margin shall be maintained during subcritical operation. This shutdown margin may be provided by control rods actually inserted, control rods available to insert (considering a stuck rod), and/or soluble boron.

Basis: Specification C limits the time a dropped control rod may be in the core because lower fuel depletion and fission product inventory in the vicinity of the dropped rod, relative to the rest of the core, increases the worth of that rod. The lack of fuel depletion and lack of fission products other than Xenon in the vicinity of a control rod which has been inserted for one full power month will have a negigible effect on the worth of that control rod. Xenon redistribution causes an appreciable increase in the worth of a dropped rod. The increased worth has been measured and found to be acceptable.

The methods of reactivity control to be used are fully explained in Section 4.2.2 of the FDSA. The control rod program was developed to insure that three major safety considerations are satisfied throughout core life. They are:

1. Power distributions (DNB ratios) with equilibrium xenon shall be at least as favorable as those used in the safety analysis and shall be within the limits of $F_{\Delta H}$ and Sepcification 3.17.

2. Sufficient shutdown margin shall be available to ensure that: (a) the reactor can be made sufficiently subcritical after a trip from any operating condition, including an allowance for the maximum worth stuck rod, and (b) the reactor does not return to criticality for any FDSA Chapter 10 postulated accident.

3. Potential ejected rod worths shall not exceed the limits specified in Section 10.2.7 of the FDSA.

The above safety considerations are satisfied because:

(1) As seen in Section 4.3 of the FDSA, the calculated minimum DNB ratio using the rod program is 3.05 compared with 2.82 using the power distributions considered in the safety analysis.

(2) Since shutdown margin requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS $T_{average}$, the limits of figure 3.10-1 are set to ensure that the shutdown margin after a reactor trip from the most limiting set of reactor conditions meets the design criteria (FDSA Section 4-2-2) of at least 3% δK with all rods inserted and at least 1% δK with the highest worth rod stuck out. In addition, operation within the limits of figure 3.10-1 ensures that sufficient shutdown margin (including a stuck rod) is available to prevent the reactor from returning to criticality during the most limiting FDSA Chapter 10 postulated accident.

(3) Of the three requirements, the third was calculated to be most limiting, and this forms the basis for Specifications A and D above. As shown in FDSA Section 10.2.7, the analysis for the rod ejection was quite conservative and a large margin would exist to fuel melting and dispersion.

(4) Power distribution, control rod worths and shutdown margins will be evaluated prior to initial startup and subsequent startups following refueling. Conformance to the above requirements will be checked at these times and the limit of Figure 3.10-1 adjusted to meet these requirements.

Specification C limits the time a dropped control rod may be in the core because lower fuel depletion and fission product inventory in the vicinity of the dropped rod, relative to the rest of the core, increases the worth of that rod. The lack of fuel depletion and lack of fission products other than Xenon in the vicinity of a control rod which has been inserted for one full power month will have a negligible effect on the worth of that control rod. Xenon redistribution causes an appreciable increase in the worth of a dropped rod. The increased worth has been measured and found to be acceptable.

Should a control rod be dropped, no immediate adverse effects would occur due to automatic load cut-back as described in FDSA Section 7.2.3.

Specification F insures at least 3% δK shutdown margin is available when the reactor is subcritical. This margin is required to offset the reactivity addition that would occur during a postulated large steamline break accident.

References: 1. FDSA Section 4.2

2. FDSA Section 7.2.3