Proposed Change No. 112
Supplement No. 2

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YANKEE ATOMIC ELECTRIC COMPANY

YANKEE

20 Turnpike Road Westborough, Massachusetts 01581

October 23, 1975

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation
Robert A. Purple, Chief
Operating Reactor Branch No. 1

Reference:
(1) License No. DPR-3 (Docket No. 50-29)
(2) Proposed Change No. 112, January 3, 1974
(3) USNRC letter, dated May 20, 1975, signed by Robert A. Purple
(4) USNRC letter, dated July 16, 1975, signed by Robert A. Purple
(5) Proposed Change No. 125, July 14, 1975

Dear Sir:

To assist you in your review of our Proposed Technical Specification, Reference 2, we are attaching three enclosures. Enclosure 1 contains our response to Reference 3 and Enclosure 2 contains our response to Reference 4. Where applicable we have included in each enclosure copies of revised specifications. Enclosure 3 contains copies of specifications that we have determined must be revised to reflect current plant conditions. The revised specifications show the changes from those submitted in Reference 2 by vertical lines in the right margin.

This supplement to the Proposed Change No. 112 has been reviewed by the Nuclear Safety Audit and Review Committee.

Any questions regarding this Proposed Change should be directed to our Operations Office, 20 Turnpike Road, Westboro, Massachusetts 01581, (617) 366-9011, Extension 56-208.

Respectfully submitted,

YANKEE ATOMIC ELECTRIC COMPANY

W. P. Johnson

W. P. Johnson
Vice President

COMMONWEALTH OF MASSACHUSETTS)
)ss.
COUNTY OF WORCESTER)

Then personally appeared before me, W. P. Johnson, who being duly sworn, did state that he is a Vice President of Yankee Atomic Electric Company, that he is duly authorized to execute and file the foregoing request in the name and on the behalf of Yankee Atomic Electric Company, and that the statements therein are true to the best of his knowledge and belief.

Armand R. Soucy

Armand R. Soucy Notary Public
My Commission Expires September 9, 1977

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

YANKEE NUCLEAR POWER STATION (YANKEE ROWE)

DOCKET NO. 50-29

TECHNICAL SPECIFICATIONS

Our response to your letter dated May 20, 1975 concerning certain sections of our proposed Technical Specifications is as follows:

QUESTION NO. 1

Specification 16.2.3.A includes design features of the reactor core which are no longer applicable. Revise this Specification to reflect the design features of the operating reactor core that we have reviewed and approved.

RESPONSE NO. 1

The attached revised Specification 16.2.3, Reactor, includes design features for our Core 12 to be inserted during our October-November 1975 refueling. Details of this Core were submitted in reference 5.

QUESTION NO. 2

Specification 16.2.3.B defines the design pressure and temperature of the reactor coolant system as 2500 psig and 650°F. In the Safety Analysis Report the design pressure and temperature of piping and fittings in the primary coolant system are given as 2285 psig and 550°F, respectively. Explain this apparent discrepancy and revise the specification accordingly.

RESPONSE NO. 2

The Safety Analysis Report is correct. The pressure vessels in the reactor coolant system are designed in accordance with the ASME Boiler and Pressure Vessel Code, which requires that vessels be designed for the most severe condition of coincident pressure and temperature expected in normal operation. Pressure relief devices must be provided that will prevent pressure from exceeding 110 percent of the design pressure.

The reactor coolant system piping and fittings are designed per the ANSI (formerly ASA) B31-1, Power Piping Code. The Power Piping Code allows the design to be based on normal operating pressure and temperature and also allows exceeding the design conditions for periods of time. The stress level can be increased 15 percent above the Code allowable design value for not more than 10 percent of the design life and up to 20 percent above the allowable for up to 1 percent of the design life. Since normal plant operating pressure is 2000 psig, there is no conflict with either design condition.

The setting of the primary system safety valves could allow pressure to increase to 2560 psig during a transient. The amount of time this condition is expected to exist is well within the allowances of B31.1.

For these reasons it is concluded that the design conditions for the reactor coolant system are consistent with the standards to which the plant was constructed. Specification 16.2.3 has been revised to correct the discrepancy and is attached.

QUESTION NO. 3

Specification 16.2.4.C sets forth the design features of the containment purge system for venting the containment atmosphere following a LOCA. This system includes an installed charcoal filter for which you have taken credit in the accident analysis. To ensure high confidence that this system will function reliably, when needed, at a degree of efficiency equal to or better than that assumed in the accident analysis, you should propose Limiting Conditions for Operation and Surveillance Requirements for this system to be incorporated in the Technical Specifications. Enclosed for your guidance are model specifications (and bases) for the containment purge system (see Enclosure II) which meet current requirements. Since these were not prepared explicitly for Yankee-Rowe, some editing may be necessary to adapt them to the Yankee-Rowe design and nomenclature.

RESPONSE NO. 3

The post accident hydrogen removal system (containment purge) was designed to maintain a low hydrogen gas concentration, therefore only a very low vent rate is required. We have revised page 6.2-12 of the FSAR to show no credit has been taken for the filter and that the venting doses are substantially below the values specified in 10CFR100, "Reactor Site Criteria."

QUESTION NO. 4

Specification 16.3.1 relating to limiting safety system settings for the Yankee-Rowe reactor protection system should define the setpoints for the "Low Main Coolant Flow-Reactor Scram" in terms of parameters (current to the main coolant pumps) consistent with the existing loss-of-flow instrumentation. The proposed setpoint at 13 inches for the "Steam Generator Low Water Level" should be revised to 15 inches as given in the existing Technical Specification or you should justify this discrepancy. Present Technical Specifications include the setpoint (5.2 dec/minute/max) for the "High Startup Rate-Reactor Scram" which should also be included in your proposed new format Technical Specifications. Although you state in the basis that you have not taken credit for this trip in the accident analysis, it is our position that its functional capability at the presently specified setting is required to enhance the overall reliability of the reactor protection system. Our position would also apply to the nuclear overpower low setpoint and the turbine-generator trip for which trip setpoints should be included in the Technical Specifications.

RESPONSE NO. 4

Specification 16.3.1, attached, has been revised to incorporate the above conditions with the exception of the Steam Generator Water Level. We have retained the 13 inches since the latest accident analysis (reference 5) used that value. This is a more conservative value since it is two inches closer to the normal water level.

QUESTION NO. 5

The proposed "Safety Limits-Reactor Core" in Specification 16.3.2 are no longer applicable. Revise this specification and its associated basis to reflect the conditions analyzed by you and approved by us for the operating Yankee-Rowe core.

RESPONSE NO. 5

Specification 16.3.2 has been revised to reflect the conditions of Core XII to be inserted in October 1975. Details of this Core were submitted in reference 5.

QUESTION NO. 6

The proposed "Safety Limit-Reactor Coolant System Pressure" in Specification 16.3.3 is inconsistent with the concept of a safety limit as defined in the regulations, since you did not allow a necessary margin between the safety valve setting and the specified safety limit. Revise this specification, by either reducing the code safety valve settings to provide an acceptable margin, or propose an appropriate higher value for the safety limit, consistent with the provisions permitted by the applicable codes for higher than the design pressures for the reactor coolant system components (reactor vessel, piping, etc.). Revise the basis accordingly.

RESPONSE NO. 6

The reactor coolant system pressure vessels are designed to the ASME Boiler and Pressure Code, Section VIII. The Code requires that pressure relieving devices be provided to prevent the pressure from exceeding 110 percent of the design pressure; this corresponds to a pressure of 2750 psia at 650°F for all vessels except the pressurizer and 2750 psia at 668°F for the pressurizer.

As discussed in the answer to Question No. 2, the settings of the primary system safety valves have been selected to ensure that the requirements of the design codes are met.

Revised Specification 16.3.3 is attached.

QUESTION NO. 7

With respect to the incore instrumentation the existing Technical Specifications permit continued operation only with two leaking thimbles (incore detector pathways) isolated and out of service. All other incore instrumentation is required to be operable. The proposed LCO's in Specification 16.4.1 "Core Instrumentation" would allow continued operation with one moveable incore neutron detector operable in one of the hottest instrumented fuel assemblies and with ten radial thermocouple positions. Furthermore, continued operation would be allowed indefinitely without the incore instrumentation operable provided the plant load and the nuclear overpower trip setpoint are reduced by 10%. To monitor power distribution and to verify that the total peaking factor (F_Q) remains below the specified limit the Yankee-Rowe incore instrumentation has 26 thermocouple positions and 22 incore pathways for the neutron detectors. Please provide an evaluation that will show how you can accurately determine with the proposed small number of operable incore instrumentation that F_Q is below the allowable limit. You should also include information that will show that there will be assurance that the F_Q will remain below the allowable limit during operations not exceeding 90% of full power to justify continued operation at that lower power level without F_Q surveillance.

RESPONSE NO. 7

We have revised this specification to delete the Exception permitting power operation for 48 hours with the loss of core instrumentation. To answer your questions concerning continued power operation with only one movable incore neutron detector and only ten radial thermocouple positions we have also expanded the Basis section of the specification. A copy of the revised specification is attached.

QUESTION NO. 8

Specification 16.4.5 relating to LCO's for the "Chemical Shutdown and the Charging and the Volume Control System" requires operable components for boron injection to assure the capability for boron injection at a rate in excess of 132,000 ppm-gal/min. Provide an explicit basis for this specified minimum rate of boron injection into the reactor coolant. The exception to the requirement to have two operable flow paths for boron injection when the reactor is critical, would permit operation with but one flow path when one reactor coolant loop is isolated. Propose a time limit for this mode of operation consistent with the required use of the loop fill header and the charging pump, to reduce the time operation is allowed in this mode to a minimum.

RESPONSE NO. 8

We have revised this specification to eliminate the 132,000 ppm-gal/min rate. The concentration of boron in the safety injection water and rate of injection are described elsewhere in the Technical Specifications. Additionally, we have established a time limit for continued power operation with only one chemical shutdown injection path. A copy of the revised specification is attached.

QUESTION NO. 9

Specification 16.4.6 relating to LCO's for the "Emergency Core Cooling System" should be revised to be more closely consistent with the Maine-Yankee Technical Specification 3.6 as applicable to Yankee-Rowe. Specifically, components in the ECCS required for long-term recirculation cooling shall also be required to be operable.

RESPONSE NO. 9

A revised Specification 16.4.6 is attached.

QUESTION NO. 10

The basis for Specification 16.4.7 "Minimum Volume and Boron Concentration Safety Injection Tank" state that the analysis of the loss-of-coolant incidents show that 77,000 gallons (to be transferred to the containment via core cooling before recirculation is normally established) will be sufficient to limit core temperatures and containment pressure for the spectrum of breaks. Provide an explicit reference for this basis.

RESPONSE NO. 10

We believe section 15.4.3 of the FSAR is an explicit reference for specification 16.4.7. Note that the basis of 16.4.7 refers to section 6.3 and that to section 15.4.3.

QUESTION NO. 11

Specification 16.4.8 "Reactor Core Energy Removal" includes requirements that the reactor shall not be at power unless a minimum steam relieving capacity of 1,000,000 lbs/hr is available above 10% of full rated power and a minimum steam relieving capacity of 1,900,000 lbs/hr is available above 75% full rated power. The number of on-line safety valves providing these steam relieving capacities should be specified. The explicit basis to show that these relieving capacities are adequate to maintain the pressure in the turbine cycle components within allowable limits of the ASME Code should be given.

RESPONSE NO. 11

Specification 16.4.8 Reactor Core Energy Removal has been revised to include figure 16.4.8-1 that prescribes the minimum steam generator safety valve relieving capacity versus percent of rated power. Revised Specification 16.4.8 is attached.

16.2.3 Reactor

Applicability: Applies to the reactor vessel, vessel core and internals as well as the reactor coolant system and components, including associated emergency core cooling systems.

Objective: To define those design criteria essential in providing for safe system operation which are not covered in Chapters 4 and 5 of the FSAR.

Specifications: A. Reactor Core

1. The reactor core approximates a right circular cylinder consisting of approximately 17,734 Zircaloy -4 clad fuel rods arranged into 76 Zircaloy clad assemblies respectively. The 76 assemblies are arranged in two zones. Each rod contains slightly enriched Uranium in the form of sintered UO_2 pellets. The initial fuel assemblies are enriched with U-235 to 4.0 wt. percent.

The fully loaded core contains approximately 18 metric tons of Uranium.

The core excess reactivity is controlled by a combination of boric acid chemical shim and control rod assemblies. There are 24 control rods. Twenty-two of the control rods are Inconel clad containing Ag-In-Cd in the wt. percents 80, 15 and 5, respectively, over 90 inches of their length. The absorber section of the remaining two control rods are blades of Hafnium.

B. Reactor Coolant System

1. The reactor coolant system is designed and constructed in accordance with the ASME Boiler and Pressure Code, Section VIII, Rules for Construction of Unfired Pressure Vessels, including all addenda through the 1956 edition; and the 1955 edition of the ASA Code for Pressure Piping, B31.1.

2. The reactor coolant system pressure vessels are designed for a pressure of 2500 psig and a temperature of 650°F, except for the pressurizer which has a design temperature of 668°F. The reactor coolant piping and fittings are designed for a pressure of 2300 psia and a temperature of 550°F.

3. The volume of the reactor coolant system is approximately 2940 cu. ft.

C. Emergency Core Cooling System

The emergency core cooling system consists of various subsystems each with internal redundancy. These subsystems include three high pressure and three low pressure safety injection pumps, one low pressure safety injection accumulator tank, a safety injection tank and the valves and interconnecting piping as shown in Section 6.3 of the FSAR.

Reference: FSAR, Chapters 4, 5, and 6.

1. Fraction of Radiation Energy Absorbed by Coolant

a.	Betas from fission products in fuel rods	0
b.	Fraction from fission products mixed with coolant	1
c.	Gammas from fission products in fuel rods, coolant in core	0.1

2. $G(H_2)$ - molecules/100 ev. 0.5

3. Extent of metal-water reaction, percent 5

4. Fission product Distribution Model

a.	Halogens	50% total inventory
b.	Solids	1% mixed with coolant

The hydrogen generation rate has been calculated and is presented in Figure 6.2-10. The hydrogen concentration within the containment, based upon the generation rate in Figure 6.2-10, and an initial 5% metal-water reaction is presented in Figure 6.2-11. The total quantity of Zircaloy present in the reactor core area is less than 10,000 pounds, assuming a full core of Zircaloy clad fuel elements.

A 4% hydrogen concentration is not reached until more than 1×10^7 seconds (139 days) have elapsed. It was assumed that venting commenced at 120 days at a 0.32%/day rate, which is sufficient to prevent a further hydrogen concentration increase. The vent flow would be directed through the system filter, however, no credit is taken for iodine or noble gas removal. An atmospheric dilution factor of 1×10^{-4} sec/m^3 is used for the site boundary dose calculation which is considered to represent a conservative value based on the on-site measurements. An atmospheric dilution factor of 1.35×10^{-5} sec/m^3 is used for the low population zone dose calculation based on analysis of the on-site data (Section 2.3.4.3).

The results of the dosecalculations are shown in the following table:

Post-Accident Venting Doses (rem)

	30 Day Vent Period		60 Day Vent Period	
	Thyroid	Whole Body	Thyroid	Whole Body
Site Boundary	4.5×10^{-2}	3.7×10^{-2}	4.8×10^{-2}	7.3×10^{-3}
Low Population Zone	6.0×10^{-3}	5.0×10^{-3}	6.5×10^{-2}	9.8×10^{-3}

The calculated doses at the site boundary and low population zone even for the 60 day vent period are substantially below the values specified in 10 C[illegible]00, "Reactor Site Criteria".

6.2.5.4 Testing and Inspection

Following installation the system will be tested to 150% of design pressure to verify strength and tightness.

16.3 Safety Limits and Limiting Safety Settings

16.3.1 Limiting Safety System Settings - Reactor Protective System

Applicability: Applies to reactor trip settings and bypasses for the instrument channels monitoring the process variables which influence the safe operation of the plant.

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

Specification: The reactor protective system trip setting limits and bypasses for the instrument channels shall be as follows:

16.3.1.1 Core Protection

a) Nuclear Overpower, High setpoint
$\leq$108% of rated power with 4 reactor coolant pumps operating.
$\leq$81% of rated power with 3 reactor coolant pumps operating.

b) Nuclear Overpower, Low setpoint
$\leq$35% of rated power whenever below 15 MWe gross.

c) Low Reactor Coolant Flow
$\geq$80% of flow in any two loops (4 pump operation).
$\geq$80% of flow in any single loop (3 pump operation).

M.C. pump motor current not less than 240 Amperes or more than 960 Amperes in any two loops (4 pump operation).

M.C. pump motor current not less than 240 Amperes or more than 960 Amperes in any single loop (3 pump operation).

These trips are bypassed below 15 MWe.

d) High Pressurizer Water Level
$\leq$200 inches whenever the reactor is critical.

e) Low Reactor Coolant System Pressure
$\geq$1800 psig whenever the reactor is critical.

f) Low Steam Generator Water Level
$\leq$13 inches below normal water level in two or more steam generators.

This trip is bypassed below 15 MWe.

g) High Startup Rate
Not more than 5.2 decades per minute.

This trip is automatically bypassed above 15 MWe.

16.3.1.2 Other Reactor Trips

a) Turbine
High thrust bearing movement, low bearing oil pressure, low condenser vacuum and overspeed.

These trips are bypassed below 15 MWe.

B) Generator
Overcurrent, differential and loss of field.

These trips are bypassed below 15 MWe.

Basis: Core Protection

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding resulting from those reactivity excursions too rapid to result in a high pressurizer water level trip. The prescribed setpoint, with allowances for errors, is consistent with the trip point used in the accident analysis. The lower setting for three loop operation provides the protection at the reduced power level equivalent to that provided by the setting for four loop operation at full power.

The low setting for neutron flux trip provides additional protection during reactor startup and shutdown even though this setting is not used in the transient or accident analysis.

The low reactor coolant flow trips protect the core against DNB should the coolant flow suddenly decrease significantly. The setpoints specified are consistent with the values assumed in the accident analysis.

The low reactor coolant pressure trip is provided to prevent operation when the DNBR is less than 1.3, including allowance for instrument error. This trip ensures that the thermal and hydraulic safety limits given in Section 16.3.2 are not exceeded. The low pressure trip also protects the core in the unlikely event of a loss-of-coolant accident.

The high pressurizer water level reactor trip prevents excessive pressure build-up on the loss of load accident. It also provides core protection for an uncontrolled rod withdrawal incident.

The low steam generator water level reactor trip mitigates the circumstances of loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to provide 15 minutes of margin before initiation of the emergency feedwater system.

The high-rate-of change reactor trip provides additional protection during reactor startup even though this setting is not used in the transient and accident analysis.

Other Reactor Trips

Above 15 MWe the usual turbine-generator protection devices also trip the reactor.

16.3.2 Safety Limits - Reactor Core

Applicability: Applies to the limiting combinations of reactor power, and reactor coolant system flow, temperature and pressure during operation.

Objective: To maintiain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant.

Specification: The reactor power level shall not exceed the allowable limit for the pressurizer pressure and the reactor coolant system cold leg temperatures, shown in Figures 16.3.2-1 and 16.3.2-2 for 4 and 3 loop operation respectively.

The safety limit is exceeded if the point defined by the combination of reactor coolant system cold leg temperature and power level is at any time above the appropriate pressurizer pressure line.

Basis: To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high-cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of reactor thermal power and of reactor coolant system flow, temperature and pressure, can be related to DNB through the use of the "W-3 DNB Correlation." The W-3 DNB Correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.3. A DNBR or 1.3 corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.

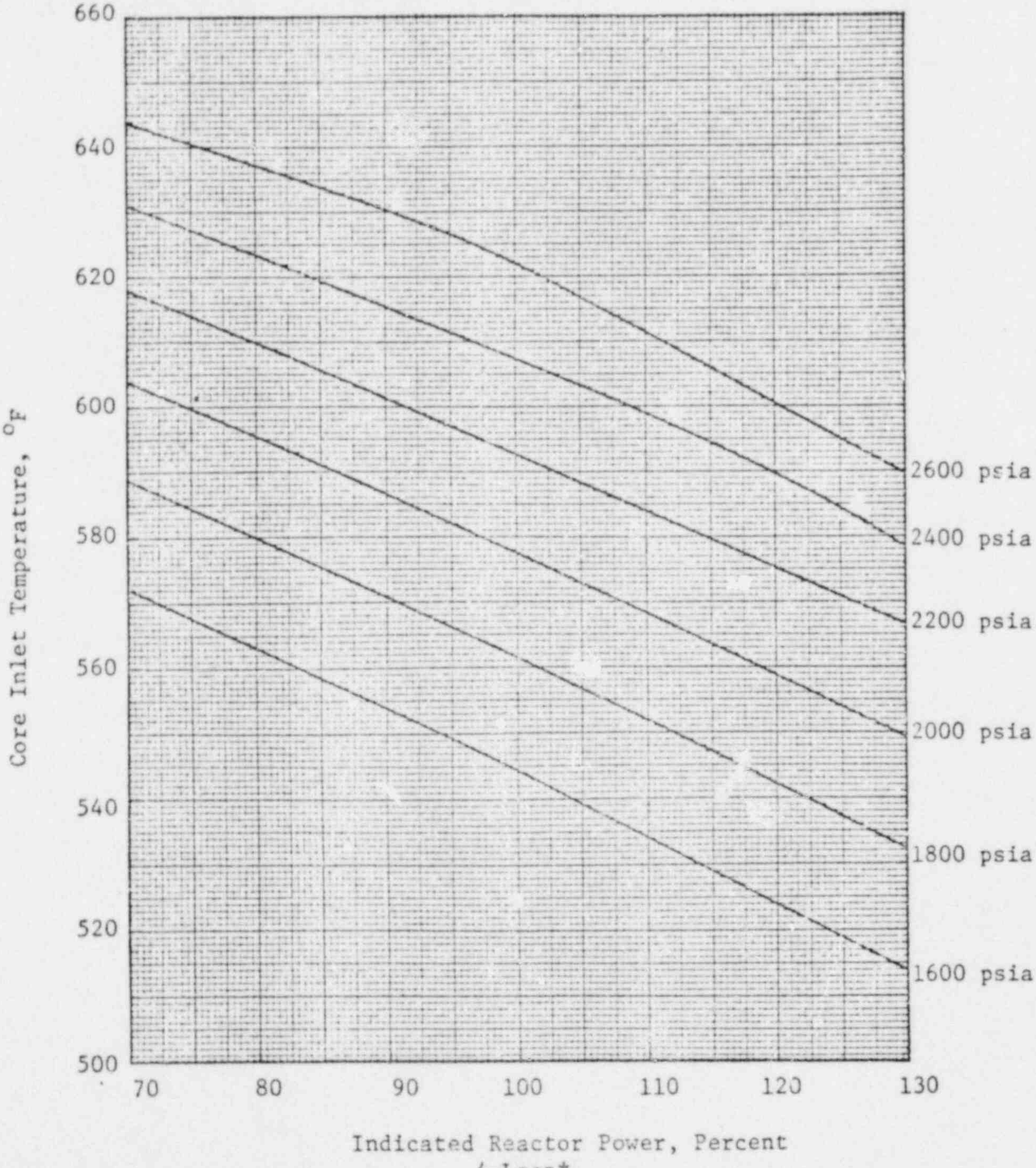
Because of flow instability, DNB may occur prematurely should the core exit quality become too great. The limiting core exit quality for preventing flow instability is taken conservatively as 0.08.

The curves of Figures 16.3.2-1 and 16.3.2-2 represent the loci of points of reactor thermal power, reactor coolant system pressure and cold leg temperature of various pump combinations for which the DNBR is no less than 1.3 and the exit quality is not greater than 0.08. The area of safe operation is below these limits.

The limiting hot channel factors used in determining the thermal limit curves are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion (control rod insertion limits are covered in Technical Specification 16.4.10).

References: FSAR, Chapter 15.

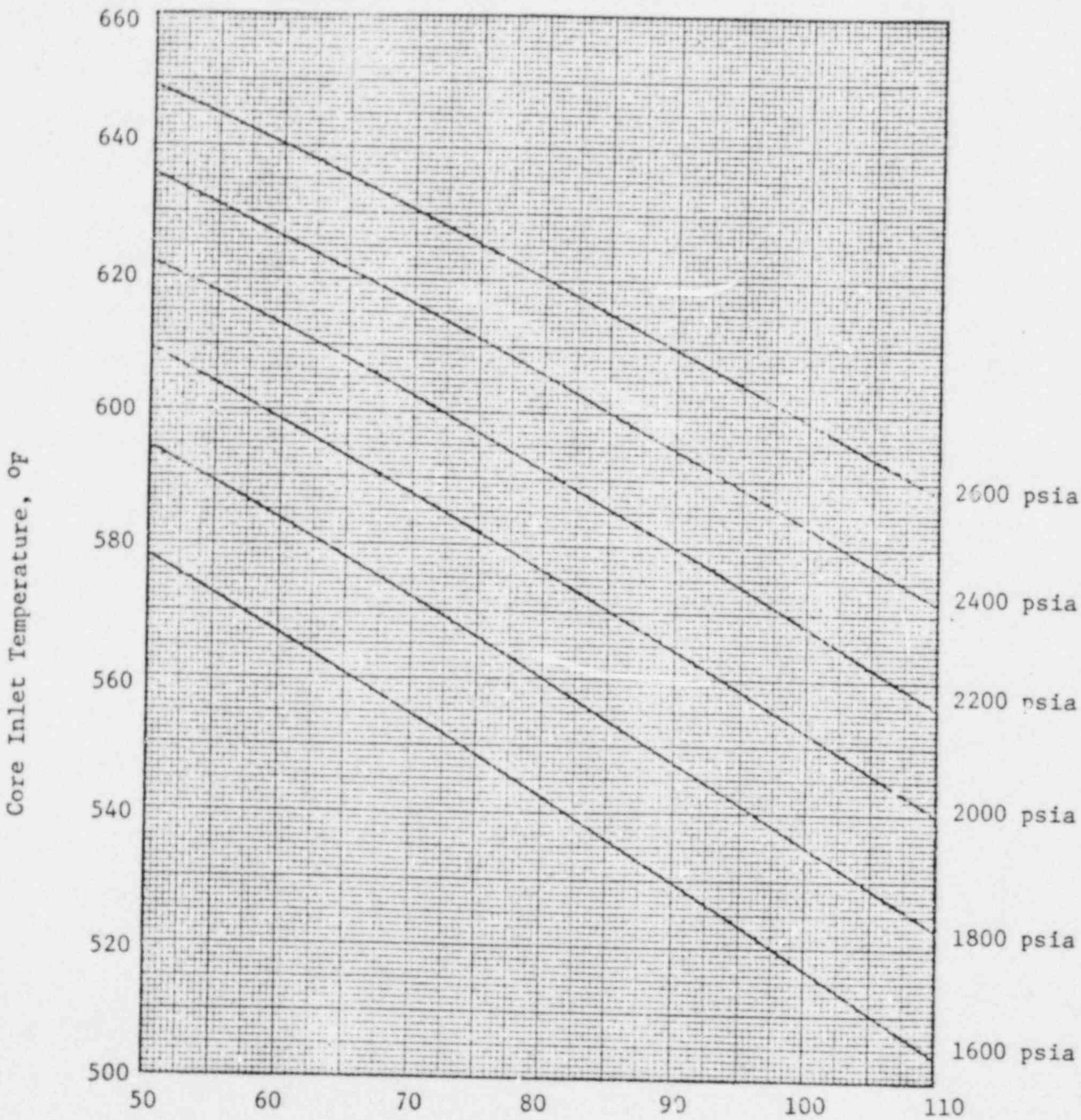
FIGURE 16.3.2-1
CORE XII SAFETY LIMIT CURVES FOR
4-LOOP OPERATION



* 100% = 600 MWt (Indicated)

POOR ORIGINAL

FIGURE 16.3.2-2
CORE XII SAFETY LIMIT CURVES FOR
3-LOOP OPERATION



* 100% = 600 MWt (Indicated)

16.3.3 Safety Limit - Reactor Coolant System Pressure

Applicability: Applies to the upper limit on reactor coolant system pressure.

Objective: To maintain the integrity of the reactor coolant system.

Specification: The reactor coolant system pressure shall not exceed 2735 psia when there are fuel assemblies in the reactor vessel.

Basis: The reactor coolant system serves as a barrier to prevent radionuclides in the reactor coolant from reaching the containment atmosphere. In the event of a fuel cladding failure, the reactor coolant system is the foremost barrier against the uncontrolled release of fission products. Establishing a system pressure limit gives assurance of the continued integrity of both the reactor coolant system and the fuel cladding. The reactor coolant system pressure vessel, pressurizer and pumps were constructed to the ASME Code, Section VIII, cases 1224 and 1234. The piping and valves were constructed to ASA Standards, Sections B31.1 (1955) and B16.5 (1957) respectively.

The settings and capacities of the steam generator safety valves (935; 985 and 1035 psig); the pressurizer safety valves (2485 and 2560 psig); and the high pressurizer water level trip (200 inches), have been established to assure that the reactor coolant system pressure safety limit is never reached.

The initial hydrostatic test of the pressurizer and reactor coolant system was conducted at 3435 psig.

References: FSAR, Section 5.2

16.4 Limiting Conditions for Operation

16.4.1 Core Instrumentation

Applicability: Applies to the operability of the core instrumentation systems.

Objective: To specify functional requirements on the use of the in-core monitoring systems to verify core conditions.

Specification: The incore system shall be operable at the startup of a new core and during power operation to check power distribution. After the power escalation phase at the startup of each new core, the following will be sufficient:

1. Moveable in-core instrumentation operable in one of the hottest instrumented fuel assemblies.

2. Ten radial position thermocouples including those positions as specified in 1.

Basis: Following the attainment of equilibrium xenon, radial power peaking in the Yankee Rowe core decreases monotonically as a function of cycle burnup. This has been verified both by calculation and by measurement on Yankee cores, and is in accord with the expected behavior in a core that does not contain burnable poison. The radial power peaking measured at any time in core life thus provides an upper bound on radial power peaking for the remainder of that cycle. In addition, the fuel exit thermocouples provide a continuous indication of radial power distribution between the times when moveable incore instrumentation maps are made. Thus, increases in F_q can be attained only by virtue of increases in axial peaking, which can be adequately monitored using moveable in-core instrumentation in one of the hottest instrumented fuel assemblies. Because of its small size, the Yankee Rowe core is very tightly coupled and calculations and measurements both indicate only minor variation in F_z between assemblies.

16.4.5 Chemical Shutdown and the Charging and Volume Control System

Applicability: Applies to the operational status of the chemical shutdown and the charging and volume control system.

Objective: To specify those limiting conditions for operation of the chemical shutdown and t e charging and volume control system which must be met in order to ensure adequate boration capability.

Specification: A. Whenever there is fuel in the reactor:

1. The boric acid storage tank or the safety injection storage tank shall contain sufficient boric acid solution to bring the reactor coolant system to the cold shutdown boron concentration. Solution temperatures shall be maintained at least 10°F above the concentration saturation temperature.

2. There shall be at least one operable path for boron injection consisting of system pumps, piping, heat tracing, valves, instrumentation and controls operable as to assure the capability of boron injection.

B. Whenever the reactor is critical, there shall be at least two operable paths from the charging pumps meeting the requirements of A.(2) above.

Exception:

The requirements may be modified to permit operation with but one operable flow path for not more than eight hours in every 24 hours.

Basis: The chemical shutdown and the charging and volume control system provides control of the reactor coolant boron inventory. Reduction of concentration is accomplished by dilution with unborated primary grade water or by boron removal through ion exchange. An increase in concentration may be accomplished by using two of the three charging pumps taking suction from the safety injection tank, or from the boric acid mix and storage tank. The exception permits power operation while performing maintenance on one of the two paths. Full redundancy may be quickly restored should conditions require.

References: FSAR, Sections 9.3.4 and 9.3.5.

16.4.6 Emergency Core Cooling System

Applicability: Applies to the operating status of the core cooling systems.

Objective: To define the conditions under which components of the core cooling system must be operable.

Specification: A. The following equipment must be operable in the manual mode whenever the reactor coolant system temperature and pressure exceed 250°F and 300 psig.

1. The ECCS train, with valves aligned and/or locked in the position required, and controls set for automatic initiation where appropriate consisting of the following components:

    a. One low pressure safety injection pump.

    b. One high pressure safety injection pump.

    c. One emergency diesel generator.

2. The safety injection tank filled and available in accordance with Section 16.4.7.

3. The recirculation system including valves, controls, and one purification pump.

Exception:

The requirements may be modified with regard to the position of controls and valves during periods of hydrostatic testing provided the reactor is not critical.

B. The following equipment must be operable whenever the reactor is in a power operation condition:

1. The low pressure, safety injection accumulator set for automatic initiation. The accumulator shall contain a minimum of 800 cubic feet of water borated to the cold shutdown condition and pressurized with nitrogen between 410 psig and 435 psig.

2. Three operable and redundant ECCS trains, each train consisting of the components specified in A-1 above.

3. The recirculation system including valves, controls and two purification pumps.

Exception:

If any ECCS train or one purification pump becomes inoperative, power operation is permitted for a maximum of 7 days. In this situation, the operable ECCS trains or operable purification pump shall be tested within two hours after discovery of the outage, and once a day for the duration of the outage.

4. The safety injection tank filled and available in accordance with Section 16.4.7.

Basis: Adequate core cooling is provided for the entire break spectrum up to and including the design basis accident. This protection covers all modes of operation from shutdown to full power.

The accident analysis requires two operable ECCS trains, the recirculation system with one purification pump, and the low pressure, safety injection accumulator available for core cooling when at full power.

Specification A provides a pressure and temperature limit above which ECCS must be operable. It recognizes the decreased probability of a loss of coolant accident and the negligible amount of energy stored in the primary coolant under these conditions.

Reference: FSAR, Section 6.3.

16.4.8 Reactor Core Energy Removal

Applicability: Applies to the operating status of plant components for removal of reactor core energy.

Objective: To specify conditions of the plant equipment necessary to ensure the capability to remove energy from the reactor core.

Specification: A. The shutdown cooling system must be in operation except when two or more steam generators are operable for core decay heat removal.

Exception:

The shutdown cooling system may be secured for a period to facilitate special maintenance, refueling functions or tests. During such periods reactor coolant temperatures shall be continuously monitored and initiation of core cooling shall be continuously available.

B. The following conditions must be met for a steam generator to be considered operable for decay heat removal:

1. The reactor coolant system must be closed and pressurized to 100 psi above saturation pressure.

2. The steam generator must have both the cold and hot leg stop valves fully open, with the bypass valve closed.

3. The steam generator water level must be above the top of the tube bundle.

4. An inventory of over 85,000 gallons of primary grade feedwater must be available.

5. A boiler feed pump must be operable.

C. The reactor shall not be in a power operation condition which generates steam at a rate in excess of the on-line steam generator relieving capacity in accordance with Figure 16.4.8-1.

D. The reactor shall not be maintained in a power operating condition unless the following conditions are met to assure post shutdown heat removal capability:

1. The emergency boiler feed pump is operable.

Exception:

If the emergency boiler feed pump becomes inoperable, continued power operation is permitted for a maximum of 4 hours.

2. An inventory of over 85,000 gallons of primary grade feedwater is available.

Basis: Specification A assures that decay heat removal capability is always available.

A steam generator is capable of removing core decay heat by natural or forced circulation provided the conditions specified in B are met.

The 85,000 gallons of water specified is sufficient for a period in excess of twenty-four hours. This is ample time in which to replenish the supply from backup sources of feedwater.

Specification C assures sufficient steam relieving capacity during all modes of power operation.

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as long as feedwater to the steam generators is available. Normally, the capability to supply feedwater to the steam generators is provided by operation of the feedwater system.

In the unlikely event of complete loss of electrical power to the station, decay heat removal is by steam discharge to the atmosphere via the main steam safety valves or the atmospheric steam dump valve. The emergency boiler feed pump can supply sufficient feedwater for removal of decay heat from the plant.

References: FSAR, Chapter 10.

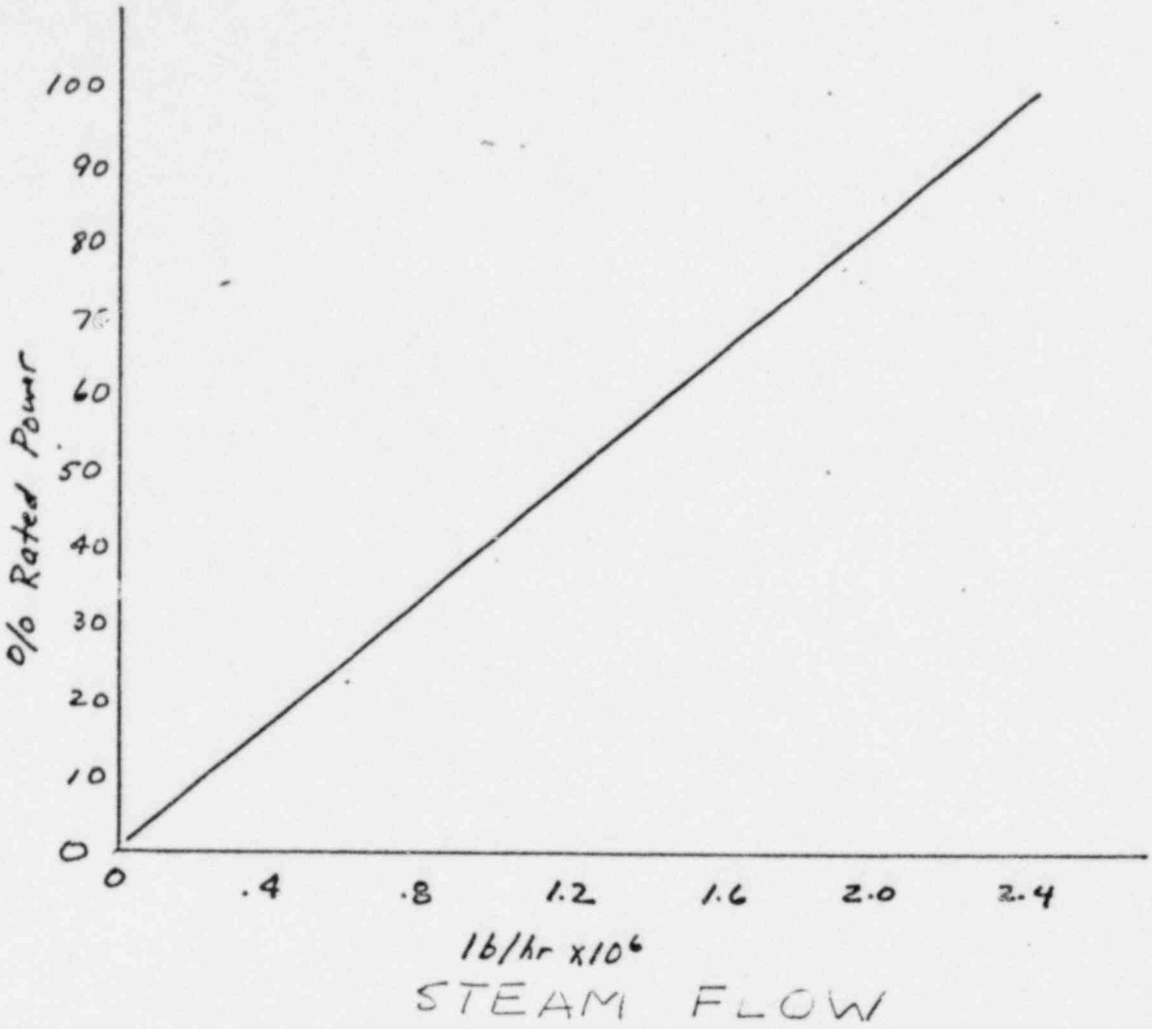


FIGURE 16.4.8-1

DETERMINATION OF MINIMUM STEAM GENERATOR SAFETY VALVE RELIEVING CAPACITY

1. Safety valve capacity must be sufficient to relieve steam at a rate equal to or greater than the flow corresponding to the maximum expected reactor power, as determined from this figure.

2. The available capacity is as follows:

Safety valves 409 A, B, C, and D	118,260 lbm/hr each
Safety valves 409 E, F, G, and H	80,872 lbm/hr each
Safety valves 409 I, J, K, and L	573,329 lbm/hr each

3. The sum of capacities of all valves in service must be equal to or greater than the flow determined in Step 1.

ENCLOSURE 2

REQUEST FOR ADDITIONAL INFORMATION

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

DOCKET NO. 50-29

TECHNICAL SPECIFICATIONS

Our response to your letter dated July 16, 1975 concerning three sections of our proposed Technical Specifications is as follows:

Question No. 1

Specification 16.4.4 relates to temperature and pressure conditions during heat-up and cooldown of the reactor. The temperature-pressure operating limitations for Yankee-Rowe are based on NDT temperature plus 60°F. In order to allow us to determine if the procedure used by you is as conservative as the requirements in 10 CFR Part 50, Appendix G, you should provide the following information:

a. The material basis, including initial RT_{NDT}, copper content and estimated fluence for the limiting material used to calculate the pressure-temperature limit curves.

b. Provide the data points used to obtain Figure 16.4.4-2.

c. Provide the relationship between accumulated reactor output (MWtH) and fluence. The fluence may be taken as the maximum value on the vessel wall at 1/4T thickness.

d. Indicate the material basis for Figures 16.4.4-1 and 2 on these figures.

Response No. 1

The above question is directly related to the application of Appendix G, 10 CFR 50 to Yankee Rowe. Section 16.4.4 was not developed under these rules, therefore, it is prudent at this time to recalculate the Yankee Rowe operating curves using the latest criteria - Appendix G of Section III and Appendix G of 10 CFR 50. It is our preliminary opinion that the results will be less restrictive than currently in force. We anticipate submission of the revised section 16.4.4 on or about December 1, 1975.

Question No. 2

Specification 16.4.14 relates to limiting conditions for operation (LCO) of the reactor under varying rates and conditions of primary coolant leakage. Specification 16.4.14A as proposed requires that at least two independent reactor coolant leakage detection systems shall be in operation when the reactor is at power. It is our position that the leakage detection system also shall be operating when the reactor is critical and at 2 percent power in order to acceptably define the operating power level. Please revise this LCO accordingly. Specification 16.4.14B specifies a 10 gpm limit on primary coolant leakage to uncontained systems. The existing Technical Specifications (Section D.2.d.13) limits this leakage to 6 gpm. Provide an explicit justification for the proposed increase of this leakage limit.

Response No. 2

We have attached revised specification 16.4.14 that incorporates the 2 percent power ($1x10^{-6}$ Amps) requirement. We have also changed the 10 gpm limit on primary coolant leakage to 6 gpm to conform with the existing Technical Specifications.

Question No. 3

Specification 16.4.18 relates to LCO's with respect to oxygen, chloride and fluoride concentration in the primary coolant system. The associated surveillance requirements are included in Table 16.5.2-1 (items 1c and 1f). It is our position that these LCO's and surveillance requirements as proposed do not provide low enough concentration limits for operation of the reactor in the steady state mode. In addition, the sampling frequencies for chemical analysis and the corrective actions as proposed are not acceptable. We have developed a model Specification 16.4.18 (see Enclosure II) that would be acceptable to us for Yankee Rowe. Either confirm that you will adopt this model specification or propose a revised Section 16.4.18 and associated surveillance requirements that include comparable requirements and provide your justification therefor.

Response No. 3

We have revised specification 16.4.18 to incorporate the limits and sampling frequencies of the model specification. A copy of the revised specification is attached.

16.4.14 Primary System Leakage

Applicability: Applies to limiting operation of the plant under varying rates and conditions of primary system leakage.

Objective: To specify primary plant operability with primary system leakage.

Specification: A. When the reactor power is > 2% ($1x10^{-6}$ Amps.), at least two independent reactor coolant leak detection systems shall be in operation. The principle of operation shall differ between them; one of the two being radiosensitivity.

Exception:

The system which operates on the principle of radiosensitivity may be out of service for a period of up to 48 hours, provided two means of detecting reactor coolant leakage are operational.

B. A normal reactor shutdown will be initiated within 12 hours if any of the following limits are exceeded:

1. One gpm of unexplained primary system leakage to the containment atmosphere.
2. Six gpm of primary system leakage to uncontained systems.
3. A steam generator secondary coolant iodine-131 concentration of 0.25 uc/ml.

Basis: Primary system leakage, as limited by Item B.1, refers to reactor coolant leakage to the containment atmosphere and does not include valve leak-off, steam generator tube leakage or leakage to any other contained system.

The limitation of 1 gpm for unexplained leakage is sufficiently above the minimum detectable leakage rate to provide reliable indication, and is low enough to minimize the chance of a progressive failure to an unsafe condition.

Where the source of the leakage can be identified, a leakage rate limit of 6 gpm will be applicable. Any primary system leakage below this limit will be evaluated to determine if the plant can continue to operate in a safe manner. This leakage rate is well below the makeup capability of one charging pump and is within the storage and processing capability of the waste disposal system. Leakage to the contained systems, such as closed collection tanks or the low-pressure surge tank, is not included in this category. It is intended to include leakage to areas where the leaking fluid is not collected in a closed tank or system.

The steam break accident represents the maximum release of secondary activity and forms the basis for determining a maximum allowable secondary coolant activity. The steam break accident is based upon a postulated release of the entire contents of the secondary system to the atmosphere using a site boundary dose limit of 1.5 rem.

The limiting dose for this accident results from iodine in the secondary coolant. The reactor coolant distribution of iodine isotopes with 1% failed fuel was used for this calculation. I-131 is the dominant isotope because of its low MPC in air and because the other iodine isotopes have shorter half-lives and therefore cannot build up to significant concentrations in the secondary coolant, given the limitations on primary system leak rate and activity. The entire secondary system contains approximately 132 m^3 of water at standard conditions. One-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate-out and retention in water droplets.

The maximum inhalation dose at the site boundary is then as follows:

$$\text{Dose (rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot \chi/Q \cdot DCF$$

where: C = secondary coolant sample activity .25 uCi/cc = .25 Ci/m^3

V = water volume in entire secondary system = 132 m^3 at standard conditions

B(t) = breathing rate (3.47×10^{-4} m^3/sec)

$\chi/Q = 5.21 \times 10^{-4}$ sec/m^3

DCF = 1.48×10^6 rem/Ci I-131 inhaled

The resulting dose is less than 1 rem.

Primary system leakage may be determined by one or more of the following methods:

1. Reactor containment air particulate monitoring.
2. Tritium balance within the containment.
3. Relative humidity within the containment.
4. Containment sump level.
5. Periodic visual examinations of all systems for evidence of leakage.
6. Water inventories and balances.
7. Vapor container sound system.

Techniques used to determine primary system to secondary system include the following:

1. Radiochemical analyses for tritium.
2. Analyses for boron.
3. Radiochemical analyses for beta-gamma activity in the steam generator blowdown and/or air ejector discharge.

Primary system leakage will be maintained at the lowest practicable value to facilitate detection and identification of change in leak rate.

Reference: FSAR, Sections 5.2.4, 15.4.2.

16.4.18 Reactor Coolant System Oxygen, Chloride and Fluoride Concentration

Applicability: Applies to the measured maximum oxygen, chloride and fluoride concentrations in the reactor coolant system.

Objective: To ensure that the oxygen, chloride and fluoride in the reactor coolant system do not exceed concentrations detrimental to the functional integrity of the system materials.

Specification:

A. The concentrations of individual contaminants in the Reactor Coolant System shall be maintained within the limits specified in Table 16.4.18-1.

B. The concentration of O_2, F^- and Cl^- shall be determined by chemical analysis at sampling frequencies of at least 5 times per 7 days with a maximum of 48 hours between samples.

C. With any one or more contaminant in excess of its Steady State Limit but within its Transient Limit, restore the contaminant concentration to within its Steady State Limit within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

D. With any one or more contaminant concentration in excess of its Transient Limit, be in COLD SHUTDOWN within 36 hours.

Basis: By maintaining the oxygen, chloride and fluoride concentration in the reactor coolant within the limits as specified above, the functional integrity of the materials in the Reactor Coolant System is assured under all operating conditions.

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the low pressure surge tank, and further because of the time dependent nature of any adverse effects arising from oxygen, chloride and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the period of 24 hours for corrective action to restore the concentrations within the Steady State Limits has been established. If the corrective action has not been effective at the end of the 24 hour period, then the reactor will be brought to a Cold Shutdown condition within the next 36 hours while continuing the corrective action. If one or more contaminant concentration exceeds the Transient Limits the plant will be brought to a Cold Shutdown condition within 36 hours while continuing corrective action.

References: FSAR, Chapter 5 and Section 9.3.

TABLE 16.4.18-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

CONTAMINANT	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN	$\leq$ 0.10 ppm*	$\leq$ 1.00 ppm*
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

*Limit not applicable with $T_{avg} \leq 250^{o}F$.

ENCLOSURE 3

YANKEE NUCLEAR POWER STATION (YANKEE ROWE)

Docket No. 50-29

TECHNICAL SPECIFICATIONS

The following specifications have been revised to reflect changes in plant conditions since their submittal January 3, 1974, reference 2. Copies of the revised specifications are attached.

Specification 16.5.6 Periodic Testing

We have changed the low pressure safety injection pump flow from 2480 gpm to 2180 gpm to reflect the actual flow rate determined by test. Our letter to the AEC, dated December 11, 1974 described the circumstances necessitating the revision.

We have also changed the minimum monthly operation time of the V.C. Recirculation Pumps from 10 minutes to 30 minutes to conform to the requirements of our Change 95, approved November 4, 1970.

Specification 16.5.4 Containment Testing

We have revised this specification to reflect the provisions of 10 CFR 50.54 (o), Appendix J as dated February 14, 1973. Present Technical Specifications, Appendix C, paragraph C.5 permit testing approximately 25% of the electrical penetrations in each Class B test. On June 12, 1975 we requested an exemption from testing 100% of the electrical penetrations on an annual basis. In our letter dated December 12, 1973 we requested exemption from the testing of the equipment hatch because it has no local testing facilities and the personnel air lock after each use because it is used so frequently for vapor containment inspections. These exemptions were granted by an AEC letter from Donald J. Skovholt dated January 14, 1974. The continuous leak monitoring system effectively accomplishes the intent of Appendix J for the equipment hatch, air lock and electrical penetrations.

16.5.6 Periodic Testing

Applicability: Applies to the safety injection system; the steam generator emergency boiler feed system; the chemical shutdown system; and the vapor container recirculation system.

Objective: To verify that the above systems will perform their intended functions.

Specification: A. Safety Injection System

1. The following tests will be performed during each month of plant operation:

    a. Safety injection system pumps:

    The safety injection pumps shall be tested by operating in the recirculation mode.

    Acceptable performance shall be that pumps operate for at least 5 minutes, and that the associated instrumentation and controls function properly.

    b. Safety injection valves:

    All automatically operated valves that are required to operate for core flooding shall be exercised through their full travel. All locked valves shall be visually checked to verify proper operating position.

2. The following tests will be performed at each refueling:

    a. Safety injection pumps:

    Two high pressure safety injection pumps shall be flow tested at 84 feet discharge head, and shall pump at least 500 gpm.

    Two low pressure safety injection pumps shall be flow tested at 84 feet discharge head, and shall pump at least 2180 gpm.

    b. Safety injection valves:

    All automatically operated valves shall be exercised through their full travel.

B. Steam Generator Emergency Boiler Feed Pump

During normal plant operation the emergency boiler feed pump shall be tested monthly to demonstrate its operability in the recirculation mode.

C. Chemical Shutdown System

During normal plant operation a monthly test shall be conducted to verify the flow of concentrated boric acid from the boric acid mix tank to the charging pump suction header.

D. V.C. Recirculation System

1. The following tests shall be performed during each month of plant operation:

   a. The pumps shall be operated a minimum of 30 minutes to demonstrate their operability.

   b. The motor operated valves in the system shall be operated remotely from the control room. Remote position indication shall be verified.

2. One pump will be tested at each refueling. Acceptable performance will be that the pump shall discharge at a minimum rate of 100 gpm at a head of 26 psig.

Basis: The safety injection system and the vapor container recirculation system are principal plant safeguards systems that are normally operable during reactor operation.

Complete system tests cannot be performed when the reactor is operating because of their inter-relation with operating systems. The method of assuring operability of these systems is a combination of complete system tests performed during refueling shutdowns; and monthly tests of active system components (pumps and valves) during reactor operation. The test interval is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required).

The monthly testing interval of the steam generator emergency boiler feed pump verifies its operability by recirculating water to the demineralized water tank. Proper functioning of the steam turbine and the emergency boiler feed pump will be made by direct visual observation.

B. Steam Generator Emergency Boiler Feed Pump

During normal plant operation the emergency boiler feed pump shall be tested monthly to demonstrate its operability in the recirculation mode.

C. Chemical Shutdown System

During normal plant operation a monthly test shall be conducted to verify the flow of concentrated boric acid from the boric acid mix tank to the charging pump suction header.

D. V.C. Recirculation System

1. The following tests shall be performed during each month of plant operation:

   a. The pumps shall be operated a minimum of 30 minutes to demonstrate their operability.

   b. The motor operated valves in the system shall be operated remotely from the control room. Remote position indication shall be verified.

2. One pump will be tested at each refueling. Acceptable performance will be that the pump shall discharge at a minimum rate of 100 gpm at a head of 26 psig.

Basis: The safety injection system and the vapor container recirculation system are principal plant safeguards systems that are normally operable during reactor operation.

Complete system tests cannot be performed when the reactor is operating because of their inter-relation with operating systems. The method of assuring operability of these systems is a combination of complete system tests performed during refueling shutdowns; and monthly tests of active system components (pumps and valves) during reactor operation. The test interval is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required).

The monthly testing interval of the steam generator emergency boiler feed pump verifies its operability by recirculating water to the demineralized water tank. Proper functioning of the steam turbine and the emergency boiler feed pump will be made by direct visual observation.

Specification "C" assures that the chemical shutdown system lines are free, and in condition to initiate an emergency boration if required.

The arrangement of components in the recirculation system and the testing performed assure the capability for long term cooling in the event of a loss-of-coolant accident.

16.5.4 Containment Testing

Applicability: Applies to reactor containment leakage testing.

Objective: To verify the integrity of the reactor containment.

Specifications: Reactor containment testing will be performed in full accordance with the provisions of 10 CFR 50.54, (o) Appendix J as published February 14, 1973, with the following Type B exemptions:

1. Local testing of the equipment hatch.
2. Personnel air lock after each use.
3. Annual testing of all electrical penetrations.

Basis: A leakage value of 0.2% per 24 hours at the calculated peak accident pressure of 31.6 psig will under the most adverse accident conditions maintain public exposure below 10 CFR 100 values in the event of the hypothetical accident (see Section 15.4.5 FSAR). Tests at the containment reduced test pressure of 16 psig will assure the continued ability of the containment to perform its function. The continuous leak monitoring system has been accepted by the Commission as an acceptable substitute for local testing of the equipment hatch and testing after each use of the personnel hatch. Due to the number and difficulty of testing electrical penetrations 25% will be tested annually.

References: FSAR, Sections 15.4.5 and 15.4.6.

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