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T.F. B7.1.2

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

Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

References: (a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated April 28, 1982, "Request for Additional Information - Procedures and Test Review Branch", F. J. Miraglia to W. C. Tallman

Subject: Response to 640 Series RAIs; (Procedures and Test Review Branch)

Dear Sir:

We have enclosed responses or revised responses to the following 640 Series Requests for Additional Information which were forwarded in Reference (b):

640.5, 640.8, 640.27, 640.33, 640.35, 640.51(4.t)

The enclosed responses will be incorporated in OL Application Amendment 49.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. DeVincentis

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Project Manager

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ALL/fsf

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(1)

640.5 Regulatory Guide 1.140, Design Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants (page 1.8-54).

The following exceptions to this Regulatory Guide are not justified:

(1) C.2.b - Either reduce the total system flow rate specification to approximately 30,000 ft^3/min or provide technical justification that will assure the staff that the higher flow rate will provide the same operational efficiencies.

(2) C.2.c - Regulatory Guide 1.140 specifies ANSI N510 - 1975 "Testing of Nuclear Air Cleaning Systems: and ANSI/ASME N509-1976, "Nuclear Power Plant Air Cleaning Units and Components", as the standards for the design and testing of atmospheric clean-up systems.

ERDA 76-21 as referred to in C.2.c outlines operational standards for the systems. Modify your position to conform to the monitoring requirements of ERDA 76-21.

RESPONSE: We are revising Section 14.2.7 to add Regulatory Guide 1.140, Rev. 1, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Pumps".

Seabrook Station is in conformance with Regulatory Guide 1.140 except for the following:

(1) One atmospheric clean-up system exceeds the air flow rate of 30,000 ft^3/min listed in C.2.b. The operational efficiency will not be changed since the unit has been sized to handle the increase in air flow rate, while keeping the system velocity the same as that found in a unit capable of handling 30,000 ft^3/min. Provisions will be made to provide an air sampling system to insure that a representative gas sample is taken during testing.

(2) Each filter bank has a differential pressure gauge to read pressure drop and a differential pressure switch across the entire filtration system. All non-safety systems are equipped with flow switches which indicate flow/no flow and alarm on a low flow condition. These switches are located on the fan discharge. The fan status is monitored by lights located on the main control board. All safety systems measure volumetric flow.

(2)

Revised 2/83

RAI 640.8

We have noted on other plant startups that the capacities of pressurizer or main steam relief valves and turbine bypass valves are sometimes in excess of the values assumed in the accident analysis for inadvertent opening or failure of these valves. Provide a description of the testing that demonstrates that the capacity of these valves is consistent with your accident analysis assumptions.

Response:

The accident analysis assumptions do not assume values for the pressurizer or main steam relief valves or the turbine bypass valves. The accident analysis is based on the capacities of the pressurizer safety and steam generator safety valves. The capacities of the safety valves greatly exceed that of the relief valves or turbine bypass valve.

The pressurizer safety valve is sized to relieve approximately twice the steam flow rate of a relief valve (15.6.1.1). The pressurizer safety valves are Crosby Model 6M6 valves with a nozzel bore diameter of 2.154 inches (bore area of 3.644 sq. inches). Test data for these valves may be found in the EPRI PWR Safety and Relief Valve Test Program, Safety and Relief Valve Test Report, EPRI NP-2628-LD, interim report, September 1982. The pressurizer relief valves are Garrett electric solenoid valve (part number 3750014) with a limiting flow area (seat) of 1.584 sq. inches. Test data for these valves may be found in the EPRI/Wyle Power Operated Relief Valve, Phase III, Test Report, Volume II, EPRI NP-2670-LD, interim report, October 1982.

Main steam safety valves have steam capacities ranging from 893,200 lbs./hr. at set pressure of 1185 psig (1200 psia) to 945,300 lbs./hr. at 1255 psig (10.3.2.6). The main steam relief valve capacity is 400,000 lbs./hr. at 1135 psig (10.3.2.4). The turbine bypass (steam dump to condenser) valve capacity is 510,000 lbs./hr. at 1107 psia (10.4.4.2). These capacities are based on manufacturer's standard ratings based on generic testing. These steam flow capacities do not exceed the maximum recommended by the NSSS supplier (ref. WCAP-7451, Rev. 2). The accident analysis for inadvertent opening of a single steam dump, relief or safety valve (15.1.4.2) uses a steam flow rate of 268 lbs./sec. at 1200 psia. This is equivalent to 964,800 lbs./hr. at 1200 psia. It is noted that this is the maximum assured flow, at initial conditions, which decreases during the accident as the steam pressure falls.

Since the accident analysis is based on a substantially higher steam flow capacity than the relief or bypass valves are capable of providing, no testing is planned to demonstrate the capacity of these valves.

(3)

640.27 Include a description of the test(s) (Table 14.2-3) that will be performed to ensure conformance to Regulatory Guide 1.95 Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release.
(NOTE: for a Type 1 control room, refer to Positions C.1, C.2, C.3a and C.4-6)

<u>RESPONSE</u>: As explained in section 1.8 the plant design does not include the storage of chlorine within 100 meters of the control room, excluding small laboratory quantities, nor is there chlorine stored in excess of the maximum allowable chlorine inventory.

Control room ventilation testing will be performed (Table 14.2-3 item 28). This testing shall include a demonstration that the control room envelope boundary seals to maintain the required positive pressure.

This testing will also demonstate the following regulatory positions of RG 1.95 are satisfied.

a. Manual isolation of the control room is provided (RG 1.95, C.2).

b. The gross leakage characteristic of the control room is determined by pressurizing the control room to 1/8 - inch water gage and measuring the pressurization flow rate (RG 1.95, C.5).

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640.33 (14.2.7) Conformance of Test Programs with Regulatory Guides, Regulatory Guide 1.20-Rev. 2 (Page 14.2-5). Regulatory Guide 1.20 requires analysis and either extensive measurements or full inspection for a non-prototype Category I System. Subsection 3.9(N).2.4 of the FSAR refers to inspection of reactor internals. Provide additional discussion of Regulatory Guide 1.20-Rev. 2. Modify Sections 1.8 and 14.2.7, accordingly.

Response: Seabrook FSAR Section 3.9(N).2.4 clearly and completely provides the information necessary to properly assess the type and degree of inspections to be performed on the Seabrook reactor internals with respect to Regulatory Guide 1.20, Rev. 2. The recommendations of Regulatory Guide 1.20 will be met by conducting the confirmatory preoperational testing examination for integrity. The information provided in Seabrook FSAR Section 3.9(N).2.4 on the preoperational flow-induced vibration testing of reactor internals has previously been found acceptable by the NRC Mechanical Engineering Branch. Additional discussion of Regulatory Guide 1.20 is not considered necessary.

(5)

Revised

640.35 Conformance to Regulatory Guide 1.68 (Revision 2), Item (2). It is the staff position that the requirements for systems relied on to prevent, limit or mitigate the consequences of postulated accidents be completed prior to exceeding 25% power. Modify Sections 14.2.7, 14.2.11 and the appropriate test abstracts accordingly.

RESPONSE: FSAR SECTIONS 14.2.7 AND 14.2.11 WILL BE REVISED TO CLARIFY THE POWER ASCENSION TEST SCHEDULE.

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voltage study to verify the adequacy of the analytical model. (Appendix A, Section 1.g.2). 45

2. ~~During power escalation, testing will be performed at approximately 30% rather than the 25% power plateau. Westinghouse supplied plants have historically conducted tests at 30% and, therefore, generic data is available for review. (Section C.8; Appendix A, Section 5.)~~

INSERT A

3. Throughout core loading and precritical tests, the shutdown margin will be verified by periodic sampling of core coolant and verification that boron concentration is maintained at or above the Technical Specification concentration limit for refueling conditions. (Appendix A, Section 2.a)

4. Control rod runback and partial scram features are not used in the Seabrook Station design and, therefore, will not be tested during power escalation. (Appendix A, Section 5.j.)

5. A demonstration of the capability of systems and components to remove residual heat or decay heat from the Reactor Coolant System will be performed during power ascension testing only if not performed during hot functional or low power tests. (Appendix A, Section 5.1.)

6. The failed fuel detection system is not applicable to the Seabrook design and, therefore, will not be tested during power escalation. (Appendix A, Section 5.q.)

7. The integrated control system and the reactor coolant flow control system are not applicable to the Seabrook Station design and therefore, will not be tested during power escalation. The Startup and Emergency Feedwater Control Systems and the Steam Pressure Control Systems are only used in the hot shutdown, hot standby or low power operating modes. These systems can not be tested during power ascension. (Appendix A, Section 5.s.)

8. A demonstration of the dynamic response of the plant to a loss of or bypassing of a feedwater heater(s) will not be performed. In lieu of the tests, an analysis will be performed to determine the dynamic plant response to a single failure or operator error affecting feedwater heaters that would result in the most severe case of feedwater temperature reduction. Since plant response to power swings and plant trips at various power levels is demonstrated in other tests, and the protection system setpoints are conservatively chosen to prevent exceeding established parameters, there is no basis for performing this test and subjecting the plant equipment to an additional unnecessary transient. (Appendix A, Section 5.k.k.)

9. Any simultaneous MSIV closure test will not be performed. Since the simultaneous MSIV closure is an analyzed event and proper dynamic response of the plant will be demonstrated during other 44

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2. During the power ascension testing phase, tests will be scheduled such that the safety of the plant will not be dependent on the performance of an untested system or feature. Power ascension testing will be performed at power plateaus of approximately 30%, 50%, 75% and 100%. It is required that testing be performed at 30% rather than 25% because individual system stability is increased at 30% (e.g. feedwater system), this allows comparison steady-state conditions with the design at low power. Westinghouse-supplied plants have historically conducted tests at 30% and, therefore, generic data is available for review and comparison.

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Written procedures specify the plant conditions, precautions and specific instructions for the approach to criticality and for limiting the period to more than thirty (30) seconds once criticality is achieved.

14.2.11 Test Program Schedule 44

The initial test program will consist of a preoperational test phase and a startup test phase. The preoperational phase of testing of individual plant systems will begin after construction of the system is essentially complete and construction verification tests (hydrostatic tests, control circuits checks, etc.), system flushing, and preliminary system operational checks (instrument calibration, pump and motor operation, valve checks, etc.) are complete. Each system preoperational or acceptance test will demonstrate, to the extent practical, the ability of the system and equipment to perform its design function in accordance with FSAR requirements.

The individual system preoperational and acceptance tests will proceed concurrently as individual system construction and preliminary testing is completed. When the appropriate systems have been turned over to the station staff, integrated system preoperational tests are performed. The principal milestones during this phase of the program are expected to be the reactor vessel hydrostatic test and integrated hot functional tests. Tests of other systems will be scheduled as appropriate to support these events. Section 14.2.11 provides more detailed information on each test which will be performed during the preoperational phase of the program.

The startup test phase will commence at initial fuel loading. Initial fuel loading and initial criticality are discussed in Subsection 14.2.9. Subsequent to initial criticality, low power reactor physics tests are performed. During these tests, measurements will be performed to verify the calculated values of control rod bank reactivity worths, isothermal temperature coefficients, and differential boron concentrations as a function of control rod configuration.

When the reactivity control characteristics of the reactor have been verified by the low power tests, a program of power level escalation will bring the unit to its full rated power level. During the power escalation, predetermined tests are conducted to verify that the reactor and unit perform as expected at 30%, 50%, 75% and 100% power. Tests are scheduled such that the safety of the plant will not be dependent on the performance of an untested system or feature. ~~and that plant equipment and structures that are relied upon to prevent, mitigate, or limit the consequences of an accident are tested prior to exceeding 30% power.~~ Subsection 14.2.11 provides more detailed information on each test that will be performed during low power testing and power escalation.

Figure 14.2-1 depicts the projected schedule of the initial test program for each unit showing major milestone events, expected time of performance, and duration referenced to fuel loading. The periods for major activities in the procedure preparation, staffing and training areas are also shown with respect to testing activities. It should be noted that the major staffing and training effort applies only to unit 1, since there is a difference of 44

(9)

Revised

RAI 640.51

(4.t) Performance of natural circulations tests of reactor coolant system to determine that design heat removal capability exists. Your natural circulation test should comply with our letter to you dated June 12, 1981. We suggest you contact Westinghouse in reference to the Westinghouse letter to the NRC dated July 8, 1981, on the subject of Special Low Power Test Program which complies with the staff position on TMI-2 Action Item I.G.1 requirement. To comply with this series of natural circulation tests. To date, such tests have been performed at the Sequoyah 1, North Anna 2, and Salem 2 facilities. Based on the success of the programs at these plants, the staff has concluded that augmented natural circulation training should be performed for all future PWR operating Licenses. Included description of natural circulation tests that fulfill the following objectives:

Testing

The tests should demonstrate the following plant characteristics: Length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation with or without onsite and offsite power, the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation, and subcooling monitor performance.

Training

Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should participate in the initiation, maintenance and recovery from natural circulation mode. Operators should be able to recognize when natural recirculation has stabilized, and should be able to control saturation margin, RCS pressure; and heat removal rate without exceeding specified operating limits. If these tests have been performed at a comparable prototype plant, they need be repeated only to the extent necessary to accomplish the abovt training objectives and to obtain data for "fine tuning" your simulator (as stated in FSAR Subsection 13.2.1.1.b.5) for natural circulation operation.

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Response

The natural circulation testing to be performed at Seabrook Station, conforms to the program outlined in the Westinghouse Letter to the NRC, dated July 8, 1981. The details of the Seabrook program as as follows:

1. During hot functional testing, Heat Removal Demonstration, will be performed. This test will use one or more reactor coolant pumps as the heat input to the secondary system and will control the plant using the steam driven emergency feedwater pump and the steam generator atmospheric relief valves. This test will demonstrate the ability to maintain plant conditions without using on-site or off-site AC power.

2. At hot no-flow conditions (in conjunction with rod drop testing), the pressurizer heaters will be turned off and a portion of ST-22, Natural Circulation Test, will be used to collect data and determine a depressurization rate.

3. Performance of ST-39, Station Blackout Test, will demonstrate the ability to maintain the plant at stable natural circulation conditions without off-site AC power.

4. The Natural Circulation Test, ST-22, will be used to demonstrate the natural circulation characteristics of Seabrook Station. This test will determine the length of time necessary to stabilize natural circulation and will demonstrate the reactor coolant flow distribution by obtaining in-core thermo-couple and fixed incore flux detector maps. Auxiliary spray will be used to partially depressurize the primary plant, and the depressurization rate will be determined. At reduced pressure the effect of changes in charging flow and steam flow on subcooling margin will be exhibited and subcooling monitor performance will be verified. During performance of ST-22, a target of 50% of the available licensed operators will witness the test. Additional data will be collected to verify the ability of the Seabrook (site-specific) Simulator to accurately depict natural circulation.

5. After the natural circulation performance of the Seabrook Simulator has been verified, all operators will receive hands-on training (at the simulator) in establishing, recognizing, and maintaining natural circulation conditions, including boration and cooldown operations. Where applicable, data from