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Dockets Nos. 50-269, 50-270
and 50-287

RECEIVED FEB 06 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. William O. Parker, Jr.
Vice President, Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 92, 92, and 89 for Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to each license and to the Station's common Technical Specifications (TSs) and are in response to your applications dated May 21, 1979; October 2, 1980, as supplemented October 30, 1980; and October 20, 1980.

These amendments: 1) revise the TSs regarding the high pressure trip setpoint and the pressurizer power operated relief valve setpoint; 2) add three license conditions and additional TSs which incorporate certain of the Three Mile Island Unit No. 2 Lessons Learned Category "A" requirements; and 3) revise the TSs to include an additional portion of Regulatory Guide 1.16 in the reporting requirements.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Robert W. Reid
Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 92 to DPR-38
2. Amendment No. 92 to DPR-47
3. Amendment No. 89 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

cc w/enclosures: See next page

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

January 28, 1981

Dockets Nos. 50-269, 50-270
and 50-287

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Vice President, Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
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These amendments: 1) revise the TSs regarding the high pressure trip setpoint and the pressurizer power operated relief valve setpoint; 2) add three license conditions and additional TSs which incorporate certain of the Three Mile Island Unit No. 2 Lessons Learned Category "A" requirements; and 3) revise the TSs to include an additional portion of Regulatory Guide 1.16 in the reporting requirements.

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Robert W. Reid, Chief
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5. Notice of Issuance

cc w/enclosures: See next page

Duke Power Company

cc w/enclosure(s):

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County Supervisor of Oconee County
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5/21/79, 10/2, 10/20 & 10/30/80

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-38

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

   A. The applications for amendment by Duke Power Company (the licensee) dated May 21, 1979; October 2, 1980, as supplemented October 30, 1980; and October 20, 1980, comply 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 applications, 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-38 is hereby amended by revising paragraph 3.B. and adding paragraphs 3.H., 3.I., and 3.J. as follows and by changing the Technical Specifications as indicated in the attachment to this license amendment:

   3.B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 92 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8102200 184

3.H. Systems Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

3.I. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

3.J. Backup Method for Determining Subcooling Margin

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
License No. DPR-47

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

   A. The applications for amendment by Duke Power Company (the licensee) dated May 21, 1979; October 2, 1980, as supplemented October 30, 1980; and October 20, 1980, comply 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 applications, 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-47 is hereby amended by revising paragraph 3.B. and adding paragraphs 3.H., 3.I., and 3.J. as follows and by changing the Technical Specifications as indicated in the attachment to this license amendment:

   3.B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 92 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3.H. Systems Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

3.I. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

3.J. Backup Method for Determining Subcooling Margin

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-55

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

   A. The applications for amendment by Duke Power Company (the licensee) dated May 21, 1979; October 2, 1980, as supplemented October 30, 1980; and October 20, 1980, comply 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 applications, 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-55 is hereby amended by revising paragraph 3.B. and adding paragraphs 3.H., 3.I., and 3.J. as follows and by changing the Technical Specifications as indicated in the attachment to this license amendment:

   3.B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3.H. Systems Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

3.I. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

3.J. Backup Method for Determining Subcooling Margin

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 28, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 92 TO DPR-38
AMENDMENT NO. 92 TO DPR-47
AMENDMENT NO. 89 TO DPR-55

DOCKETS NOS. 50-269, 50-270 AND 50-287

Revise Appendix A as follows:

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--	3.1-24
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3.5-2	3.5-2
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3.5-4	3.5-4
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--	3.5-5a
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2.2 SAFETY LIMITS - REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1967.

Bases

The reactor coolant system [1] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110% of design pressure.[2] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under USAS Section B31.7 is 110% of design pressure. Thus, the safety limit[2] of 2750 psig (110% of the 2500 psig design pressure) has been established. The settings, the reactor high pressure trip (2300 psig) and the pressurizer safety valves (2500 psig)[3] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the Reactor Coolant pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

level trip and associated reactor power/reactor power-imbalance boundaries by 1.08% - Unit 1 for 1% flow reduction.
1.08% - Unit 2
1.08% - Unit 3

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2300 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure (11.14 T_{out}-4706) trip
(1800) psig (11.14 T_{out}-4706)
(1800) psig (11.14 T_{out}-4706)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

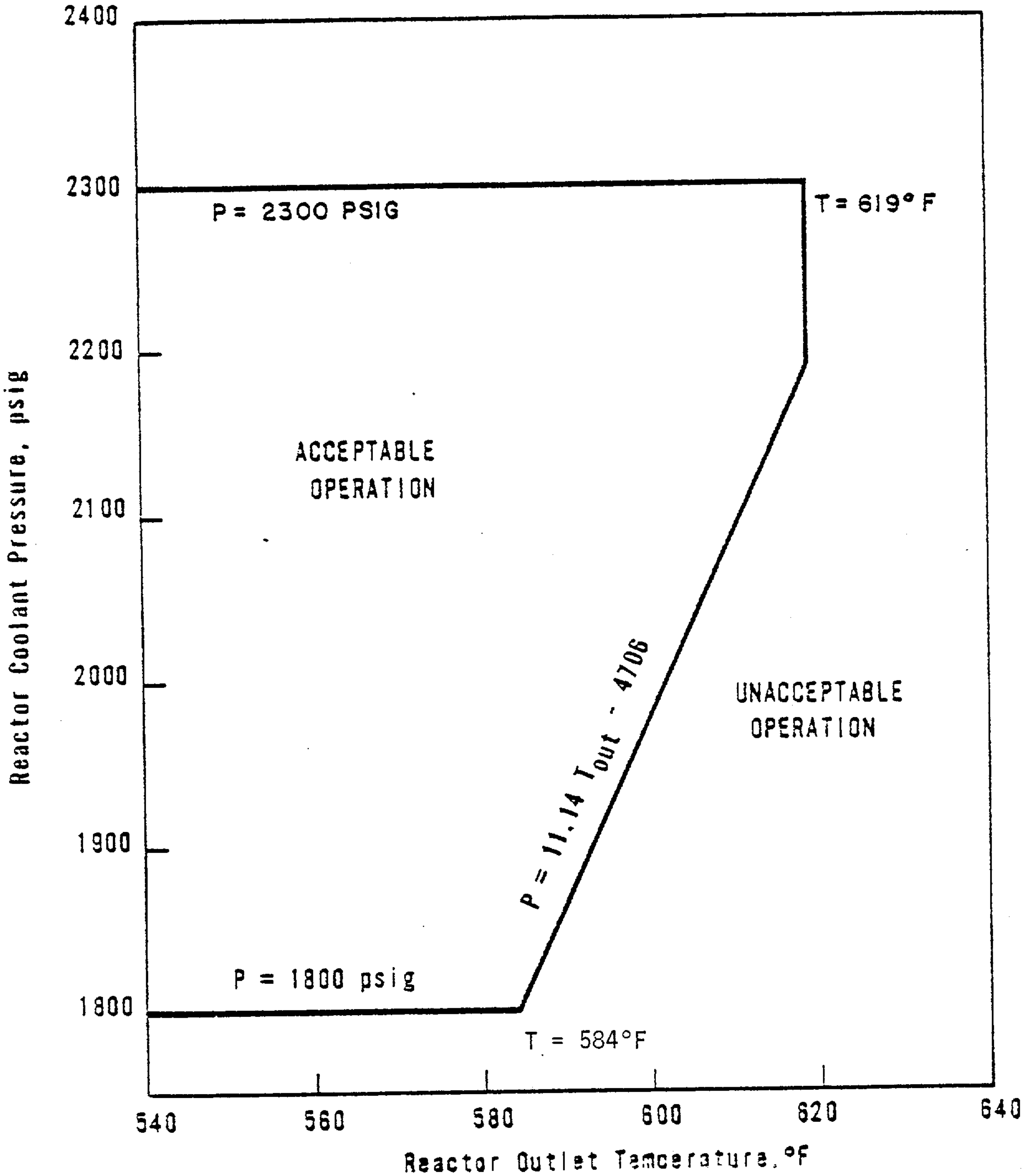
Due to the calibration and instrumentation errors, the safety analysis used a variable low reactor coolant system pressure trip value of (11.14 T_{out} - 4746)
(11.14 T_{out} - 4746)
(11.14 T_{out} - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619°F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620°F.

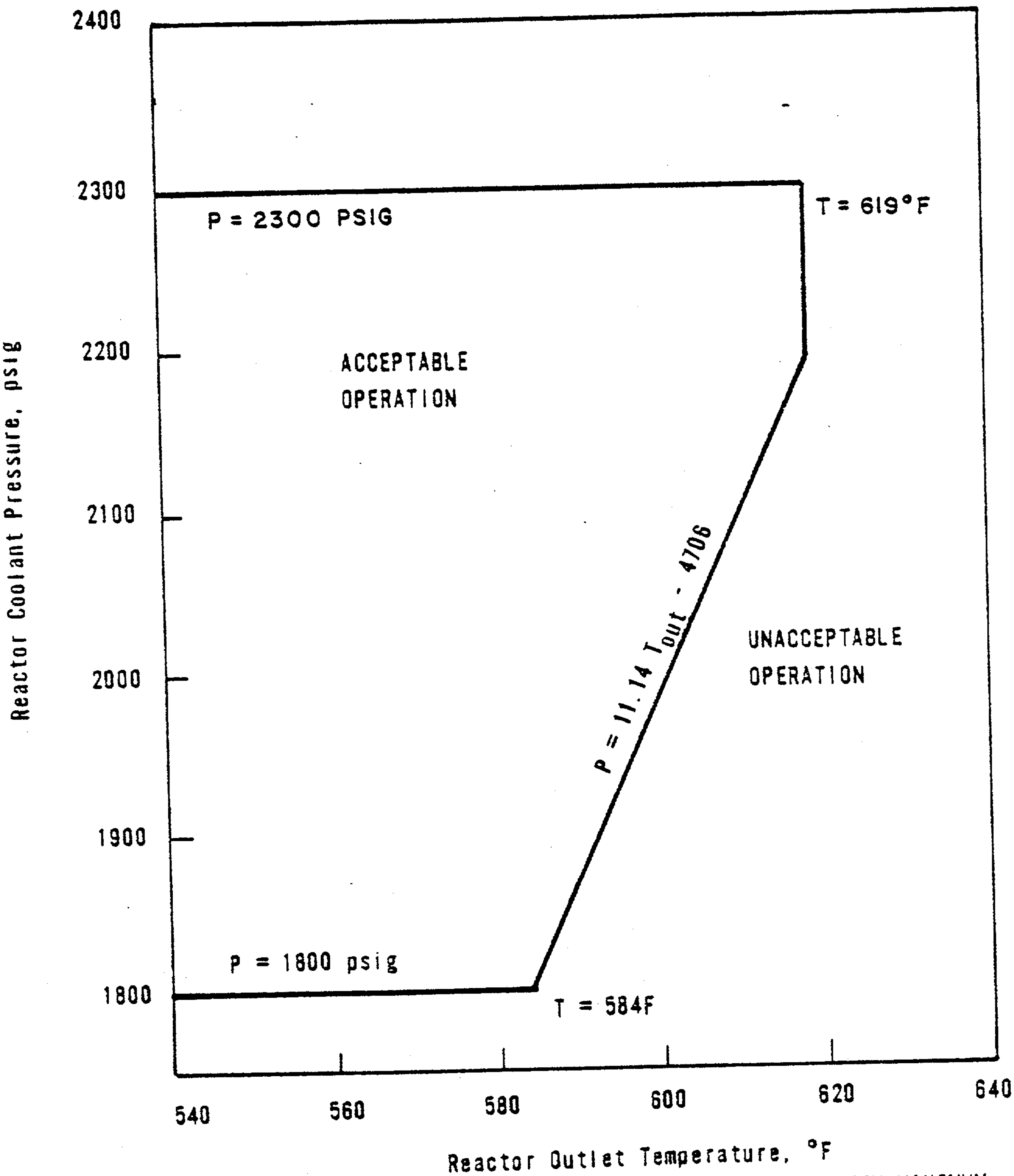
Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 1

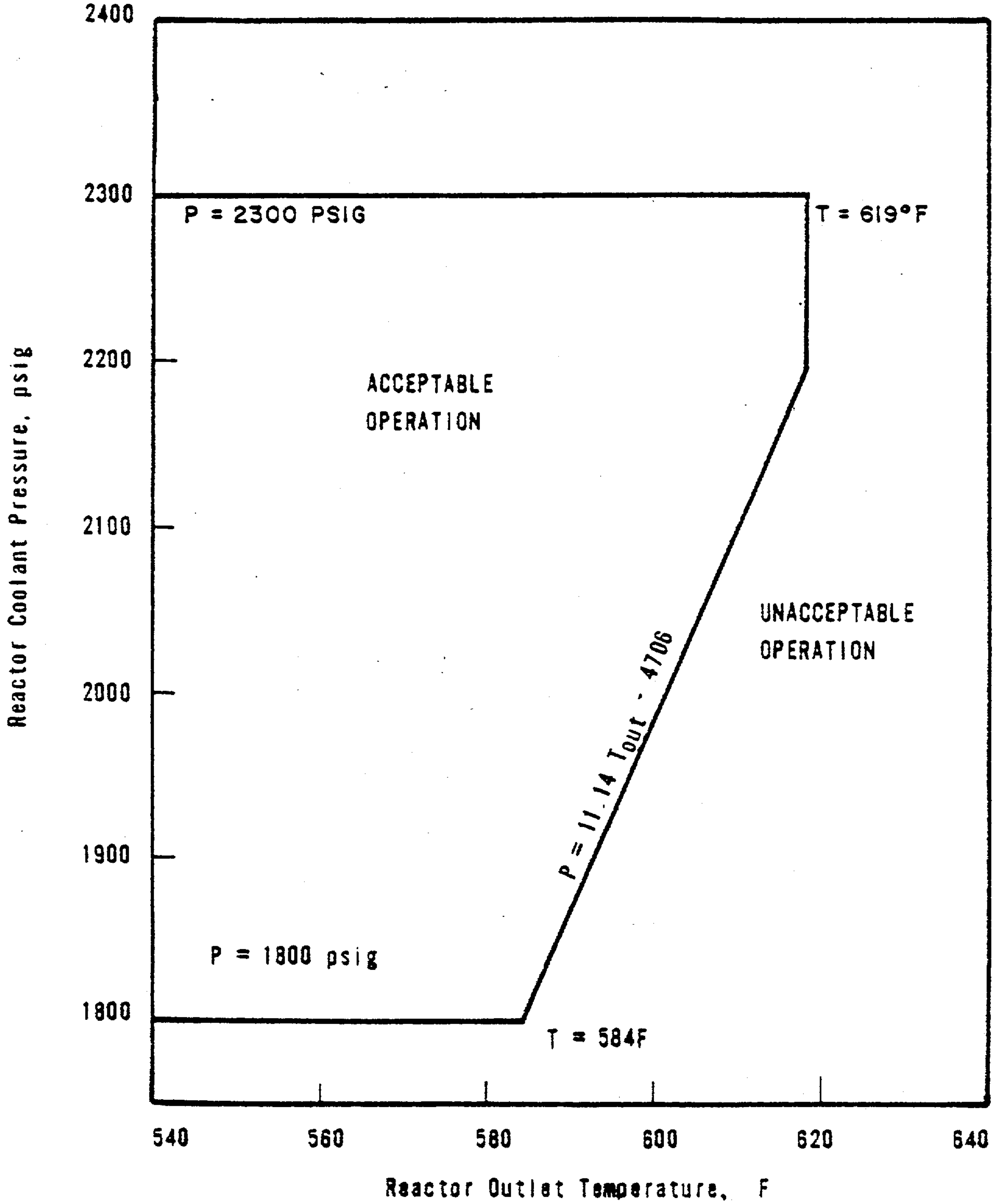
OCONEE NUCLEAR STATION
Figure 2.3-1A



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.3-1B



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.3-1C

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power 100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating In Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14T_{out}-4706)^{(1)}$	$(11.14T_{out}-4706)^{(1)}$	$(11.14T_{out}-4706)^{(1)}$	Bypassed
7.	Reactor Coolant Temp. F., Max.	619	619	619	619
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (°F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

Table 2.3-1B
Unit 2

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	Bypassed
7.	Reactor Coolant Temp. F., Max.	619	619	619	619
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (°F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

Table 2.3-1C
Unit 3

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig. Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure, psig, Min.	$(11.14\ T_{out}-4706)$(1)	$(11.14\ T_{out}-4706)$(1)	$(11.14\ T_{out}-4706)$(1)	Bypassed
7.	Reactor Coolant Temp. F., Max.	619	619	619	619
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (°F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

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3.1.12 Reactor Coolant System Subcooling Margin Monitor

Specification

3.1.12.1

a. The Reactor Coolant System subcooling monitors shall be operable when the average RCS coolant temperature is above 300°F.

b. If one monitor is inoperable, the monitor shall be restored to operable status within seven days or the unit shall be in hot shutdown within the next 12 hours.

c. If both of the subcooling monitors are inoperable because of an outage of the operational computer, and the computer is out of service for less than four hours, and the backup method for determining subcooling margin is available, then a capability to determine subcooling margin is available and a report pursuant to Specification 6.6.2 is not required.

d. If both of the subcooling monitors are inoperable, then restore at least one monitor to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

Bases

The operability requirements of the Reactor Coolant System subcooling margin monitors ensures that sufficient information is available to the operators to provide prompt recognition of saturated conditions in the primary coolant system and advanced warning of the approach to inadequate core cooling. Guidance for these requirements was provided by the NRC letter of July 2, 1980, and derived from the implementation of the TMI-2 lessons learned program.

3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

Specification

3.4.1 Emergency Feedwater System

The reactor shall not be heated above 250°F unless the following conditions are met:

a. Three emergency feedwater pumps (one steam-driven pump capable of being powered from an operable steam supply system and two-motor-driven pumps), and associated initiation circuitry, shall be operable.

b. Two 100% emergency feedwater flow paths shall be operable. Each flow path shall have at least one flow indicator operable.

c. If one emergency feedwater pump or emergency feedwater flow path is inoperable then, restore it to operable status within 60 hours. Otherwise, the unit shall be in a hot shutdown condition within an additional 12 hours and below 250°F in another 12 hours.

3.4.2 The 16 steam system safety valves shall be operable.

3.4.3 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hotwell.

3.4.4 The emergency condenser circulating water system shall be operable.

3.4.5 The controls of the emergency feedwater system shall be independent of the Integrated Control System.

Bases

The Main Feedwater System and the Turbine Bypass System are normally used for decay heat removal and cooldown above 250°F. Feedwater makeup is supplied by operation of a hotwell pump condensate booster pump and a main feedwater pump.

The Emergency Feedwater (EFW) System assures sufficient feedwater supply to the steam generators of each unit, in the event of loss of the main Feedwater System, to remove energy stored in the core and primary coolant. The EFW System is designed to provide a sufficient secondary side steam generator heat sink to enable cooldown from reactor power operation down to cold shutdown conditions.

A 100% emergency feedwater flowpath shall be considered to be either: 1) the steam-driven turbine pump, associated valves and piping capable of feeding either steam generator or 2) both motor-driven pumps, associated valves and piping each capable of feeding the associated steam generator.

One flow indicator or steam generator level indicator per steam generator is sufficient to provide indication of emergency feedwater flow to the steam generators and to confirm emergency feedwater system operation. In the event that at least one indicator per steam generator is not available, then the flowpath to this steam generator is considered to be inoperable.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps or low main feedwater header pressure. The EFW System will supply sufficient feedwater for approximately five-hour cooldown at a flowrate of at least 720 gpm to enable the Reactor Coolant System to reach conditions at which the Decay Heat Removal System may be operated.

Two motor-driven emergency feedwater pumps are installed in each unit in addition to the steam-driven emergency feedwater pump. The motor-driven pumps are powered from diverse emergency power supplies.

All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Normally, decay heat is removed by steam relief through the turbine bypass system to the condenser. Condenser cooling water flow is provided by a siphon effect from Lake Keowee through the condenser for final heat rejection to the Keowee Hydro Plant tailrace. Decay heat can also be removed from the steam generators by steam relief through the main steam relief valves.

The minimum amount of water in the upper surge tank, condensate storage tank and hotwell is the amount needed for 11 hours of operation per unit. This is based on the conservative estimate of normal makeup being 0.5% of throttle flow. Throttle flow at full load, 11,200,000 lbs/hr., was used to calculate the operation time. For decay heat removal the operation time with the volume of water specified would be considerably increased due to the reduced throttle flow.

The total relief capacity of the 16 steam system safety valves is 13,105,000 lbs/hr.

<u>REFERENCE</u>

FSAR, Section 10

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and two channels each of the following are operable: four reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels, four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and high reactor building pressure instrument channels. The engineered safety features actuation system must have two analog channels functioning correctly prior to a startup. Additional operability requirements are provided by Technical Specifications 3.1.12 and 3.4 for equipment which are not part of the RPS or ESFAS.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column B (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE-279 as described in FSAR Section 7.

There are four reactor protective channels. A fifth channel that is isolated from the reactor protective system is provided as a part of the reactor control system. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other channels is one out of two.

The four reactor protective channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protective system bypass switch key permitted in the control room. That key will be under the administrative control of the Shift Supervisor. Spare keys will be maintained in a locked storage accessible only to the station Manager.

Each reactor protective channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. There are four shutdown bypass keys in the control room under the administrative control of the Shift Supervisor. The use of a key operated shutdown bypass switch for on-line testing or maintenance during reactor power operation has no significance when used in conjunction with a key operated channel bypass switch since the channel trip relay is locked in the untripped state. The use of a key operated shutdown bypass switch alone during power operation will cause the channel to trip. When the shutdown bypass switch is operated for on-line testing or maintenance during reactor power operation, reactor power and RCS pressure limits as specified in Table 2.3-1A, B, or C are not applicable.

The source range and intermediate range nuclear instrumentation overlap by one decade of neutron flux. This decade overlap will be achieved at 10^{-10} amps on the intermediate range instrument.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 600 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state on-line repairs to the failed device, when practical, will be made, and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

Containment isolation valves on non-essential systems are isolated by diverse signals from high containment pressure and low reactor coolant system pressure devices. The systems considered to be non-essential include:

1. Letdown line
2. RC Pump seal return line
3. Quench Tank sample line
4. Quench Tank gaseous vent
5. Reactor Building purge lines
6. Reactor Building sump drain line
7. Reactor Building atmosphere sample line
8. Pressurizer sample line
9. OTSG sample line
10. OTSG drain line

Containment isolation valves on essential systems are isolated by high containment pressure only. The systems considered to be essential include:

1. Component cooling to RC pumps
2. Low pressure service water cooling to RC pump motor

REFERENCE

FSAR, Section 7.1

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS

	Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree Of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
1.	Nuclear Instrumentation Intermediate Range Channels	1	0	Bring to hot shutdown within 12 hours (b)
2.	Nuclear Instrumentation Source Range Channels	1	0	Bring to hot shutdown within 12 hours (b)(c)
3.	RPS Manual Pushbutton	1	0	Bring to hot shutdown within 12 hours
4.	RPS Power Range Instrument Channels	3(a)	1(a)	Bring to hot shutdown within 12 hours
5.	RPS Reactor Coolant Temperature Instrument Channels	2(d)	1	Bring to hot shutdown within 12 hours
6.	RPS Pressure-Temperature Instruments Channels	2(d)	1	Bring to hot shutdown within 12 hours
7.	RPS Flux Imbalance Flow Instrument Channels	2	1	Bring to hot shutdown within 12 hours
8.	RPS Reactor Coolant Pressure			
	a. High Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours
	b. Low Reactor Coolant Pressure Channels	2	1	Bring to hot shutdown within 12 hours
9.	RPS Power-Number of Pumps Instrument Channels	2	1	Bring to hot shutdown within 12 hours
10.	RPS High Reactor Building Pressure Channels	2	1	Bring to hot shutdown within 12 hours

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
11. ESF High Pressure Injection System & Reactor Building Isolation (Non-essential Systems)			
a. Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
b. Reactor Building 4 PSIG Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
12. ESF Low Pressure Injection System			
a. Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
b. Reactor Building 4 PSIG Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
13. ESF Reactor Building Isolation (Essential Systems) & Reactor Building Cooling System			
a. Reactor Building 4 PSIG Instrument Channel	2	1	Bring to hot shutdown within 12 hours (e)
b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
14. ESF Reactor Building Spray System			
a. Reactor Building High Pressure Instrument Channel	2	1	Bring to hot shutdown within 12 hours (e)

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (Cont'd)

	Functional Unit	(A) Minimum Operable Analog Channels	(B) Minimum Degree Of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
	b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
15.	Turbine Stop Valves Closure	2	1	Bring to hot shutdown within 12 hours (f)

(a) For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

(d) Single loop operation at power (after testing and approval by the NRC/DOL) is not permitted unless the operating channels are the two receiving Reactor Coolant Temperature from operating loop.

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in the cold shutdown condition within 24 hours.

(f) One operable channel with zero minimum degree of redundancy is allowed for 24 hours before going to the hot shutdown condition.

3.5.3 Engineered Safety Features Protective System Actuation Setpoints

Applicability

This specification applies to the engineered safety features protective system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safety features protective system in the event of a breach of RCS integrity.

Specification

The engineered safety features protective actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building Pressure	Reactor Building Spray	≦30 psig
	High-Pressure Injection Reactor Building Isolation (Non-essential Systems)	≦4 psig
	Low-Pressure Injection	≦4 psig
	Start Reactor Building Cooling & Reactor Building Isolation (Essential Systems)	≦4 psig
	Penetration Room Ventilation	≦4 psig
Lower Reactor Coolant System Pressure	High Pressure Injection(1) & Reactor Building Isolation (Non-essential Systems)	≧1500 psig
	Low Pressure Injection(2)	≧500 psig

(1) May be bypassed below 1750 psig and is automatically reinstated aboved 1750 psig.

(2) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover the entire spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1500 psig low reactor coolant pressure setpoint for high pressure injection initiation and 500 psig for low pressure injection is to

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days.

Bases

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	MO	RF	
13. High Reactor Building Pressure	DA	MO	RF	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
15. High Pressure Injection				
Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
41. Engineered Safeguards Channel 1 HP Injection & Reactor Building Isolation Manual Trip	NA	RF	NA	Includes Reactor Building isolation of non-essential systems only
42. Engineered Safeguards Channel 2 HP Injection & Reactor Building Isolation Manual Trip	NA	RF	NA	Includes Reactor Building isolation of non-essential systems only
43. Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	RF	NA	
44. Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	RF	NA	
45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	RF	NA	Includes Reactor Building isolation of essential systems only
46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	RF	NA	Includes Reactor Building isolation of essential systems only
47. Engineered Safeguards Channel 7 Spray Manual Trip	NA	RF	NA	
48. Engineered Safeguards Channel 8 Spray Manual Trip	NA	RF	NA	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
49. Emergency Feedwater Flow Indicators	MO	NA	RF	
50. PORV and Safety Valve Position Indicators	MO	NA	RF	

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly

QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage

Table 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rod Movement (1)	Movement of Each Rod	Monthly
2. Pressurizer Safety Valves	Setpoint	**Each Refueling** (4)
3. Main Steam Safety Valves	Setpoint	Each Refueling (4)
4. Refueling System Interlocks	Functional	Prior to Refueling
5. Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6. Reactor Coolant System Leakage (2)	Evaluate	Daily
7. Condenser Cooling Water System Gravity Flow Test	Functional	Each Refueling
8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9. Spent Fuel Cooling System	Functional	Prior to Refueling
10. High Pressure and Low Pressure Injection System (3)	Vent Pump Casings	Monthly and Prior to Testing
11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	Each Refueling
12. RCS Subcooling Monitor	Functional	Each Refueling

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above $200^{\circ}F$ and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

(4) Number of safety valves to be tested each refueling shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

4.9 EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine-driven and motor-driven emergency feedwater pumps and associated valves.

Objective

To verify that the emergency feedwater pumps and associated valves are operable.

Specification

4.9.1 Pump Test

Monthly, the turbine-driven and motor-driven feedwater pumps shall be operated on recirculation to the upper surge tank for a minimum of one hour.

4.9.2 Valve Test

Quarterly, automatic valves in the emergency feedwater flow path will be determined to be operable in accordance with the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

4.9.3 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly.

Bases

The monthly testing frequency is sufficient to verify that the emergency feedwater pumps are operable. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pumps. The parameters which are observed are detailed in the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

REFERENCES

(1) FSAR, Section 10.2.2

(2) FSAR, Section 14.1.2.8.3

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, REVIEW, AND AUDIT

6.1.1 Organization

6.1.1.1 The station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.1.2 In all matters pertaining to actual operation and maintenance and to these Technical Specifications, the station Manager shall report to and be directly responsible to the Vice President, Steam Production, through the Manager, Nuclear Production. The organization is shown in Figure 6.1-2.

6.1.1.3 The station organization for Operations, Technical Services and Maintenance shall be functionally as shown in Figure 6.1-1. Minimum operating shift requirements are specified in Table 6.1-1.

6.1.1.4 Incorporated in the staff of the station shall be personnel meeting the minimum requirements encompassing the training and experience described in Section 4 of ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel" except for the Site Health Physicist.

The Site Health Physicist shall have a bachelor's degree in a science or engineering subject or the equivalent in experience, including some formal training in radiation protection, and shall have at least five years of professional experience in applied radiation protection of which three years shall be in applied radiation protection work in one of Duke Power Company's nuclear stations.

A qualified individual who does not meet the above requirements, but who has demonstrated the required radiation protection management capabilities and professional experience in applied radiation protection work at one of Duke Power Company's multi-unit nuclear stations, may be appointed to the position of Site Health Physicist by the station Manager, based on the recommendations of the System Health Physicist and as approved by the Manager, Nuclear Production.

6.1.1.5 Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

6.1.1.6 A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.

6.1.1.7 The two functions of the Shift Technical Advisor, namely accident assessment and operating experience assessment, are fulfilled in the following manner:

a. An experienced SRO, who has been instructed in additional academic subjects, will be assigned on-shift to provide the accident assessment capability.

b. Several engineers, familiar with plant operations and representing diverse technical backgrounds will be assigned to provide the operating experience assessment.

6.1.2 Technical Review and Control

6.1.2.1 Activities

a. Procedures required by Technical Specification 6.4 and other procedures which affect station nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared by a qualified individual/organization. Each such procedure, or procedure change, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or procedure change, but who may be from the same organization as the individual/group which prepared the procedure, or procedure change. Such procedures and procedure changes may be approved for temporary use by two members of the station staff, at least one of whom holds a Senior Reactor Operator's License on the unit(s) affected. Procedures and procedure changes shall be approved prior to use or within seven days of receiving temporary approval for use by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the station Manager.

b. Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the station Manager.

c. Proposed modifications to station nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to station nuclear safety-related structures, systems and components shall be approved prior to implementation by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent, or the Maintenance Superintendent, as previously designated by the station Manager.

d. Individuals responsible for reviews performed in accordance with 6.1.2.1.a, 6.1.2.1.b, and 6.1.2.1.c shall be members of the station supervisory staff, previously designated by the station Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated station review personnel.

e. Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the station Manager.

TABLE 6.1-1
MINIMUM OPERATING SHIFT REQUIREMENTS
(With Fuel in the Three Reactor Vessels)

	One Unit Operating*	Two Units Operating*	All Units Operating*	All Units Shutdown
Shift Supervisor (SRO)	1	1	1	1
Additional SRO	1	2**	2	1
Shift Technical Advisor (SRO)	1	1	1	None
Reactor Operator	4	4	5	3
Nuclear Equipment Operator	2	5	4	3

* Above cold shutdown

** Only one SRO required if both units are operated from one Control Room.

ADDITIONAL REQUIREMENTS

1. One licensed operator per unit shall be in the Control Room at all times when there is fuel in the reactor vessel.

2. Two licensed operators shall be in the Control Room during startup and scheduled shutdown of a reactor.

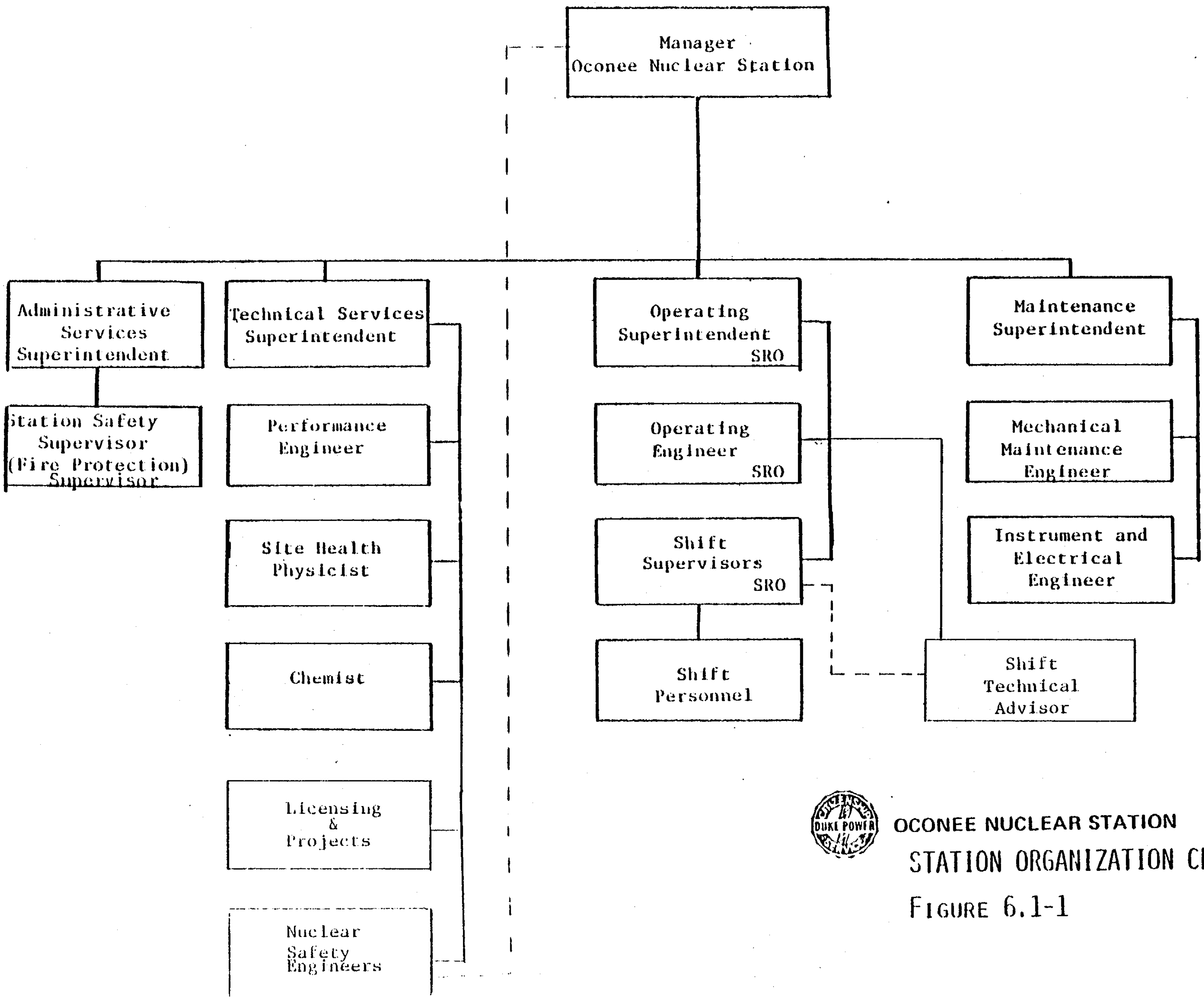
3. At least one licensed operator shall be in the reactor building when fuel handling operations in the reactor building are in progress.

4. An operator holding a Senior Reactor Operator license and assigned no other operational duties shall be in direct charge of refueling operations.

5. At least one person per shift shall have sufficient training to perform routine health physics requirements.

6. If the computer for a reactor is inoperable for more than eight hours, an operator, in addition to those required above, shall supplement the shift crew.

7. A fire brigade of 5 members shall be maintained on site at all times. This excludes 3 members of the minimum operating shift requirements that are required to be present in the control rooms.



OCONEE NUCLEAR STATION

STATION ORGANIZATION CHART

FIGURE 6.1-1

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

b. Thirty-Day Written Reports

The types of events listed below shall be the subject of written reports to the Director, Office of Inspection and Enforcement, Region II, within 30 days of discovery of the event. (Copy to the Director, Office of Management Information and Program Control)

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or shutdown required by a limiting condition for operation.

(3) Observed inadequacies in the implementation of administrative or procedural controls during operation of a unit which could cause reduction of degree of redundancy provided in the Reactor Protective System or Engineered Safety Feature Systems.

6.6.2.2 Environmental Monitoring

a. If individual milk samples show I-131 concentrations of 10 picocuries per liter or greater, a plan shall be submitted within one week advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

b. If milk samples collected over a calendar quarter show average concentrations of 4.8 picocuries per liter or greater, a plan shall be submitted within 30 days advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

c. If, during any annual report period, a measured level of radioactivity in any environmental medium other than those associated with gaseous radioiodine releases or liquid effluent releases exceeds ten times the control station value, a written notification will be submitted within one week advising the NRC of this condition. This notification should include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

d. If, during any annual report period, a measured level of radioactivity in any environmental medium associated with liquid effluent releases exceeds 50 times the control station value for sampling points at or upstream of location 000.7 or ten times the control station value for sampling points downstream of location 000.7, a written notification will be submitted within one week advising the NRC of this condition. This notification should include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous results.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-55

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DUKE POWER COMPANY

DOCKETS NOS. 50-269, 50-270 AND 50-287

I. INTRODUCTION

This Safety Evaluation deals with three separate license amendment requests. Duke Power Company (DPC or the licensee) in each request proposed changes to the Technical Specifications (TSs) appended to Facility Operating License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station (ONS) Units Nos. 1, 2 and 3. The request dated October 2, 1980, also proposed three additional license conditions.

a. By letter dated May 21, 1979, the licensee requested revisions to the high pressure reactor trip setpoint and the pressurizer power operated relief valve (PORV) setpoint.

b. By letter dated October 2, 1980, as supplemented October 30, 1980, the licensee requested the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements into the ONS license and TSs.

c. By letter dated October 20, 1980, the licensee requested the inclusion of an additional section of Regulatory Guide 1.16 concerning reporting requirements into the TSs.

II. BACKGROUND INFORMATION

The following discussion concerns the October 2, 1980 request only. The backgrounds of the May 21, 1979 and October 20, 1980 requests are provided in conjunction with the Evaluation sections. By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to the licensee dated April 7, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TSs to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

III. EVALUATION

a. High Pressure Reactor Trip and PORV Setpoints

In response to IE Bulletin 79-05B, the licensee submitted by letter dated May 21, 1979, a proposed amendment to the TSs lowering the setpoint of the high pressure reactor trips from 2355 psig to 2300 psig. Concurrently, the licensee proposed raising the setpoint of the pressurizer electromatic relief valves, also known as the power operated relief valves (PORV), from 2255 psig to 2450 psig. The licensee physically changed the setpoints prior to the TS change request, as part of his response to IE Bulletin 79-05B. The lowered high pressure trip setpoint was within the envelope of the existing TS, thus the licensee did not need this change to be issued to lower the setpoint. The PORV setpoint is not subject to TS limitations; however, the PORV setpoint is identified in the basis of TS Section 2.2 - Safety Limits-Reactor Coolant System Pressure.

In the past, during turbine trip and loss of feedwater transients, the PORV was lifted and the reactor would not trip unless the pressure setpoint was exceeded. With the new setpoints, these transients do not result in lifting of this valve. Therefore, this valve does not open as frequently and the likelihood of the valve failing to close, i.e., producing a small break loss of coolant event, is reduced. However, the likelihood of a reactor trip is increased. We do not consider this increase in frequency of a reactor trip to be significant over the lifetime of the plant. The PORV function is to control an operational transient and not to protect the reactor coolant system pressure boundry. The safety valves provide this function. The new setpoints would not reduce the margin of safety or increase the probability or consequences of accidents. We find the proposed changes to the TSs acceptable.

b. TMI-2 Lessons Learned Category "A" Requirements Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has the requisite emergency power supplies. The licensee has proposed adequate TSs which provide for a 31-day channel check and an annual channel calibration and actions in the event of component inoperability. We have reviewed these proposed TSs and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components and are thus acceptable.

Direct Indication of Valve Position

The licensee has provided a direct indication of PORV and safety valve position in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a 31-day channel check and an annual channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of low reactor coolant level and inadequate core cooling. These instruments, sub-cooling meters, receive and process data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated April 7, 1980. The licensee submitted TSs with a 31-day channel check for the input components of these instruments and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude the TSs are acceptable as they meet our July 2, 1980 model TS criteria.

Diverse Containment Isolation

The licensee has modified the containment isolation system so that diverse parameters will be sensed to ensure automatic isolation of non-essential systems under postulated accident conditions. These parameters are reactor building pressure and reactor coolant pressure. We have reviewed this system in our Lessons Learned Category "A" Safety Evaluation dated April 7, 1980. The modification is such that it does not result in the automatic loss of containment isolation after the containment isolation signal is reset. Reopening of containment isolation would require deliberate operator action. The TSs submitted by the licensee list each affected containment isolation valve and provide for the appropriate surveillance and actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

Auto Initiation of Auxiliary Feedwater Systems

The licensee has provided for the automatic initiation of auxiliary (emergency) feedwater flow on loss of normal feedwater flow. The auto-initiation signals used by the licensee are loss of main feedwater header discharge pressure and main feedwater pump control oil pressure. We have previously reviewed the design and installation of this system as part of our Lessons Learned Category "A" program. The circuits are designed to be testable and the design retains the capability of manual actuation from the control room even in the event of failure of the auto-initiating circuitry. The TSs submitted by the licensee list the appropriate components, describe the tests and provide for proper test frequency. The TSs contain appropriate actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

Auxiliary (Emergency) Feedwater Flow Indication

The licensee has installed auxiliary (emergency) feedwater flow indication that meets our testability and vital power requirements. We reviewed this system in our Safety Evaluation dated April 7, 1980. The licensee has proposed a TS with 31-day channel check and annual channel calibration requirements. We find this TS acceptable as it meets the criteria of our July 2, 1980 model TS criteria.

Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift to perform the function of accident assessment. The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. The operating experience review function will be performed by a group of on-site staff engineers who will keep the STA informed of the results of the operational analysis function; this is in agreement with our April 7, 1980 Safety Evaluation. Based on our review, we find the licensee's submittal to satisfy our requirements and is acceptable.

Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from reactor containment. By letter dated October 2, 1980, the licensee agreed to adopt such a license condition; accordingly, we have included this condition in the license.

Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. By letter dated October 2, 1980, the licensee agreed to adopt such a license condition; accordingly, we have included this condition in the license.

Backup Method for Determining Subcooling Margin

Our letter of July 2, 1980, indicated that the license should be amended by adding a license condition related to the determination of subcooling margin; this is a precursor to warn of inadequate core cooling in the event of an accident. Such a condition would require the training of personnel and the generation of procedures to accurately monitor the reactor coolant system subcooling margin. By letter dated October 2, 1980, the licensee agreed to adopt such a license condition; accordingly, we have included this condition in the license.

c. Reporting Requirement for Generic Issues

By letter dated May 29, 1980, from NRC to the licensee, we requested that Item (9) of Section C.2.a of Regulatory Guide 1.16, Revision 4, be added to Section 6.6.2.1. of the ONS Common TSs. As noted in the Regulatory Guide, this item is intended to provide for reporting of potentially generic safety problems. By letter dated October 20, 1980, the licensee submitted a request to add Item (9) to the TSs. As this addition fulfills the regulatory position of Regulatory Guide 1.16, we find the change acceptable.

IV. ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

V. CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 28, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION 7590-01

DOCKETS NOS. 50-269, 50-270 AND 50-287

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 92 , 92 , and 89 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the licenses and the Common Technical Specifications for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments: 1) revise the Technical Specifications regarding the high pressure trip setpoint and the pressurizer power operated relief valve setpoint; 2) add three license conditions and additional Technical Specifications which incorporate certain of the Three Mile Island Unit No. 2 Lessons Learned Category "A" requirements; and 3) revise the Technical Specifications to include an additional portion of Regulatory Guide 1.16 in the reporting requirements.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated May 21, 1979; October 2, 1980, as supplemented October 30, 1980; and October 20, 1980, (2) Amendments Nos. 92 , 92, and 89 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oconee County Library, 201 South Spring Street, Walhalla, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 28th day of January 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing