DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Attachment 1

Proposed Technical Specification Revisions

Remove Pages	Insert Pages
2.3-7	2.3-7
3.3-5	3.3-5
3.3-6	3.3-6
-----	3.3-7 (New Page)

8704130102 870406
PDR ADOCK 05000269
P PDR

TABLE 2.3-1

Reactor Protective System Trip Setting Limits

	RPS Trip		RPS Trip Setpoint	Shutdown Bypass
1.	Nuclear Overpower		105.5% Rated Power (4)(5)	5.0% Rated Power (1)
2.	Flux/Flow/Imbalance		1.07	Bypassed
3.	Pump Monitors	a.	> 0% Rated Power loss of two pumps in one reactor coolant loop	Bypassed
		b.	> 55% Rated Power loss of two pumps	
		c.	> 0% Rated Power loss of one or two pumps during two pump operation	
4.	High Reactor Coolant System Pressure		2300 psig	1720(2)
5.	Low Reactor Coolant System Pressure		1800 psig	Bypassed
6.	Variable Low Reactor coolant System Pressure		P (psig) = (11.14 T_{out} - 4706)(3)	Bypassed
7.	High Reactor Coolant Temperature		618°F	618°F
8.	High Reactor Building Pressure		4 psig	4 psig

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

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

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

(4) Until the 1A LPI cooler is cleaned, tested and evaluated for full power operation or until EOC 10, the Setpoint for Unit 1 is 91.5% rated power.

(5) Until the 2A LPI cooler is cleaned, tested and evaluated for full power operation or until midnight of April 22, 1987, the Unit 2 setpoint is 81.7% rated power.

3.3.7 Low Pressure Service Water (LPSW):

a. Prior to initiating maintenance on any component of the LPSW system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

(1) Two LPSW pumps for the shared Unit 1, 2 LPSW system and two LPSW pumps for the Unit 3 LPSW system shall be operable with valves LPSW-108, 2LPSW-108, and 3LPSW-108 locked open.

(2) Tests or maintenance shall be allowed on any component of the LPSW system provided the redundant train of the LPSW system is operable. If the LPSW system is not restored to meet the requirements of Specification 3.3.7.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition with 12 hours. If the requirements of Specification 3.3.7b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250° F within an additional 24 hours.

3.3.8 For Unit 1 until the 1A LPI cooler is cleaned, tested and evaluated for full power operation or until EOC 10:

a. The maximum allowable Power level shall be 91.5% rated power.

b. In addition to the requirements of Specification 3.3.2, the remaining non-ES LPI pump, capable of taking suction from the reactor building emergency sump and discharging into the RCS, shall be operable.

(1) The remaining non-ES LPI pump may be inoperable for a period of 24 hours. If the non-ES LPI pump is not restored to operable status within 24 hours, the reactor shall be placed in a hot shutdown condition within an additional 12 hours. If the requirements of 3.3.8(b) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250 F within an additional 24 hours.

3.3.9 For Unit 2 until the 2A LPI cooler is cleaned, tested and evaluated for full power operation or until midnight of April 22, 1987:

a. If lake water temperature is equal to or less than 55°F, the maximum allowable power level shall be 81.7% rated power.

b. In addition to the requirement of Specification 3.3.2, the remaining non-ES LPI Pump, capable of taking suction from the reactor building emergency sump and discharging into the RCS, shall be operable.

(1) The remaining non-ES LPI pump may be inoperable for a period of 24 hours. If the non-ES LPI pump is not restored to operable status within 24 hours, the reactor shall be placed in a hot shutdown condition within an additional 12 hours. If the requirements of 3.3.9(c) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within additional 24 hours.

Bases

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required. In the event that the need for emergency core cooling should occur, functioning of one high pressure injection pump, one low pressure injection pump, and both core flood tanks will protect the core, and in the event of a main coolant loop severence, limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad.(1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling.(2) The analysis of these breaks indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

Three hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70°F without any control rods in the core. The minimum value specified in the tanks is 1835* ppm boron.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the Reactor Building design pressure will not be exceeded with one spray and two coolers operable. (4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable for seven days or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Unit 1, 2, and 3. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

Prior to initiating maintenance on any of the components, the redundant component(s) shall be tested to assure operability. Operability shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. The 24 hour period prior to removal is adequate to permit efficient scheduling of manpower and equipment testing while ensuring that the testing is performed directly prior to removal. The basis of acceptability is the low likelihood of failure within a clearly defined 48 hours following redundant component testing.

REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.3.3.2

(4) FSAR, Section 15.14.5

* 2010 ppm boron for Unit 3, Cycle 10 only.

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Attachment 2

Justification For Emergency Situation

Justification For Emergency Situation

On February 15-18, 1986 the Unit 1 LPI coolers were tested and the test data was analyzed to determine performance capability of the LPI coolers to transfer heat. The results indicated that the 1A LPI cooler had a performance capability of 53% of design UA, and that the 1B LPI cooler had a performance capability of 107% of design UA (where UA is the overall heat transfer coefficient times the total heat transfer area). The 1A LPI cooler was cleaned prior to returning the unit to service. Unit 1 was in a refueling outage when the LPI coolers were tested and cleaned. In addition, Duke determined that given the measured performance capability of the 1A LPI cooler (53% of design UA), that the 1A LPI train was still operable (i.e., capable of performing its safety function). On August 7, 1986, the 1A LPI cooler was re-tested and the performance capability of the cooler was 74% of design UA.

In September of 1986, the Unit 1 Reactor Building Cooling Unit (RBCU) coolers were tested and the analysis of the test data indicated a performance capability of the coolers as follows:

RBCU Coolers	% of Design UA (Normal Conditions)
1A	85
1B	84
1C	95

At this time, it should be noted that the test data on the RBCU coolers was obtained with the RBCU operating in the normal mode. (Chapter 6 of the ONS FSAR provides information on the operation of the RBCU System.) Under emergency conditions, Duke expected the capabilities of the RBCU coolers to improve due to improved heat transfer with a "wet" surface and the self-cleaning effect which would occur under the elevated temperature and saturated air conditions. However to be conservative, Duke assumed that the performance capability under emergency conditions would be the same as that determined under normal conditions and pursued independent analysis of the test data for emergency conditions through an outside consultant. The information on the performance capability of the Unit 1 RBCU coolers did not impact the original determination of operability of the 1A LPI cooler.

Based on the information known at this time, Duke concluded that, although the performance of the 5 heat exchangers (2LPI coolers and 3 RBCU coolers) were degraded, these systems (LPI System and RBCU System) were still capable of performing their intended safety function.

The Unit 2 RBCU coolers and the Unit 2 LPI coolers were tested in August of 1986. The results of the analysis of the test data indicated the following performance capability of the 5 Heat Exchangers:

RBCU Coolers	% of Design UA (Normal Conditions)
2A	111
2B	153
2C	118

LPI Coolers	% of Design UA
2A	57
2B	100

To further substantiate the test results on Unit 2, a visual inspection of the 2B RBCU cooler and 2A LPI cooler was performed. The results of the inspection indicated the 2B RBCU cooler was clean on both the air and water side, and that the 2A LPI cooler was fouled. Based on the information known at that time, Duke again concluded that, although there was a noted degradation of performance capability of the 5 Heat Exchangers, the LPI system and the RBCU system were still operable.

In parallel to the testing and analysis of the test data, a consultant was obtained to develop a methodology/program to analyze the test data from the RBCU coolers in order to determine the performance capability of these coolers during emergency conditions. The results of the consultant's efforts were provided to Duke March 30, 1987. The test data from the RBCU cooler testing was analyzed by the consultant's methodology/program and the results obtained were used to assess the ability of the LPI system and the RBCU system to perform their intended safety function. See Attachment 4 for the resultant heat removal capability of the RBCU coolers under accident conditions. The evaluation of the impact of the newly determined performance capability of the RBCU coolers concluded that the components were operable and capable of performing their accident mitigation functions as long as the maximum reactor power was 91.5% of rated power for Unit 1 and 66.1% rated power for Unit 2. Subsequent evaluation of Unit 2 indicated that a maximum power level of 81.7% of rated power could be supported, provided lake water temperature was equal to or less than 55° F.

To support the operation of Units 1 and 2 at these power levels, a proposed technical specification to define the new maximum power level is submitted by a Duke letter dated April 6, 1987. With Units 1 and 2 operating at or less than the power levels, the LPI and RBCU systems are capable of performing their intended safety functions of mitigating the consequences of a loss-of-coolant accident.

Additionally, it should be noted that the testing of coolers to determine heat transfer capability can only be performed at certain times and require the temporary installation of precision instrumentation. The testing technique and methodology to analyze the test data has evolved over the last two years. Current industry testing techniques and regulatory required surveillance tests did not indicate heat exchanger performance degradation of a magnitude sufficient to limit power. Only the specialized testing techniques and methodology being developed by Duke are indicating performance degradation of this magnitude. One fundamental requirement in the performance of these tests is the need to have a heat load on the coolers. For the RBCU coolers, this can only be done when the reactor is at a hot shutdown condition or greater and after temporary installation of additional instrumentation. For the LPI coolers the only time test data can be obtained to determine heat transfer capability is when the RCS pressure is approximately between 125 psig and 300 psig.

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Attachment 3

No Significant Hazards Consideration

No Significant Hazards Consideration Evaluation

Duke has determined that the proposed amendment request poses no significant hazards as defined by NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The proposed amendment establishes interim maximum allowable power levels and Reactor Protective System (RPS) high flux trip setpoints for Units 1 and 2. In addition, the proposed amendment will specify Limited Condition of Operation (LCO) for the Non-Engineered Safeguard (ES) Low Pressure Injection (LPI) Pump, (1C, 2C LPI pumps). For Unit 1, the maximum allowable power level and the RPS High Flux trip setpoint is 91.5% rated power. For Unit 2, the maximum allowable power level and the RPS High Flux trip setpoint is 81.7% rated power if the lake water temperature is equal to or less than 55°F. The LCO requires that 1C and 2C LPI pumps be operable and defines what actions to take when the pump is inoperable.

Recent testing of the two LPI coolers and the three Reactor Building Cooling Unit (RBCU) coolers have indicated that the performance of these coolers have degraded from their original design capacities. An evaluation of this situation concluded that these coolers are operable and capable of performing their accident mitigating functions at the respective reduced power levels for Units 1 and 2. A discussion of the methodology, assumptions and analysis techniques utilized in the evaluation is provided in Attachment 4. The criteria for operability for the LPI and RBCU coolers are as follows:

(1) Unit Cooldown: Both LPI coolers can cool the unit from 250° F to 140° F in 14 hours.

(2) LOCA: The combined performance of the LPI and the RBCU coolers can remove decay heat 30 minutes after a LOCA, assuming a single failure in each system (i.e., worst LPI cooler and two worst RBCUs available for decay heat removal).

(3) Equipment Qualification (EQ): The Reactor Building cooling capacity must be sufficient to prevent post-LOCA conditions from exceeding the EQ envelope, again assuming a worst case single failure in the LPI and RBCU systems.

The RPS high flux trip setpoint is provided to prevent damage to the fuel cladding from the reactivity excursions too rapid to be defected by pressure and temperature measurements. By reducing the high flux trip setpoint for units 1 and 2 additional operating margin is provided to assure that no reactivity excursions will result in fuel cladding damage.

To assure that at least two LPI pumps are available, if required, the 1C and 2C LPI pumps are required to be operable. For certain worst case single failures, two LPI pumps are required to provide sufficient LPI flow through one LPI cooler. By requiring all three LPI pumps to be operable, there will always be at least two LPI pumps available to be aligned to LPI cooler for the worst case single failure. The operator will need to take manual action to align two LPI pumps through the operable cooler. To accomplish this, at the most two valves in the crossover line will need to be opened. This action can be taken from the control room and does not have to occur before thirty minutes into the event. No additional action will be required by the operator beyond what the operator currently would perform given a LOCA.

The following provides a discussion of how the proposed amendment satisfies each of the three standards provided by 10 CFR 50.92(c).

First Standard

(Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to the proposed changes. Operation of the units at a reduced power level and the reduction of the RPS high flux trip setpoint does not increase the probability or consequences of any FSAR accident that has been evaluated. The requirement for the operability of the 1C and 2C LPI pump does not increase the probability or consequences of any previously evaluated accidents. The plant was licensed at full power operations and all accident analysis that are impacted by power are analyzed at full power operation. Accordingly, operations at a reduced power level will still be bounded by the previous analysis of the accident. The fouling of the RBCU and LPI coolers only impact the ability of the coolers to transfer heat released during LOCA. Attachment 4 provides the evaluation of the degraded performance capability of these coolers to transfer heat during a LOCA. The evaluation concluded that the coolers are operable and capable of performing their accident mitigating functions at the reduced power levels for Units 1 and 2. Accordingly, the proposed amendment does not significantly increase the probability or consequences of a previously evaluated accident.

Second Standard

(Amendment would not) create the possibility of a new or different kind of accident from any accident previously evaluated.

The reduction in power level and the resetting of the RPS High Flux setpoint for Units 1 and 2 will not create the possibility of a new or different kind of accident. The lower power level assures that the LPI coolers and RBCU coolers are capable of mitigating the consequences of an accident. A discussion how this was verified, given the degraded performance of these coolers, is provided by Attachment 4. Operation at a reduced power does not, in itself create a new or different kind of accident. Further the time period in which each unit will be operating at its reduced power level is relatively short. Unit 1 will be operating at its reduced power level for approximately six months, Unit 2 less than three weeks.

Requiring that all three LPI pumps will not create any new or different kind of accidents. 1C and 2C LPI pumps are routinely verified operable per Oconee Nuclear Station Inservice Testing Program.

Third Standard

(Amendment would not) involve a significant reduction in a margin of safety.

The reduction of the RPS High Flux trip setpoint provides an additional margin of safety beyond what is currently present with the setpoint at 105.5% rate power. This trip setpoint prevents fuel clad damage that could possible occur due to a rapid reactivity excursion. A reactivity excursion to cause fuel clad damage at the reduced power levels will need to be significantly greater than what would be required at operation at or near 100% rated power. Operating at a reduced power level assures that the LPI and RBCU coolers will, for all situations, be able to perform their intended safety function (see Attachment 4 for the evaluation of these coolers to perform their safety function). The requirement for the operability of the 1C and 2C LPI pumps is consistent with the conclusions of the Safety Evaluation (Attachment 4). As such, there is not significant reduction in the margin of safety.

The above evaluation shows that the three standards of 10 CFR 50.92(c) are satisfied. Accordingly, Duke submits that the proposed changes do not represent any significant hazard.

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Attachment 4

Technical Justification

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

ATTACHMENT 4

Technical Justification

Basis of Maximum Allowable Power
Level with Degraded Low Pressure
Injection Coolers and Reactor Building Coolers

I. BACKGROUND

Performance testing of the low pressure injection (LPI) coolers and the Reactor Building (RB) coolers have identified that 5 of the 6 LPI coolers (2 per unit) and all 9 RB coolers (3 per unit) were performing at less than full design heat transfer capacity. This degradation is a result of the accumulation of normal service fouling since the equipment was placed in service in 1973-74. Following confirmation of the validity of the cooler test data, the available heat transfer capability was analyzed with respect to the design basis capability in order to determine if the equipment was able to perform the required heat transfer functions. The results of these analyses determined that with the worst case single failure assumed, operation at full power could not be supported. A methodology was developed to determine what power level could be justified based on the existing cooler performance. This methodology utilizes increased low pressure service water (LPSW) flow on the shell side of the LPI coolers, increased LPI flow on the tube side of the LPI coolers, and lower LPSW temperature to the LPI coolers. Station operating procedures have been revised to incorporate the necessary guidance for achieving the higher flowrates in the event that a LOCA occurs along with the limiting single failure.

II. COOLER PERFORMANCE ACCEPTANCE CRITERIA

In order to evaluate the acceptability of degraded LPI cooler and RB cooler performance, the Oconee FSAR and other technical documentation were reviewed. Three heat transfer functions were identified that had the potential for exceeding the existing heat transfer capability.

Plant Cooldown

This criterion, as stated in FSAR Section 9.3.3.2, requires the capability to cool down a unit from 250 F to 140 F in 14 hours. The intent is to ensure that when the LPI coolers are aligned at 250 F, the refueling shutdown condition (as defined by Technical Specification 1.2.6) can be expeditiously attained. This cooldown rate must be attained using both LPI cooler trains.

Loss of Coolant Accident

Following a LOCA the LPI coolers function to remove energy from the sump inventory in the recirculation mode. The cooled water is then returned to the reactor vessel where it absorbs decay heat and stored structural energy before returning to the Reactor Building through the break. The RB coolers are actuated on high Reactor Building pressure and function by recirculating the steam/air mixture and removing sensible heat and the latent heat due to condensation.

The LPI coolers were originally sized to remove decay heat at 30 minutes following a LOCA. Statements to that effect do not exist in the current FSAR. In the original FSAR Section 6.1.3.2, it was stated that "One low pressure injection pump and cooler combination is capable of removing the heat generated after loss-of-coolant

accident." In the preceeding sentence it was stated that "... the decay heat being generated in the core at 30 minutes after the accident is approximately 1.8% of full power, or 160 x 10^6 Btu/hr." These statements were deleted in a recent annual FSAR update because it was recognized that the first statement was incorrect. An explanation of the error is as follows. The 160 x 10^6 Btu/hr value is equal to the LPI cooler design capacity at 250 F with two LPI pumps and twice the normal LPSW flow. In addition, the Reactor Building sump temperature at 30 minutes following a LOCA is 235 F, not 250 F. Therefore, even the increased flowrates to the LPI cooler would not remove decay heat at 30 minutes.

Although the original FSAR statements were incorrect, the intent was for the Reactor Building cooling systems to match decay heat at 30 minutes following a LOCA. Therefore, the acceptance criterion for the post-LOCA heat transfer performance of the LPI coolers and RB coolers is that the combined Reactor Building cooling capability matches decay heat at 30 minutes. The worst single failure results in the worst LPI cooler and the two worst RB coolers being available for Reactor Building cooling.

Equipment Qualification

The equipment qualification (EQ) criterion requires that the Reactor Building cooling capability be sufficient to prevent exceeding the EQ envelope. The EQ envelope is the Reactor Building vapor temperature assumed as the basis for equipment qualification. It is described by the following temperature vs. time points as plotted on a log scale time axis.

Time	Temperature
0 - 10 sec	125 F ramp to 286 F
10 sec - 10 min	286 F
10 min - 24 hr	286 F ramp to 125 F
24 hrs	125 F

The EQ envelope bounds the long-term containment temperature response that assumed one LPI cooler and 1 RB cooler in operation following the design basis LOCA. The heat transferred by that cooling combination as a function of time is known based on the calculated Reactor Building sump and vapor temperatures. The available heat transfer capability assuming the worst LPI cooler and the two worst RB coolers at the EQ envelope temperatures can be calculated. The existing heat transfer capability can then be compared to the EQ design basis heat transfer capability.

III. COOLER PERFORMANCE DATA

Cooler performance data originates from plant test data. In order to calculate cooler performance at conditions other than tested conditions, such as RB cooler performance in a post-LOCA environment and LPI cooler performance at higher flowrates or cooler LPSW temperatures, some engineering calculations were performed. These

calculations are described in Attachments A and B for the LPI coolers and the RB coolers respectively. This section tabulates cooler performance data used in the calculations of maximum allowable power level.

FSAR Figure 6.3-4 (Figure III-1) shows the LPI cooler performance assumed in the containment analysis. This figure is based on 3000 gpm LPSW flow, 3000 gpm LPI flow, and 75 F LPSW temperature. This figure will be referred to as 100% capacity, corresponding to 104 x 10^6 Btu/hr at 250 F.

FSAR Figure 6.2-3 (Figure III-2) shows the RB cooler performance assumed in the containment analysis. This figure is based on 1400 gpm LPSW flow and 75 F LPSW temperature. This figure will be referred to as 100% capacity, corresponding to 80 x 10^6 Btu/hr at 286 F.

FSAR Section 6.2.2 states that 100% design cooling capacity is 240 x 10^6 Btu/hr. This is simply the capacity of three RB coolers (i.e. 3 x 80 x 10^6 Btu/hr at 286 F) and is not the cooling capacity assumed in the containment analysis. As stated before, the cooling capacity in the containment analysis is that corresponding to one LPI cooler and one RB cooler. In reality, one LPI cooler and two RB coolers are available even when considering the worst single failure.

Table III-1
LPI Cooler Performance

Cooler	LPI Flow (gpm)	LPSW Flow (gpm)	LPSW Temp (F)	Capacity (%)
1A	3000	3000	75	79.5
	3000	5400	75	90.7
	5650	5400	75	106.7
	3000	3000	55	87.6
	3000	5400	55	100.0
	5650	5400	55	117.3
1B	3000	3000	75	108.6
	3000	5400	75	130.7
	5650	5400	75	171.1
	3000	3000	55	119.2
	3000	5400	55	143.2
	5650	5400	55	187.5
2A	3000	3000	75	62.6
	3000	5400	75	69.5
	5650	5400	75	78.3
	3000	3000	55	69.2
	3000	5400	55	76.8
	5650	5400	55	86.4

Cooler	LPI Flow (gpm)	LPSW Flow (gpm)	LPSW Temp (F)	Capacity (%)
2B	3000	3000	75	97.1
	3000	5400	75	115.3
	5650	5400	75	144.2
	3000	3000	55	107.6
	3000	5400	55	125.9
	5650	5400	55	157.6

Table III-2
RB Cooler Performance

Cooler	Capacity (%)
1A	46
1B	38
1C	47
2A	59
2B	88
2C	66

Some important observations regarding the data should be discussed. LPI cooler 1A exceeds 100% cooling capacity at 75 F LPSW temperature, provided that 2 LPI pumps (i.e. 5650 gpm) and 5400 gpm LPSW flow are available. LPI cooler 1A equals design capacity at 55 F LPSW temperature, provided that 5400 gpm LPSW flow is available. LPI cooler 1B exceeds design capacity. LPI cooler 2A does not reach design capacity at any conditions. LPI cooler 2B exceeds 100% cooling capacity at 75 F with 5400 gpm LPSW flow, and under normal flows at 55 F. The two worst Unit 1 RB coolers do not sum to 100% cooling capacity. The combined performance of the two worst Unit 2 RB coolers does exceed the design performance of the RB cooler.

On Unit 1, acceptable flow conditions for full power operation cannot be justified by reviewing this data, since the two worst Unit 1 RB coolers are not equivalent to the design capacity of one cooler. Similarly, full power operation of Unit 2 cannot be justified by reviewing this data since the 2A LPI cooler does not reach design cooling capacity under any conditions. The methodology of Section IV must be used to determine what power level can be justified by comparing to the acceptance criteria of Section II.

IV. MAXIMUM ALLOWABLE POWER LEVEL METHODOLOGY

Plant Cooldown

The plant cooldown analysis methodology determines if the two LPI coolers can meet the plant cooldown criterion of 250 F to 140 F in 14 hours. Based on 3.3 hours to go from T-ave of 579 F to 250 F at the maximum cooldown rate of 100 F/hr, the decay heat load is

initially that corresponding to 3.3 hours after trip. Since one reactor coolant pump is left in operation for thermal mixing and pressurizer spray capability, the heat load of one pump is included. The stored energy in the primary coolant and structures is also included. No credit is taken for steam generator heat transfer even though steaming will assist the cooldown until boiling capability is lost.

The methodology determines if additional LPSW flow in excess of the normal 3000 gpm per LPI cooler is necessary, and if the operating reactor coolant pump must be tripped in order to reduce the heat load.

Loss of Coolant Accident

The LOCA analysis methodology determines if the worst LPI cooler plus the two worst RB coolers can remove decay heat at 30 minutes. Decay heat at 30 minutes was calculated to be 151.4 x 10^6 Btu/hr, assuming operation at 102% power. This value is based on the 1979 ANS Standard 5.1 assuming three consecutive 440 effective full power day (EFPD) fuel cycles (i.e. 18-month fuel cycles) with 21 day refueling outages. The Reactor Building vapor and sump temperatures at 30 minutes are taken from Figures IV-2 and IV-3. Figures IV-1 to IV-3 are the design basis long-term containment analyses. Figures IV-1 and IV-2 are referred to in the FSAR (p. 15.14-11a) as Figures 15.14-25a and 15.14-25b. Due to an error in the issuance of the annual FSAR updates, these figures were not distributed. Figure IV-3 does not exist in the current FSAR. This situation will be corrected in the next update. Figure IV-2 is the basis for the EQ envelop. The "1 Cooler" curve on Figure IV-2 is bounded by the EQ envelope. The sump and vapor temperatures at 30 minutes were taken from the figures as 235 F.

The heat transfer capabilities of the LPI coolers and RB coolers at 235 F are as follows:

$$Q_{LPI} = \left(\frac{DC}{100}\right) \times \left(\frac{104 \times 10^6 \text{ Btu}}{\text{hr}}\right) \times \left(\frac{235 - T_{LPSW}}{250 - T_{LPSW}}\right) \times (0.96)$$

Where: DC = design capacity of worst cooler (%)

T_{LPSW} = LPSW temperature (F)

0.96 = 4% safety factor to account for additional fouling before next cooler cleaning

104×10^6 = LPI design capacity @ 250 F. Refer to Figure III-1.

235 = sump temperature

250 = LPI design temperature. Refer to Figure III-1.

$$Q_{RB} = \frac{(DC)}{(100)} \times (54.5 \times 10^6 \text{ Btu/hr})$$

Where: DC = summed capacity of two worst RB coolers less 4% on each cooler to account for additional fouling before next cooler cleaning.

54.5×10^6 = design capacity of one RB cooler at 235 F. Refer to Figure III-2.

The sum of the LPI cooler and RB cooler capacities are then compared to the 151.4×10^6 Btu/hr criterion. If the sum is greater, then 100% power operation is justified. If the sum is less, then a ratio of the available to required total cooling capacity is used to calculate the allowable power level. If the result is less than 100% power then the design capacity of the worst LPI cooler can be increased per the capacities in Table III-1 corresponding to other flow and temperature conditions, and a new power level calculated.

Equipment Qualification

The EQ analysis methodology determines if the worst LPI cooler operating at the temperatures in Figure IV-3 plus the two worst RB coolers operating at the EQ envelope temperatures can provide the same heat transfer as one LPI cooler plus one RB cooler operating at design capacities at the RB vapor and sump temperatures given by Figures IV-2 and IV-32. For various times out to 24 hours the calculation is performed. If the sum of the worst LPI cooler plus the two worst RB coolers exceeds the sum of the design capacities of one of each cooler type at all times, then 100% power operation is justified. If 100% power operation is not justified, then a table of ΔQ's is calculated as a function of time. For each time point at which a ΔQ exists (i.e. existing cooling capacity less than design), the decay heat (Q_{DH} is calculated. The allowable power level is then calculated by subtracting $\Delta Q/Q_{DH}$ (expressed as a percentage) from 100%. The smallest value (as a function of time) is the maximum allowable power level.

V. RESULTS

Unit 1

Plant Cooldown

100% power operation is justified with LPSW flow of 5400 gpm, to the 1B LPI cooler and provided the reactor coolant pump is tripped upon reaching 160 F.

LOCA

Maximum Power Level	LPSW Temp.	LPI Flow	Requirements LPSW Flow
Case 1: 98.7%	55 F	5650 gpm	5400 gpm
*Case 2: 91.5%	75 F	5650 gpm	5400 gpm
*Case 3: 87.9%	55 F	3000 gpm	5400 gpm
Case 4: 81.7%	75 F	3000 gpm	5400 gpm

EQ

Maximum Power Level	LPSW Temp.	LPI Flow	Requirements LPSW Flow
*Case 1: 94.4%	55 F	5650 gpm	5400 gpm
Case 2: 94.4%	75 F	5650 gpm	5400 gpm
Case 3: 90.2%	55 F	3000 gpm	5400 gpm
*Case 4: 75.6%	75 F	3000 gpm	5400 gpm

* = Limiting power level for each case

Unit 2

Plant Cooldown

100% power operation is justified with LPSW flow of 5400 gpm to the 2B LPI cooler, and provided the reactor coolant pump is tripped upon reaching 160 F.

LOCA

Maximum Power Level		LPSW Temp	LPSW Flow	Requirements LPI Flow
Case 1:	94.6%	55 F	5400 gpm	5650 gpm
Case 2:	89.1%	75 F	5400 gpm	5650 gpm
Case 3:	88.6%	55 F	5400 gpm	3000 gpm
Case 4:	83.7%	75 F	5400 gpm	3000 gpm

EQ

Maximum Power Level		LPSW Temp	LPSW Flow	Requirements LPI Flow
*Case 1:	81.7%	55 F	5400 gpm	5650 gpm
*Case 2:	66.1%	75 F	5400 gpm	5650 gpm
*Case 3:	67.7%	55 F	5400 gpm	3000 gpm
*Case 4:	53.7%	75 F	5400 gpm	3000 gpm

* = Limiting power level for each case

VI. ASSUMPTIONS AND CONSERVATISMS

1. A single failure is assumed to take out the best LPI cooler and the best RB cooler in all calculations. (Conservatism)

2. The improvement in RB cooler performance due to 55 F LPSW temperature has not been taken credit for in any calculations. (Conservatism)

3. Maximum worst case decay heat corresponding to 102% power operation for three consecutive 440 EFPD fuel cycles with 21-day refueling outages. (Conservatism)

4. The passive heat sinks in the existing containment analysis are known to be very conservative. (Conservatism)

5. For the cases where LPSW temperature is taken to be 75 F, the real temperature is less for the entire year in the absence of abnormal weather. The current temperature is 51.5 F. (Conservatism)

6. All cooler performance is reduced by 4% to account for additional projected fouling prior to the next cooler cleaning. (Assumption/ Conservatism)

7. The flowrate through an LPI cooler with two LPI pumps is 5650 gpm based on a recent Unit 3 test. This flowrate is used in calculations as applicable to all coolers. No significant difference is expected based on an engineering review of the systems. (Assumption)

8. The rate of energy release to the Reactor Building from stored energy in the reactor coolant and structures is assumed to be negligible (beyond a time as specified below) in comparison to decay heat in the EQ analysis methodology. This assumption is based on an assessment of the integrated energy delivered to the Reactor Building beginning at the time the EQ criterion requires that power be restricted to less than 100% power. This time is approximately 10,000 seconds for Unit 1 and 6,000 seconds for Unit 2. By these times a large percentage of the stored energy will have already been delivered to the Reactor Building. (Assumption)

9. The long-term post-LOCA Reactor Building sump and vapor temperatures may deviate slightly from those in Figures IV-2 and IV-3 when considering the impact of degraded coolers. The reduction in power level accounts for this effect, but small deviations during the 24-hour duration cannot be ruled out. (Assumption)

VII. CONCLUSIONS

Utilizing the cooler performance data in Section III in the Section IV methodology, maximum allowable power levels can be calculated for different cases involving 1 or 2 LPI pumps and 75 F or 55 F LPSW temperatures. For each case the lowest maximum allowable power level is determined by the LOCA or EQ criterion, whichever is most limiting. The methodology incorporates appropriate assumptions to ensure a conservative calculation of the maximum allowable power level.

It must be recognized that without the single failure assumption that disables the second LPI cooler train, the current degraded cooling capability situation is not a safety problem and supports 100% power operation.

By making appropriate revisions to station operating procedures in order to ensure the assumed LPI and LPSW flows, and by setting the high flux trip setpoint equal to the maximum allowable power level, the consequences of the design basis accidents will be no worse than the existing FSAR analyses.

VIII. RELATED CONCERNS

Reactor Building Pressure Response

The degraded cooling capability has little impact on the peak Reactor Building pressure. An analysis has been documented in the FSAR for which no credit was taken for either the Reactor Building Spray System or the RB coolers. The results of this analysis (FSAR Section 15.14.5) is an increase in the peak Reactor Building pressure from 53.5 psig (corresponding to one spray train and two RB coolers) to 54.6 psig, well below the Reactor Building design pressure of 59.0 psig. In the longer term the calculations performed for the LOCA criterion ensure that Reactor Building pressure will continue to decrease after 30 minutes (decay heat less than energy removal), as shown in Figure IV-1.

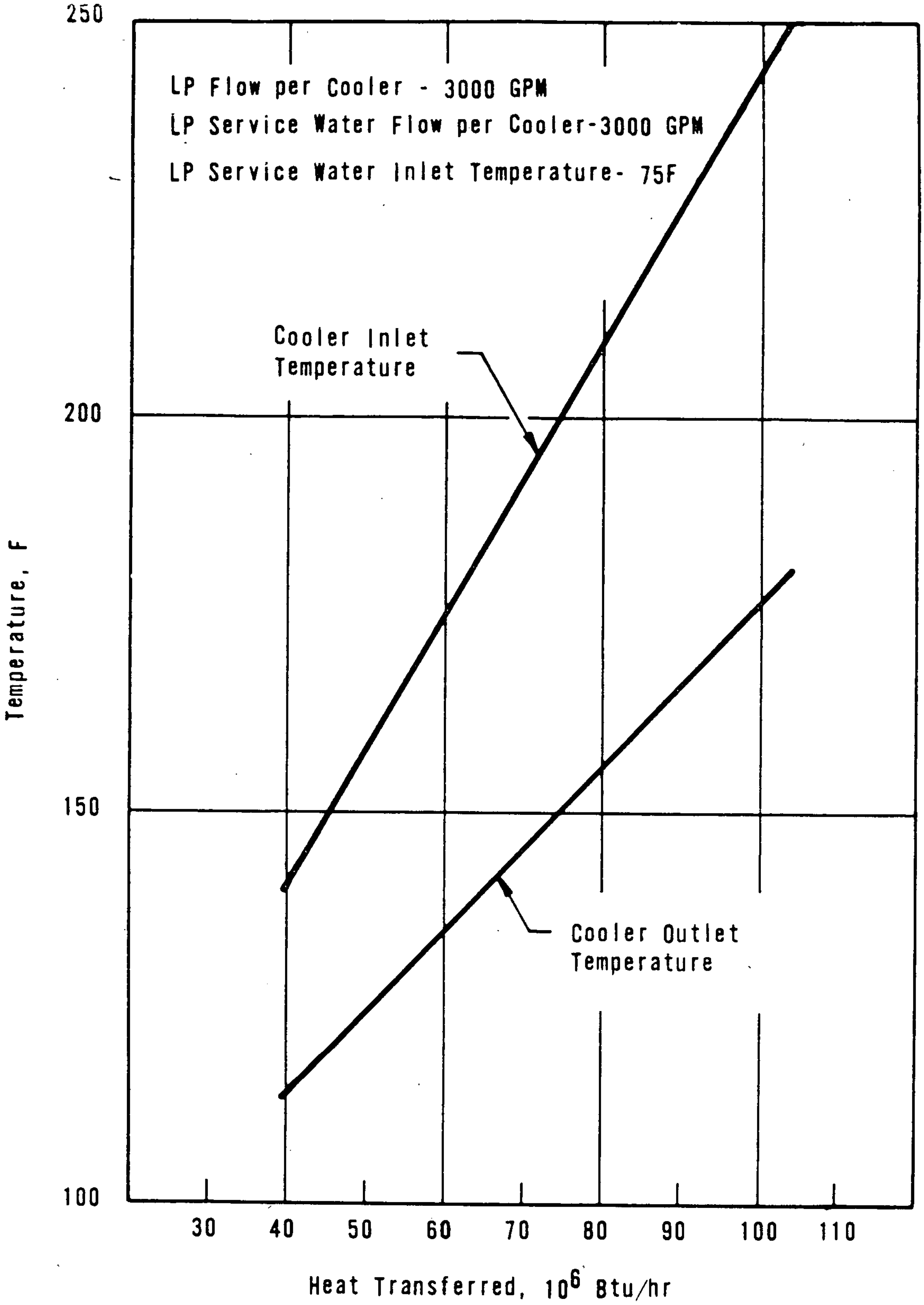
Core Cooling Following LOCA

The peak clad temperature occurs at approximately one minute following a loss of coolant accident. The core is completely reflooded within minutes. Since the impact of degraded LPI cooler only occurs after realignment for sump recirculation, which occurs at a much later time, there is no impact on post-LOCA core cooling.

Reactor Building Spray System

In the event that a single failure requires that both LPI pumps be aligned to take suction from one sump suction pipe, it will be necessary to terminate Reactor Building sprays. This is due to NPSH limitations on the spray pumps. Since the Reactor Building Spray System is redundant to the Reactor Building Cooling System, and since the acceptability of the RB cooler performance has been explicitly analyzed, it is acceptable to terminate the sprays if required.

Figure III-1

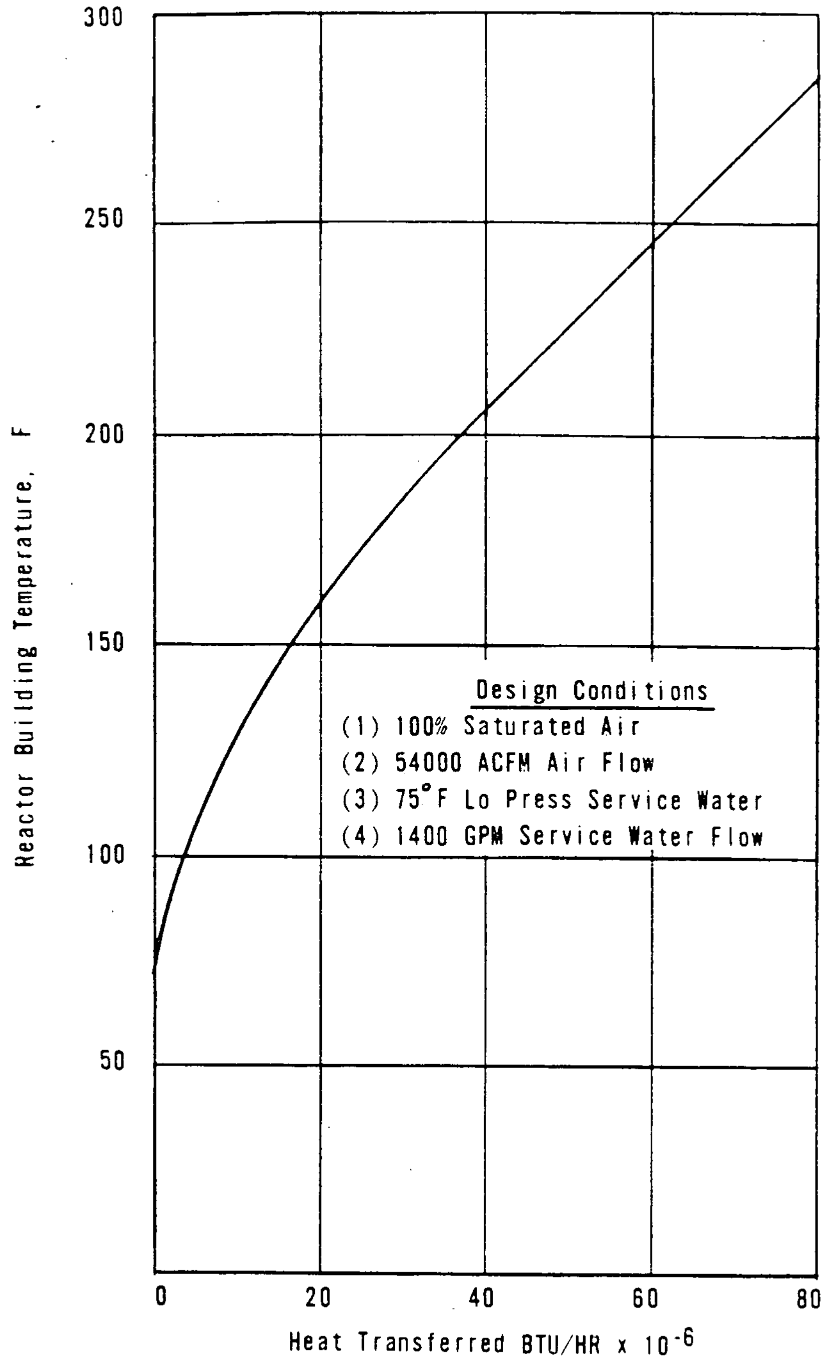


LOW PRESSURE INJECTION
COOLER CAPACITY

OCONEE NUCLEAR STATION

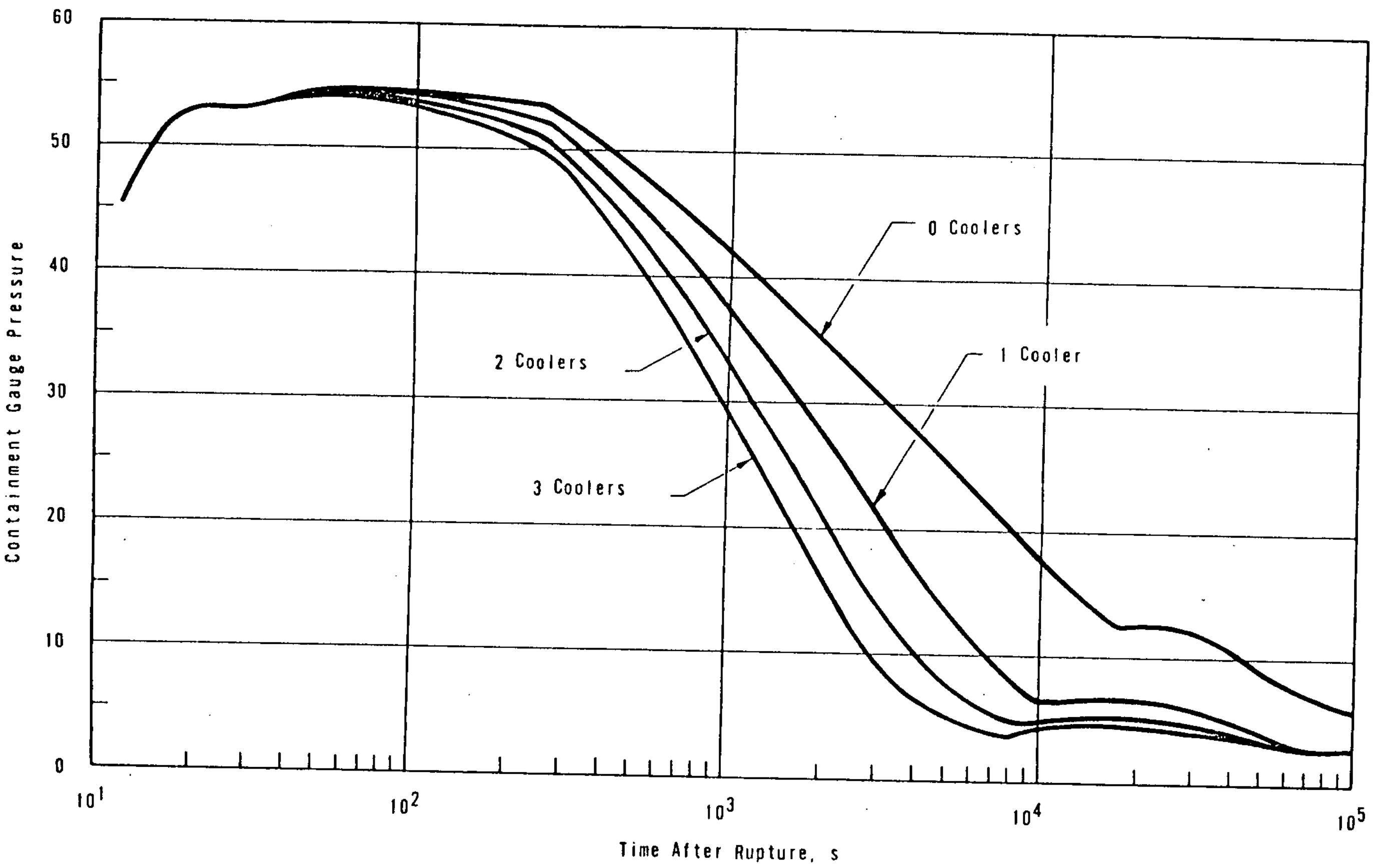
Figure 6.3-4

Figure III-2



REACTOR BUILDING COOLER
HEAT REMOVAL CAPACITY
OCONEE NUCLEAR STATION
Figure 6.2-3

Figure IV-1

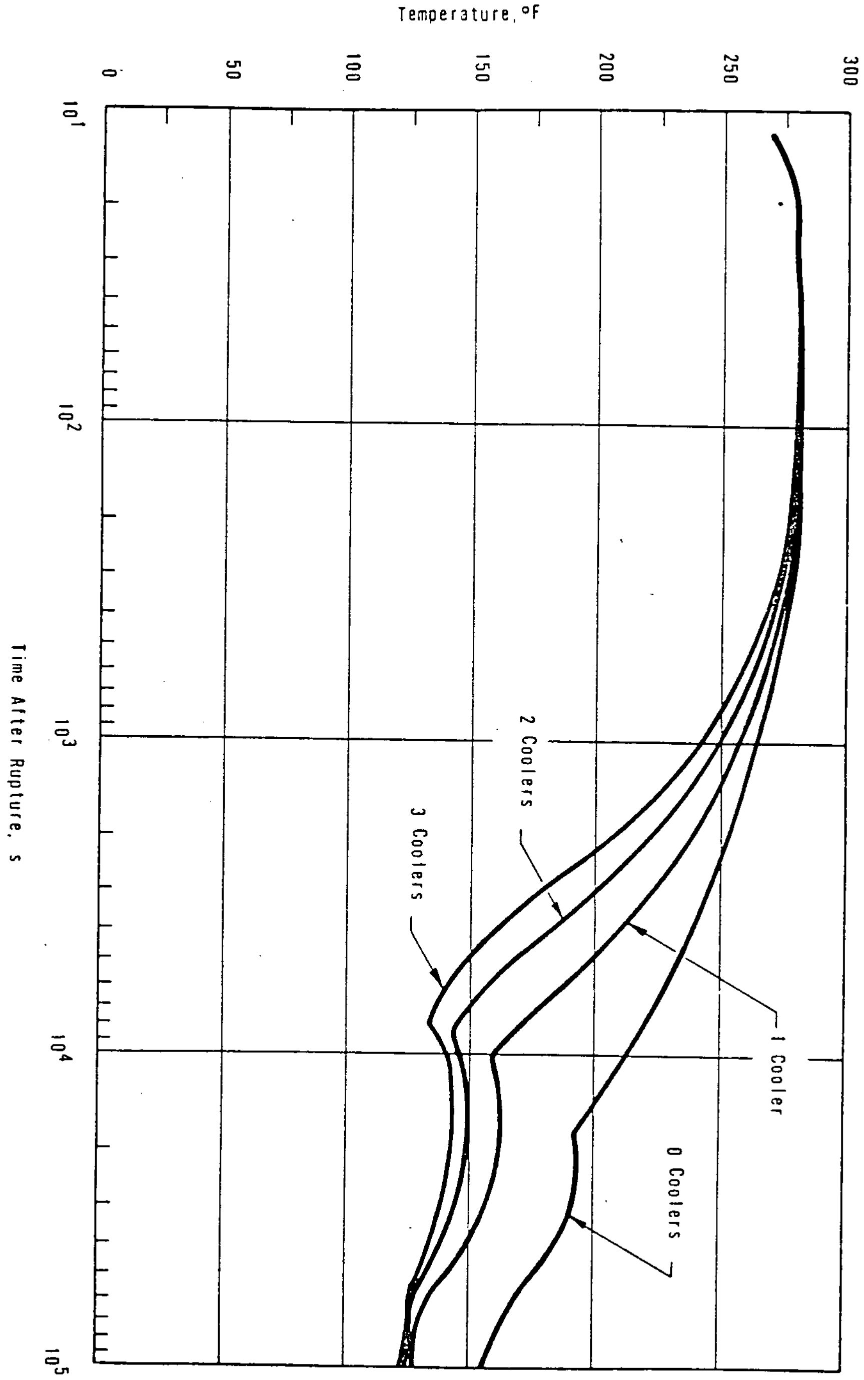


VARIATION OF REACTOR BUILDING PRESSURE VERSUS TIME FOR THE DBA 5 FT^2 BREAK WITH 3.2.1 OR 0 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION

Figure 15.14-25a

Figure IV-2

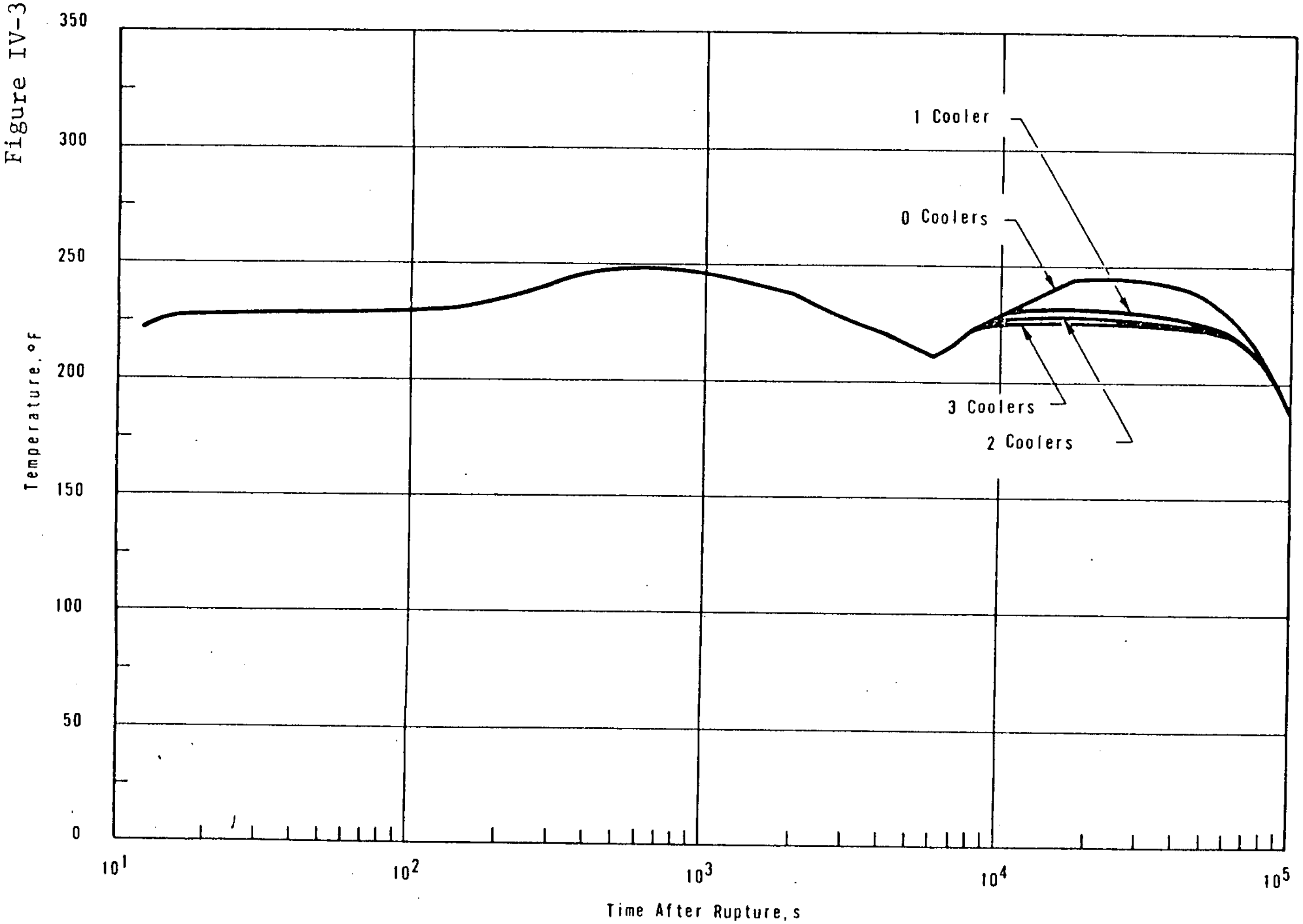


VARIATION OF REACTOR BUILDING VAPOR TEMPERATURE VERSUS TIME FOR THE DBA 5 FT^2 BREAK WITH 3, 2, 1 OR 0 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION

Figure 15.14-25b

Figure IV-3



VARIATION OF REACTOR BUILDING SUMP TEMPERATURE VERSUS TIME FOR DBA 5 FT^2 BREAK WITH 3, 2, 1 OR 0 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION
Figure 14 - 63f
Rev. 19 5/5/72

ATTACHMENT A

LPI Cooler Performance Methodology

The LPI coolers were tested to evaluate the performance of each cooler. This test data (LPI and LPSW flow rates and temperatures) was combined with cooler specifications and known conditions (number of tubes plugged) to calculate the level of fouling in the coolers. The fouling resistance was then used to determine the heat removal capability of the LPI coolers at varying flow rates and LPSW inlet temperatures.

Specifically, the heat duty (Btu/hr) of the LPI coolers was calculated at varying flow rates and LPSW inlet temperatures using the following procedure:

1. Calculate the test fouling resistance and U (design with new fouling) using the following inputs:

   a. LPI cooler performance data: Adjust LPSW flow rate to satisfy the equation Q-tube = Q-shell.
   b. Heat exchanger specification sheet.
   c. Number of tubes plugged.

2. Calculate the heat duty (Btu/hr) of the LPI coolers at varying flow rates and LPSW inlet temperatures using the following inputs:

   a. Test fouling resistance.
   b. U (design with new fouling).
   c. Heat exchanger specification sheet.
   d. Number of tubes plugged.

The test fouling resistance, U (design with new fouling) and heat duty were calculated using a computer program which is based on standard textbook equations. The accuracy of the computer program was checked in a hand calculation. If the test data did not satisfy the basic equation Q-shell = Q-tube, one of the performance test data variables must be adjusted. This analysis adjusted the LPSW flow to satisfy Q-tube = Q-shell because the LPSW measurements are considered to be the least accurate due to fouling in the LPSW piping.

ATTACHMENT B

Reactor Building Cooler Performance Methodology

Testing of the Reactor Building (RB) coolers is performed under normal operating conditions. Two RB coolers are tested at a time, while they are both in service. Maximum building heat load is required to perform the test in order to maximize the air-side temperature differences, and thus minimize any air-side temperature measurement errors. As a result, the test is normally conducted with the unit either at full power, or at hot shutdown conditions. Four hours are required for each test of 2 RB coolers. The third RB cooler is then placed in service, and one of the others is taken out of service. A minimum of 12 hours is required for flow and temperature stabilization prior to testing the third RB cooler. Permanently installed plant instrumentation is used to acquire the data, except for the water-side temperature measurements, where the temperature differences are very small. Here, temporarily installed, precision RTD's are used to measure the temperatures more accurately. All instrument calibrations are verified to be current prior to testing, and are calibrated, if necessary, to insure an accurate test. The actual data acquired during the test includes:

- LPSW inlet flow
- LPSW outlet flow
- LPSW inlet temperature
- LPSW outlet temperature
- Air inlet temperature
- Air outlet temperature

The analysis of the RB cooler test data is done using a computer program developed specifically for the Oconee RB coolers. The program was formulated using the actual test data for these coolers provided by American Air Filter in their Topical Report, AAF-TR-7101A, "Design and Testing of Fan Cooler Systems for Nuclear Applications." The program has been validated against computer runs made by American Air Filter for both "normal" and "emergency" conditions. It uses the above test data as inputs, and provides the following outputs:

- Fouling factors
- Normal Q's (heat duty in BTU/Hr)
- Emergency Q's (heat duty in BTU/Hr)

The fouling factors relate to the extent of fouling which has occurred in the RB coolers, and are based on the actual condition of the coolers, as tested. The normal and emergency Q's represent the total energy which would be transferred by the RB coolers under normal and emergency design conditions, respectively, given the calculated fouling factors. The emergency Q's are the numbers which are used in the determination of the maximum allowable power level.