MAY 18 1984

Docket No.: 50-382

Mr. R. S. Leddick
Senior Vice President - Nuclear Operations
Louisiana Power and Light Company
142 Delaronde Street
Post Office Box 6008
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Dear Mr. Leddick:

Subject: Request for Additional Information

As a result of the staff's review of the Waterford Technical Specifications we find that additional information is needed to complete our evaluation. The specific information is contained in Enclosure 1.

In order to support your licensing schedule, please provide the requested information within seven days of the date of this letter. These items have been previously discussed with members of your staff, however, we are forwarding the staff's concerns to ensure that there is complete documentation. If you cannot meet this schedule or if discussion or clarification of the enclosed request is necessary contact the project manager, J. Wilson at (301) 492-7702.

Sincerely,

G. W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure:
As stated

cc: See next page

CONCURRENCES:

DL:LB#3	DL:LB#3
JWilson:es	GKnighton
5/17/84	5/17/84

DIST:

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ENCLOSURE 1

1. Reactor Protective Instrumentation Setpoints (Table 2.2-1, Section 2.2, page 2-3)

In reviewing the Reactor Protective Instrumentation Setpoint Table, which is used to determine the relationship between the Reactor Protection Instrumentation Trip Setpoints, the allowable values and the values of these parameters which are used in the safety analyses, the following discrepancies were observed:

See Attachment 1-1

Please resolve the above listed discrepancies.

2. Reactor Coolant System Process Variables LCOs

Are the values used for process variable LCOs indicated values from the instrumentation or the actual values in the systems? If they are actual values, please explain how instrument uncertainly is accounted for when determining if an LCO is met or exceeded.

3. Moderator Temperature Coefficient (Section 3.1.1.3, page 3/4 1-4)

Both the loss of condenser vacuum and the feedwater line break events were analyzed at full power with a moderator temperature

ATTACHMENT 1-1

	Table 2.2-1		Table 15.0-3		FSAR Text
	Trip Setpoint	Allowable Value	Analysis Setpoint	Uncertainty	
Local Power Density-High	≤21.0 kw/ft	≤21.0 kw/ft	-	-	13.4 kw/ft
DNBR-Low	≥1.205	≥1.205	1.19	-	1.19
Steam Generator Level-High	≤87.7%	≤88.4%	-	-	Chapter 7 90%

coefficient of 0.0. The technical specifications (3.1.1.3) permit plant operation at 70% power with a moderator temperature coefficient of +0.2 x E-4 $\Delta k/k/°F$. Are the events analyzed at full power with a moderator coefficient of 0.0 be more limiting than at 70% power with a moderator coefficient of +0.2 x E-4 $\Delta k/k/°F$?

4. Boron Dilution (Section 3.1.2.9, page 3/4 1-15, 16, 17)

The Chapter 15 analysis for a boron dilution event relies on operator actions and safety-related alarms; however, there are no technical specifications for the alarm availability, setpoint, or surveillance. Absent this technical specifications, describe what assurance exist that this equipment will always be available and will be properly maintained to meet the Chapter 15 accident analysis assumptions. Also, provide bases for the monitoring frequencies for boron dilution detection listed in table 3.1-1.

5. RPS/ESF Response Times (Table 3.3-2, page 3/4 3-8 and Table 3.3-5, page 3/4 3-23)

Provide the bases for RPS/ESF response times listed in these tables or refer to the assumptions made in Chapter 15 of FSAR.

6. Steam Generator Water Level (Section 3/4.4)

Explain why there is no LCO on the steam generator water level. What assurance is there that the steam generator water level will not exceed the values assumed in the safety analyses?

7. Operability of the Steam Generators (Section 4.4.1.2.3 and 4.4.1.3.2, page 3/4 4-2 and 3/4 4-4)

These surveillance requirements state that the required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥10% of wide range indication at least once per 12 hours. Provide the bases for the 10% steam generator water level.

8. Pressurizer (Section 3.4.3, page 3/4 4-9)

The technical specification for pressurizer level during steady-state reactor operation is set between 350 and 900 ft3. The Chapter 15 transient and accident events assumed 370 and 800 ft3. Please justify how your safety analysis assumptions for pressurizer level bound the levels allow by your proposed technical specifications.

9. Auxiliary Pressurizer Spray System (Section 3/4.4)

The current Waterford technical specifications do not include a section to address limiting conditions for operation and surveillance requirements on the auxiliary pressurizer spray system (APSS). It is the staff's understanding that the APSS is required for RCS depressurization during plant shutdown per the requirement of the BTP RSB 5-1 (i.e., plant cooldown using only safety-related equipment) and during post-SGTR operation. The issue of whether the APSS is required for mitigation of the SGTR or for RSB BTP 5-1 is a license condition for Waterford 3. Does the applicant intend to develop appropriate technical specifications for the APSS if the resolution of this issue shows that this system is necessary?

10. Overpressure Protection System (Section 3.4.8.3, page 3/4 4-34)

The technical specification for overpressure protection systems (Section 3.4.8.3, page 3/4 4-34) references the suction line relief valves as SI-406A and SI-406B. This, we understand, is the Combustion Engineering numbering system. Section 9.3.6.2.2, page 9.3-49 refers to these valves as SI-486 and SI-487. We understand these are LPL numbers. What set of valve numbers is correct? How have you assured yourselves that there is no duplication of valve numbers as a result of the different valve numbering systems?

11. Overpressure Protection Systems (Section 3.4.8.3, page 3/4 4-35)

Section 4.4.8.3.1 states that each shutdown cooling system suction line relief valve shall be demonstrated operable by verifying that each valve in the suction path between the RCS and the shutdown cooling relief valve is key-locked open in the control room at least once per 12 hours. Could the auto-closure interlock override the key-locked open isolation valves so that the SDC system isolation valves could be closed when RCS pressure exceeds 700 psig? Otherwise, explain how the system design precludes a possible event V.

12. Reactor Coolant System Vents (Section 3.4.10, page 3/4 4-37)

The current Waterford technical specifications do not ensure the availability of the RCS vents during plant operation. It is the staff's understanding that the applicant intends to take credit for RCS vents for RCS depressurization during safe shutdown per BTP RSB 5-1. Does the applicant intend to modify the existing Technical Specification for the RCS vents if the on-going assessment shows that this system is necessary for meeting the RSB BTP 5-1 positions?

13. Safety Injection Tanks (Section 3.5.1, page 3/4 5-1)

Section 3/4.5.1 describes the modes of operation for the safety injection tanks. The basis for this item implies that the values in the Technical Specification were chosen for compliance with the accident analyses. Address why there are no specifications for the coolant temperature in SIT. Otherwise, justify why the SIT coolant temperature assumed in the ECCS analyses bounds the maximum temperature the SIT could attain.

14. Atmospheric Steam Dump Valves (Section 3/4.7)

The current Waterford technical specifications do not include a section to address limiting conditions for operation and surveillance requirements on the atmospheric steam dump valves (ADVs).

Since the ADVs are required during initial phase of plant shutdown per the requirements of the BTP RSB 5-1 (i.e., plant cooldown using only safety-related equipment), and we understand your FSAR Chapter 15 steam generator tube rupture analysis takes credit for these components, explain what assurances exist in the plant that these components will always be operable in accordance with the assumptions made in the safety analyses.

Similarly, the Staff and Commission concluded it was acceptable to defer a decision on the need to install PORVs in your plant based, in part, on the CE PRA study performed for your plant. This PRA placed high reliability on the availability of the ADVs to affect decay heat removal. In responding to the above question, please address how the assurances you are providing are consistent with the reliability assumptions made in your PRA.

15. <u>Special Test Exceptions, Reactor Coolant Loops</u> (Section 3/4.10.3 page 3/4 10-3)

This technical specification permits plant operation up to 5% thermal power on fission heat without any reactor coolant pumps operating for startup or physics tests. What safety analyses have been conducted that demonstrate that transients or accidents initiated from this operating condition would be acceptable for Waterford 3? Both the steady state and transient reactor coolant system temperature profiles, margin to saturation, core DNBR, and thermal-hydraulic stability should be assessed. The acceptability of the reactor protective system setpoints during various transients and accidents initiated from this condition must also be justified.