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MIDDLE SOUTH UTILITIES SYSTEM

May 29, 1984

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Director of Nuclear Reactor Regulation
Attention: Mr. G.W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Waterford SES Unit 3
Docket No. 50-382
Response to RSB Questions on
Technical Specifications

REFERENCE: Letter dated May 18, 1984 from
Knighton (NRC) to Leddick (LP&L)

Dear Sir:

In your referenced letter you requested additional information on the Waterford 3 Technical Specifications resulting from a reevaluation by the Reactor Systems Branch.

As you know, LP&L has met with RSB personnel to discuss these additional questions. Enclosed please find our response as requested. We trust this is sufficient information to close out your reevaluation.

Should you require further information in this matter please contact Mike Meisner at (504) 363-8938.

Yours very truly,

K W Cook

K.W. Cook
Nuclear Support & Licensing Manager

KWC/MJM/pco

Enclosure

cc: E.L. Blake, W.M. Stevenson, J.T. Collins, D.M. Crutchfield,
J. Wilson, G.L. Constable, L.B. Marsh, C.Y. Liang

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QUESTION:

1. <u>Reactor Protective Instrumentation Setpoints</u> 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.

RESPONSE:

<u>Local Power Density-High</u>

The case for the single part length CEA drop presented in the FSAR (15.4.1.3) did not explicitly use an analysis setpoint based on local power density. The case was tripped at the latest possible time that still resulted in a peak centerline temperature below that corresponding to centerline melt. The actual CPC trip time would have occurred sooner.

The CPCs calculate local power density during a PLCEA drop using core power, radial and axial peaks and a CEAC penalty factor for the misaligned CEA. Conservatisms are included in the CPCs to assure that the plant will trip in time to prevent fuel centerline melt. These conservatisms include the effects of modelling and measurement uncertainties that affect the local power density calculation and are applied to the CPC addressable constants (BERR3 and BERR4). Thus, a technical specification limit of $\leq$ 21kw/ft is acceptable, since the uncertainties are included elsewhere in the CPCs.

<u>DNBR-Low</u>

The change in DNBR limit from 1.19 to 1.20 was the result of an NRC imposed penalty due to fuel spacer grid configuration. Both the Core Operating Limit Supervisory System (COLSS) and Core Protection Calculators (CPCs) were adjusted to conservatively incorporate this penalty. The adjustments to COLSS and CPCs preserve the conclusions of Chapter 15 analyses which are:
(i) the calculated fuel failures based on DNB are conservative, and
(ii) the DNBR SAFDL is not violated for Anticipated Operational Occurrences.

If a Chapter 15 event is reported to have a minimum DNBR greater than or equal to 1.19, the event will actually have a minimum DNBR greater than or equal to 1.20 due to the extra penalty applied to COLSS and CPC.

Steam Generator Level-High

An increase in feedwater flow is hypothesized to be caused by:

a) a steam generator level instrument failing low;
b) failure of the feedwater control system (FWCS) causing a further opening of the feedwater control valve(s) or an increase in feedwater pump speed;
c) loss of instrument air to the feedwater control valve(s); or
d) startup of one or more emergency feedwater (EFW) pumps.

These events, are assumed to be mitigated as follows:

a) Upon turbine trip, each FWCS is assumed to automatically reduce the feedwater flowrate by closing the main feedwater control valves and opening the 5% bypass valves (FSAR Section 7.7.1.3). The operator then has sufficient time to terminate the event.

b) The main feedwater pumps are tripped automatically on steam generator high level signal (FSAR Section 10.4.7.5).

c) The steam control valves to the main feedwater pump turbines are assumed to fail closed on loss of instrument air, terminating main feedwater flow (FSAR Figure 10.4-2).

d) The inadvertent actuation of emergency feedwater is terminated by operator action ten minutes after receiving a high steam generator level alarm. Assuming all three EFW pumps feed one steam generator produces a flow of 210 ft^3/min. As there is approximately 6,000 ft^3 of steam space in each steam generator, there is in excess of 25 minutes before the level would reach the steam piping.

QUESTION:

2. <u>Reactor Coolant System Process Variable LCOS</u>

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 uncertainty is accounted for when determining if an LCO is met or exceeded.

RESPONSE:

LP&L's practice is to put indicated values for process parameters in the technical specifications. This avoids the need for the operator to provide a correction factor. The indicated values are obtained by applying the appropriate instrument error to the range of initial conditions used in the accident analysis.

QUESTION:

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 coefficient of 0.0×10^{-4} $\Delta k/k/°F$. The technical specifications (3.1.1.3) permit plant operation at 70% power with a moderator temperature coefficient of $+0.2 \times 10^{-4}$ $\Delta k/k/°F$. Are the e nts analyzed at full power with a moderator coefficient of 0.0 more limiting than operating at 70% power with a moderator coefficient of $+0.2 \times 10^{-4}$ $\Delta k/k/°F$?

RESPONSE:

In order to determine if a loss of condenser vacuum (LOCV) or a feedwater line break (FWLB) transient is more limiting operating at full power with a moderator temperature coefficient (MTC) of 0.0, or operating at 70% power with a MTC of $+0.2 \times 10^{-4}$ $\Delta k/k/°F$, a comparison of the core power at the time of trip must be made. The scenario which results in the greater reactor power at the time of trip will result in the greater RCS pressurization, because of the larger amount of energy that has to be removed from the RCS following the reactor trip. Therefore, the transient with the larger core power at the time of reactor trip will be more limiting. The core power at the time of trip is greater for the 100% 0 MTC cases than for the 70% power, $+ 0.2 \times 10^{-4}$ $\Delta k/k/°F$ cases. An analysis supporting this conclusion was performed for another C-E designed NSSS (St. Lucie Unit 2). For this plant, the LOCV analysis was performed with the initial power equal to 102% and an MTC equal to $+0.13 \times 10^{-4}$ $\Delta k/k/°F$. The core power at the time of trip was 103.5% of full power. The maximum RCS pressure of this run was 2738 psia. Another case was run with the initial power equal to 70% and an MTC of $+0.5 \times {}^{-4}$ $\Delta k/k/°F$. The core power at the time of trip was 74.4% of full power. The maximum RCS pressure was 2629 psia or 109 psi lower than the full power case. The increase in the average moderator temperature prior to trip is greater for the LOCV case than for the FWLB case due to the more rapid increase in steam generator pressure. As a result, the core power at the time of trip would be lower for the FWLB than for the LOCV causing the reduction in maximum RCS pressure between the 100 and 70% power cases for the FWLB to be of the same order of magnitude as that for the LOCV. Therefore, it can be concluded that a FWLB or LOCV case analyzed at 100% power with an MTC equal to $0.0 \times {}^{-4}$ $\Delta k/k/°F$ would be more limiting than a FWLB or LOCV case analyzed at 70% power with an MTC equal to $+0.2 \times 10^{-4}$ $\Delta k/k/10°F$.

QUESTION:

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 specification, describe what assurance exists 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.

RESPONSE:

The boron dilution alarm setpoint, and periodic resetting of the alarm have been added to the Technical Specification. The alarm shall be set to ≤2x the existing count rate at intervals dependent on the time after reactor shutdown (starting with a 5-hour interval, extending to weekly). The monitoring frequencies were established to ensure that the time interval between determination of boron concentration was less than the time to loss of shutdown margin depending on the number of charging pumps running. For MODES 3, 4 and 5 the time intervals are at least 15 minutes shorter and for MODE 6 at least 30 minutes shorter to allow time for operator action in accordance with Standard Review Plan 15.4.6. The times to loss of shutdown margin are shown in the responses to questions 211.95 and 211.49.

QUESTION:

5. <u>RPS/ESF response times</u> (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.

RESPONSE:

RPS/ESF response times have been included in the Chapter 15 Safety Analyses. The values presented for the RPS response times (Table 3.3-2 of the Technical Specifications) and for the ESF response times (Table 3.3-5 of the Technical Specifications) have been reviewed against Chapter 15 of the FSAR and have been found to be acceptable.

QUESTION:

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

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

RESPONSE:

An LCO on steam generator water level is not necessary since the safety analysis considers the range of steam generator water levels from the low steam generator level trip setpoint to the high steam generator water level trip setpoint. For events in which the value of this parameter would have a significant impact on the event consequences the value of this parameter is selected to maximize the consequences. For events in which the consequences have a negligible sensitivity to this parameter the analysis may assume an arbitrary initial water level within the specified initial condition space.

QUESTION:

7. <u>Operability of the Steam Generators</u> (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 bases for the 10% steam generator water level as an adequate water level.

RESPONSE:

The minimum water level of the Steam Generator secondary side is being revised to 50% wide range level. Evaluation of the high level requirement on plant operation revealed no impact. The 50% level is sufficient to conservatively account for the following:

a) Provides sufficient heat transfer area to remove maximum decay heat and Reactor Coolant Pump heat without raising the RCS temperature.

b) Provides sufficient inventory to remove decay heat for at least 30 minutes, one hour after reactor shutdown.

c) Account for instrument inaccuracies in a) and b) above.

QUESTION:

8. <u>Pressurizer</u> (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 $ft.^3$ The Chapter 15 transient and accident events assumed 370 and 800 ft^3. Please justify how your safety analysis assumptions for pressurizer level bound the levels allowed by your proposed technical specifications.

RESPONSE:

The Chapter 15 analyses generally assumed pressurizer water volumes as small as 370 $ft.^3$ and as large as 800 $ft.^3$ (see NRC Question 211.33) although in a few cases water volumes of 900 $ft.^3$ and 975 $ft.^3$ were assumed. All events which were analyzed using initial pressurizer water levels within the technical specification range were evaluated as to the impact on the consequences of these events of using the Technical Specification limits on pressurizer water volume. There was found to be negligible impact on the consequences of these events.

QUESTION:

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?

RESPONSE:

In W3P84-1009, dated April 12, 1984, LP&L committed to a resolution of the Staff concern with respect to a potential single failure vulnerability of the Auxiliary Pressurizer Spray (APS) System within six months of receipt of the low power operating license. Resolution will center on the need for APS to satisfy BTP RSB 5-1 and/or SGTR criteria. Should the APS prove necessary in this regard, LP&L agrees that issue resolution shall include a commitment to develop a technical specification to address limiting conditions for operation and surveillance requirements on the APS.

QUESTION:

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. Section 9.3.6.2.2, page 9.3-49 refers to these valves as SI-486 and SI-487. We understand these are LP&L 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?

RESPONSE:

Valves SI-486 and SI-487 are the same as SI-406A and SI-406B. The former numbers, used in the FSAR, are Combustion Engineering assigned numbers whereas the latter, used in the Technical Specifications are LP&L unique identification numbers. At present, the FSAR typically uses EBASCO valve tag numbers. However, some FSAR sections written in early Amendments still cite CE valve number, whereas the Technical Specifications use LP&L numbers and often show the corresponding EBASCO numbers. At the plant, procedures use LP&L numbers, and valves are tagged with LP&L numbers. CE numbers are not used at the plant. Therefore, should duplication within the CE and LP&L numbering systems exist, it will not affect plant operation. Ebasco numbers, which are also shown on valve tags are a different format and cannot be confused with LP&L or CE numbers. Plant procedures (not the FSAR) are used to manipulate valves and controls and thus the different numbering systems should not affect operation of the plant. Flow diagrams retain all three numbering systems, distinguishing each from the others. Retention of the CE numbers in this case is necessary for traceability to engineering documents and QA records.

QUESTION:

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 exceeding 700 psig? Otherwise explain how the system design preclude a possible event V.

RESPONSE:

The term "key-locked open" refers to the type control switch used on these valves. To operate the switch a key must be inserted; then the valve may be opened using the key. When the valve is shut (normal position during operation) the key is removed to prevent inadvertent operation. Operation of the key switch does not affect the automatic features such as isolation on high RCS pressure.

QUESTION:

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 ongoing assessment shows that this system is necessary for meeting the RSB BTP 5-1 positions?

RESPONSE:

The Reactor Coolant System Vents are tied to the resolution of Waterford compliance with BTP RSB 5-1. In W3P84-0505 dated February 29, 1984 LP&L submitted CEN-259 which included an evaluation of conformance with BTP RSB 5-1. In that document it was noted that should the APS be unavailable, the primary depressurization could be accomplished via the safety grade RCS Vents. There presently exists a technical specification (3/4.4.10) on this system which ensures operability of the RCS vents during plant operation. This technical specification will be changed as necessary depending on the results of the NRC's review of BTP RSB 5-1.

QUESTION:

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 bound the maximum temperature the SIT could attain.

RESPONSE:

The LOCA analysis assumes a temperature for the Safety Injection Tanks (SIT) of 120°F, because for these analyses a higher temperature is more adverse. The temperature of 120°F is assumed to be the maximum, since this is the limit on containment air temperature specified in Technical Specification 3/4.6.1.5. The steam line analyses were performed assuming a SIT temperature of 120°F. Calculations have shown that by assuming a SIT temperature of 40°F the average RCS temperature will be lowered by less than 1°F at the time of minimum DNBR. This change in temperature has a negligible effect on the return to power and consequently a negligible effect on the minimum DNBR. Thus, for the steam line break analyses the SIT temperature has a negligible impact on the event consequences when a range of 40°F to 120°F is considered.

QUESTION:

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

The current Waterford technical specification does 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 RSE 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 effect 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.

RESPONSE:

LP&L does not disagree with the intent of developing a technical specification to ensure availability for the Atmospheric Dump Valves. However, the issue is generic. Our position is that development of generic technical specifications requires careful consideration and review on the part of both the NRC and the industry. Unilateral action does not allow for wide review and, in the long run, could be detrimental to other utilities. We suggest, therefore, that this and other generic technical specification development be remanded to the normal Staff process for Standard Technical Specifications including CRGR review.

In the interim, without explicit Technical Specifications governing the ADVs, LP&L still considers these valves to be governed by the Technical Specifications. Section 4.0.5 invokes the requirements of ASME Section XI valve testing in which the ADVs are covered due to their importance to safe shutdown. Note that 4.0.5 d states that Section XI requirements are in addition to other specified surveillance requirements. In addition, the definition of OPERABLE-OPERABILITY requires that a system, subsystem, train, component or device must have all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment also capable of performing their related support function(s). In this manner, the ADVs are incorporated into the Specifications for Reactor Coolant Loops having the basis for decay heat removal via bleeding steam from the steam generator.

As to the PRA work on the PORV issue, it should be noted that many systems were considered in the risk assessment presented in CEN 239 which could impact the calculated core damage frequencies. Some of these systems (e.g. main feedwater system) are not safety grade nor are they included in the Technical Specifications for most plants. The availability values assumed for these systems

(including the ADVs) were based upon historical operational data. The issue of whether or not to Tech Spec these systems to meet the assumed availability values is therefore irrelevant.

At the present time, ADVs are also included in the Containment Isolation Valve Technical Specification 3.6.3, requiring that they be OPERABLE. LP&L does not concur with Containment Systems Branch's inclusion of the ADVs as well as other essential valves in Table 3.6-2, and documented this position in W3P84-0577 dated March 16, 1984. Subjecting ADVs to the Action Requirements of this Specification is felt to be non-conservative with respect to their primary function and is an example of problems that result when sufficient inter-branch review on a generic basis is not done.

QUESTION:

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

This technical specification permits plant operation without any reactor coolant pumps operating up to 5% thermal power on fission heat 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 other related 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.

RESPONSE:

The Special Test Exception in 3/4.10.3 concerning specification 3.4.1.1 is no longer required and will be removed. This was developed to support the low power Natural Circulation testing. However the test methodology was revised and this exception is no longer necessary.