SHAW, PITTMAN, POTTS & TROWBRIDGE

A PARTNERSHIP OF PROFESSIONAL CORPORATIONS

1800 M STREET, N. W.
WASHINGTON, D. C. 20036

DOCKETED
USNRC

'82 SEP 13 A10:26

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

(202) 822-1000

TELECOPIER
(202) 822-1099 & 822-1199

TELEX
89-2693 (SHAWLAW WSH)
CABLE "SHAWLAW"

WRITER'S DIRECT DIAL NUMBER
(202) 822-1051

September 10, 1982

Sheldon J. Wolfe, Esquire
Administrative Judge
Chairman, Atomic Safety and
Licensing Board
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Dr. Harry Foreman
Administrative Judge
Director, Center for
Population Studies
Box 396, Mayo
University of Minnesota
Minneapolis, MN 55455

Dr. Walter H. Jordan
Administrative Judge
881 West Outer Drive
Oak Ridge, TN 37830

In the Matter of
Louisiana Power & Light Company, (Waterford
Steam Electric Station, Unit 3), Docket No. 50-382

Dear Chairman Wolfe and Drs. Foreman and Jordan:

On August 27, 1982, the Staff submitted its response to the Board's Memorandum and Order (Requesting Staff's Affidavit), dated August 12, 1982. The Staff submittal addresses the applicability and review status of Unresolved Generic Safety Issues A-45 and A-46. The Board Memorandum and Order authorized other parties to comment on the Staff's submittal.

Unresolved Generic Safety Issue A-46 does not apply to Waterford 3, as the Staff states in the Affidavit of Tsun-Yung Chang. That affidavit goes on, nevertheless, to provide the Board with the review status of seismic qualification of equipment at Waterford 3. At pages 2 and 3, the review process is

DS03

described, including the Staff's report that an on-site audit at Waterford was scheduled for the week of August 30. Applicant can now report that the audit was conducted as scheduled. At an exit interview with Applicant on September 3, the NRC Staff advised there were no open items and a formal report would be forthcoming.

In the Affidavit of Clifford J. Anderson and Chu-Yu Liang Concerning Unresolved Generic Safety Issue A-45 (Shutdown Decay Heat Removal Requirements) included in the Staff's submittal, it was noted (page 5) that San Onofre Unit 2, a plant with a design similar to that of Waterford 3, submitted a justification for safe interim operation, notwithstanding the incomplete status of the Staff's depressurization capability review, and that the Staff and Commission approved operation of San Onofre after reviewing that justification.

Attached is a copy of a justification for interim operation which was submitted to the Staff on September 9, 1982, by Louisiana Power & Light Company related to Waterford 3. In view of the similarity of Waterford 3 to San Onofre 2 and the outcome of NRC's review of that plant's analogous justification, Applicant anticipates favorable NRC review of the attached document. As is stated in the justification submittal, based on the considerations discussed therein Applicant has concluded that the current Waterford 3 design provides adequate protection for the health and safety of the public, and full power operation is justified while further responses are prepared to the Staff's request for additional information associated with the rapid depressurization and decay heat removal capabilities for Waterford 3.

Accordingly, and taking into account the August 27 submission by the Staff, Applicant maintains that there is a reasonable basis to conclude that Waterford 3 can be operated safely pending resolution of the two unresolved generic safety issues identified by the Board in its Order of August 12, 1982.

Respectfully submitted,

SHAW, PITTMAN, POTTS & TROWBRIDGE

By Ernest L. Blake, Jr.
Ernest L. Blake, Jr.
Counsel for Applicant

Enclosure

cc: Service List attached

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
LOUISIANA POWER & LIGHT COMPANY	)	Docket No. 50-382
	)	
(Waterford Steam Electric Station, Unit 3)	)	

SERVICE LIST

Sheldon J. Wolfe, Esquire
Administrative Judge
Chairman, Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Harry Foreman
Administrative Judge
Director, Center for Population Studies
Box 395, Mayo
University of Minnesota
Minneapolis, MN 55455

Dr. Walter H. Jordan
Administrative Judge
881 West Outer Drive
Oak Ridge, TN 37830

Sherwin E. Turk, Esquire (4)
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Gary Groesch
2257 Bayou Road
New Orleans, LA 70119

Luke B. Fontana, Esquire
824 Esplanade Avenue
New Orleans, LA 70116

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing & Service Section (3)
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

LOUISIANA POWER & LIGHT / 142 DELARONDE STREET
P. O. BOX 6008 • NEW ORLEANS, LOUISIANA 70174 • (504) 366-2345
MIDDLE SOUTH UTILITIES SYSTEM

L. V. MAURIN
Vice President
Nuclear Operations

W3P82-2630
3-A1.01.04
3-A20.20

September 9, 1982

Mr. T. H. Novak
Assistant Director for Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Waterford SES 3
Depressurization and Decay Heat Removal

References: (1) R. L. Tedesco to L. V. Maurin dated 3/27/82
(2) L. V. Maurin to R. L. Tedesco, W3P82-2309, dated 8/27/82

Dear Mr. Novak:

Reference (1) transmitted questions regarding the rapid depressurization and decay heat removal capability of the Waterford 3 design. By reference (2) we indicated that the questions were being addressed by the CE Owners Group and LP&L would respond by August 15, 1983.

Reference (1) also asked that LP&L provide a justification for safe operation should our response not be complete one month prior to fuel load. Consequently, enclosed please find a justification report entitled "A Review of Depressurization and Decay Heat Removal Capabilities of Waterford 3".

Should you have any questions or comments please let me know.

Sincerely,

L V Maurin

L. V. Maurin

LVM/MJM/pco

Enclosure

cc: W. M. Stevenson, E. L. Blake, S. Black

A REVIEW OF

DEPRESSURIZATION AND DECAY HEAT REMOVAL

CAPABILITIES OF WATERFORD 3

1.0 INTRODUCTION

The NRC has requested that Louisiana Power and Light (LP&L) provide an evaluation of the rapid depressurization and decay heat removal capabilities of the Waterford 3 design. LP&L is participating with the CE Owners Group (CEOG) in developing responses to the NRC's questions. The NRC has also requested that LP&L provide a justification for safe operation of the plant at full power during the period of this evaluation. This report provides justification for safe full power operation of Waterford 3 based on the following considerations, which are amplified later in this report:

1. The Waterford 3 NSSS is coupled with a highly reliable, safety grade Emergency Feedwater System (EFWS). The EFWS design for Waterford 3 exhibits a higher level of reliability than most EFWS designs.

2. Waterford 3 is capable of achieving cold shutdown conditions using only safety grade systems, even without offsite power and with an additional single failure.

3. The Waterford 3 steam generator design includes many features which will enhance tube integrity, minimizing concerns associated with operating reactors. Additionally, careful attention to the plant water chemistry program will ensure that the magnitude of the impurity ingress into the steam generators is maintained at a low level. Because of the steam generator water chemistry program and design features which minimize steam generator tube corrosion and stress, LP&L considers that steam generator tube degradation should not be a concern during the period the NRC questions are being addressed.

4. Even if all auxiliary feedwater supply were somehow lost, heat removal could still be achieved by depressurizing the steam generators to allow the use of the low head condensate pumps.

5. Review of probabilistic analyses conducted by the NRC do not show any justification for the addition of Reactor Coolant System (RCS) valves for decay heat removal purposes.

2.0 BACKGROUND

The early CE NSSS designs used Power Operated Relief Valves (PORVs) as nonsafety grade equipment to limit overpressure transients to pressures below the ASME Code safety valve setpoint. This function was intended to reduce challenges to the safety valves, thereby minimizing weepage and avoiding potential leakage following actuation. The PORVs were not intended to prevent a high pressure reactor trip, but rather, were to be used in conjunction with the trip to mitigate the pressure transient.

As each of the early plants became operational, the effectiveness of the pressurizer spray system to limit pressure transients was demonstrated. Consequently, CE was unable to substantiate any advantages to opening PORVs during transients to protect the safety valves from leakage. PORVs were also considered to be counterproductive in light of the PORV leakage problems that had been experienced. Furthermore, system analysis has demonstrated the pressure overshoot above the high pressure trip to be so minimal that, when PORV operation was not credited, the safety valves were still not challenged.

Accordingly, the PORV function during power operation was not considered necessary, and was eliminated from subsequent CE designs.

Recently, a contingency method of core cooling employing once-through flow in the RCS has been advanced as an alternate decay heat removal system. This method would use PORVs in conjunction with the High Pressure Safety Injection (HPSI) pumps and has been referred to as "feed and bleed". In this regard, the Advisory Committee on Reactor Safeguards (ACRS), following its review of CE's System 80, (which is similar to Waterford 3 in this regard) stated:

> "In recent years, the availability of reliable shutdown heat removal capability for a wide range of transients has been recognized to be of great importance to safety. The System 80 design does not include capability for rapid, direct depressurization of the primary system or for any method of heat removal immediately after shutdown which does not require use of the steam generators. In the present design, the steam generators must be operated for heat removal after shutdown when the primary system is at high pressure and temperature. This places extra importance on the reliability of the auxiliary feedwater system used in connection with System 80 steam generators and extra requirements on the integrity of the steam generators. The ACRS believes that special attention should be given to these matters in connection with any plant employing the System 80 design. The Committee also believes that it may be useful to give consideration to the potential for adding valves of a size to facilitate rapid depressurization of the System 80 primary coolant system to allow more direct methods of decay heat removal. The Committee wishes to review this matter further with the cooperation of Combustion Engineering and the NRC Staff."

In meetings with the ACRS and NRC Staff, CE has presented its position and the bases for its design. The NRC has raised a series of concerns regarding this issue and provided a list of questions to CE and applicant utilities. In recognition of the scope of these questions the NRC has requested justification for operation during the period of time the questions are being addressed. The ACRS has agreed with this approach stating that:

> "....while this evaluation should be conducted expeditiously its resolution should not now be a condition for operation of System 80 plants at full power or of plants having similar features."

The CEOG has agreed to sponsor preparation of generic (and some plant specific) responses for affected CE utilities. This submittal provides justification for full power operation of Waterford 3 during the period of time that these questions are being addressed.

3.0 EMERGENCY FEEDWATER SYSTEM RELIABILITY

The Waterford 3 NSSS design is coupled with a safety grade Emergency Feedwater System which has been subjected to extensive development by LP&L, CE, and EBASCO. This sytem in conjunction with the safety grade atmospheric dump valves provides an assured method of RCS heat removal. The EFW system, which is documented in the Waterford 3 FSAR, is a three train system with one train independent of ac power. It is seismic category 1, electrical class 1E and designed to ASME code class 2 and 3. The EFWS design for Waterford 3 exhibits a higher level of reliability than most other EFWS designs. In its Safety Evaluation Report the NRC concluded that the Waterford 3 design satisfied all applicable Commission requirements. Additionally, the staff review of modifications initiated since the accident at Three Mile Island Unit 2 showed an increase in Waterford's EFWS reliability due to the modifications.

Although no quantitative requirement for expected system availability was explicitly imposed, the Waterford 3 EFWS design reflects the high reliability needed to meet the current SRP criteria of unavailabilities in the range of 10^{-4} to 10^{-5} per demand. This conclusion is supported by analyses presented in both the Waterford 3 FSAR and NRC staff analyses referenced on Waterford's Docket No. 50-382 in the "NRC Staff's Answer in Support of Applicant's Motion for Reconsideration of March 18, 1982 Memorandum and Order Raising Sua Sponte Issue", dated April 12, 1982. In this document the staff concludes "that a feed and bleed capability is not necessary as a back-up system to the Waterford Unit 3 EFWS".

4.0 CAPABILITY TO ACHIEVE COLD SHUTDOWN

There are numerous systems available, both within the NSSS design and BOP design for Waterford 3, to perform the various functions necessary to bring the plant to a cold shutdown condition. As a group, these systems provide the operator with the flexibility necessary to cool down and depressurize the plant in a variety of possible situations. The design meets Branch Technical Position RSB 5-1 as documented in Waterford's Safety Evaluation Report, Supplement 2, pg. 5-1. Some of the more significant features of the Waterford 3 design related to shutdown, cooldown, and depressurization capabilities are discussed below.

Normal Shutdown:

Under the vast majority of situations, the same systems used for power generation will be employed for plant cooldown. In these cases primary coolant is circulated through the RCS using the reactor coolant pumps. Steam is drawn from the steam generators, bypasses the turbine and is rejected to the main condenser. The main feedwater and condensate systems are used to return the condenser inventory to the steam generators. RCS heat removal is maintained with the steam generators. RCS pressure is maintained with the pressurizer, using the normal heater and spray control systems.

Shutdown with Heat Rejection to Atmosphere:

In the event that the main condenser or associated systems are unavailable, steam may be rejected directly to atmosphere. Either of two safety grade steam generator atmospheric dump valves located upstream of the MSIVs may be operated manually to bleed steam. Makeup water to the steam generator is supplied from either the Main Feedwater System or the safety grade EFWS. This system provides sufficient inventory to allow for plant cooldown (i.e. sensible heat removal) and decay heat removal for a period of time in excess of 24 hours. Additional makeup from other site sources allows for extended operations.

Natural Circulation:

Central to the accomplishment of the basic safety function of Core Heat Removal is the ability to transport reactor coolant to a heat sink where RCS Heat Removal can be accomplished. Reactor coolant pump forced circulation and heat transfer to the steam generators is the preferred mode of operation for residual heat removal whenever plant temperatures and pressures are above the shutdown cooling system (SDCS) entry conditions. Subcooled natural circulation provides an effective alternate means for controlled core cooling, using the steam generators, for extended periods of time if the reactor coolant pumps are unavailable. Two-phase natural circulation and reflux cooling will also occur to provide adequate core cooling following transients which result in loss of RCS inventory and/or subcooling.

Component elevations of Waterford 3 are such that satisfactory natural circulation for decay heat removal is obtained as a result of density differences between the bottom of the core and the top of the steam generator tube sheet, an elevation head of approximately 25 feet. An additional small contribution to natural circulation flow rate is the density difference obtained as the coolant passes throught the steam generator U-tubes. Additionally, several systems design features have been incorporated to assure the maintenance of natural circulation flow. A redundant pressurizer heater capacity of 150 KW from each diesel generator is available to maintain system subcooling. A reactor coolant gas vent system is provided to allow the purging of noncondensible gases should they form. Additionally, natural circulation plant performance will be extensively tested during the startup period of San Onofre Unit 2 and Waterford Unit 3.

When in natural circulation, the main pressurizer spray system is unavailable. The safety grade auxiliary spray from the charging system provides for system depressurization under these conditions. This system has been modified to provide an independent manual bypass. Thermal shock considerations are addressed by the use of a thermal sleeve in the spray nozzle. CE recommends use of the auxiliary spray system for primary depressurization whenever the main pressurizer spray system is unavailable.

In summary, the Waterford 3 design meets Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System" as described above. Waterford 3 can be brought to SDCS initiation in less than 36 hours using only seismic category 1 equipment, assuming the most limiting single failure, and with only onsite or only offsite power available.

5.0 STEAM GENERATOR INTEGRITY

The 3410 MWt steam generators are of an improved design selected to mitigate or resolve operating problems which have been experienced with U-tube steam generators of the recirculation type. The general arrangement is similar to currently operating 2570 MWt CE systems including a number of design improvements and retained features to assure improved operational reliability and maintenance of integrity for decay heat removal after reactor shutdown.

The 3410 MWt steam generator is of the vertical U-tube, natural recirculation, noneconomizer type and is somewhat larger than the earlier 2570 MWt steam generator and contains approximately 9,350 tubes instead of 8,400 tubes.

The design as it affects secondary side hydraulics has been improved to remove areas of possible localized dryout. This has been accomplished by a number of modifications in the tube bend region:

1. The vertical tube spacer strips have been separated from the diagonal "bat wing" tube supports.

2. The "bat wing" supports have been lowered to avoid intersecting the tube bends.

3. The tube supports in the small radius bend region have been located below the bends.

4. The vertical tube spacer strips are now provided with large "punchouts" to enhance cross flow freedom.

5. The former drilled upper tube support plates have been replaced with partial "eggcrate" type supports.

Thus all tube supports are of the "eggcrate" or lattice type to promote freedom of vertical as well as cross flow.

The elimination of the drilled upper tube support plates will mitigate the denting problems previously experienced in this region.

The Inconel 600 mil annealed tubing is specified, controlled and tested in a manner to preclude sensitivity to stress corrosion cracking or intergranular attack. Subsequent CE shop tube fabrication practices utilize carefully controlled and proven techniques to minimize residual tube stress, a contributor to stress corrosion cracking. These include:

1. The bending techniques used are selected to minimize residual tube stress. CE has historically used a relatively large tube bending radius for the inner tube rows.

2. CE uses the explosive technique for placing the tube in contact with the tubesheet for the full tubesheet thickness. This eliminates the tube-to-tubesheet crevice which has caused corrosion problems in this region, such as stress cracking and intergranular attack.

The steam generator design allows for sludge lancing to periodically remove accumulations of solids from the upper tubesheet face. These sludge accumulations have been the site of tube pitting type attack.

CE utilizes a mechanical joint between the primary head divider plate and its juncture with the tubesheet and primary head. This eliminates the possibility of the differential growth and deflection between these members causing tubesheet clad separation and tube damage which has occurred in nonCE units.

The 3410 MWt design utilizes large top discharge elbows for the main/auxiliary feedwater inlet sparger. In addition the drain time of this sparger ring has been increased by a sealing device located between the sparger and the feedwater inlet nozzle. Thus water hammer potential with possible feedwater line damage is reduced.

The integrity of the steam generator tubing is also protected through the use of strict controls on the steam generator water chemistry. The chemical environment of the steam generator secondary side is monitored and controlled during all phases of plant operations including power operation, startup, shutdown, and maintenance outages.

Steam generator chemistry is maintained through a combination of control of impurities delivered to the steam generator, monitoring and controlling the chemical environment within the steam generator, and removal of any materials which may be introduced. Through feedtrain features and procedures, including a high integrity condenser, startup recirculation, and chemical addition, the magnitude of impurity ingress into the steam generator is maintained at a low level. In addition, the Waterford 3 design has provisions for prestartup cleanup of the main feedwater system by flushing to the steam generator blowdown system. A chemistry control program is employed to assure that secondary water chemistry is maintained within appropriate control bounds during operation and that timely corrective actions are taken in the event abnormal chemistry occurs. An all volatile treatment water chemistry is utilized for the secondary systems. This method of secondary chemistry control precludes tube corrosion and related problems due to the chemical additives, and it minimizes the amount of sludge deposited within the steam generator. Routine corrective actions for abnormal chemistry include increasing the steam generator blowdown rate, adjustment of chemical addition rates, and more extensive monitoring of steam generator chemistry. For severe upset conditions, power reduction and/or plant shutdown is specified. Continuous sampling of and chemical addition to the steam generator monitors the effectiveness of feedtrain impurity controls and maintains a chemical environment conducive to low corrosion rates within the steam generator. Finally, steam generator blowdown, supplemented by fill and drain when required, serves to remove those impurities which are introduced. By minimizing contaminant ingress, monitoring system performance, and taking corrective action when necessary, chemistry related challenges to the integrity of the steam generator tubes are minimized.

During accident response conditions, water supplied to the steam generator by the Emergency Feedwater System originates in the condensate storage pool. This makeup quality water is chemically treated and its use will not challenge the steam generator tube integrity. In the quite unlikely event that water must be supplied from alternate sources during the accident (auxiliary Component Cooling Water System) it is not anticipated that even this impure water will cause tube failure in the time frame of the accident and subsequent plant cooldown.

In summary it is considered that the design, material and manufacturing features discussed above, along with appropriate chemistry control, will assure improved steam generator tube integrity. LP&L further considers that steam generator tube degradation should not be a concern during the period the NRC questions are being addressed.

6.0 CONTINGENCY DECAY HEAT REMOVAL (DHR)

The Waterford 3 design meets current licensing criteria with regard to DHR capabilities. The consideration of additional RCS valves for DHR essentially addressed contingency (or "last resort") capabilities that go beyond existing design bases. In this regard it is significant to note that a potential already exists for contingency heat removal by depressurizing the steam generators.

The potential mode of plant operation considered is as follows: Followi reactor trip and the very unlikely event of a total loss of all feedwater, the plant could be brought to hot standby using either the secondary safety valves of the atmospheric dump valves. The safety grade steam generator atmospheric dump valves then provide the contingency capability to blowdown and depressurize the steam generator secondary system. At the reduced steam generator pressure the low head condensate pumps could be aligned to deliver feed to the steam generator. Then, with sufficient feedwater and steam flow, continuous decay heat removal could be established at those "off design" conditions.

There appear to be several advantages to steam generator depressurization in preference to primary feed and bleed. These are:

1. <u>The reactor coolant pressure boundary is maintained intact</u>.

   Therefore the potential radiological release to the containment and possibly to the environment is avoided. Any necessary containment entry for repairs would not be impeded. Additionally the large clean-up cost that would be associated with the use of primary feed and bleed is avoided.

2. There is time available for operator action.

   Delivery of secondary makeup to a depressurized steam generator can be accomplished anytime prior to core uncovery, which is estimated to be approximately 90 minutes, to ensure adequate core cooling.

3. Equipment involved is accessible.

   The atmospheric dump valves and various low head pumps are located outside containment where access for maintenance and repair is possible. PORVs on the other hand would be inside containment and virtually inaccessible.

4. Procedures are consistent with normal DHR procedures.

   Normal procedural efforts focus upon restoration of feedwater. Initiation of primary feed and bleed would represent a dramatic departure from this strategy.

The final reason noted above is worthy of elaboration in that it was strongly supported by plant operators during procedure workshops conducted at CE. Plant operators feel that it is highly preferable to continue operation with the steam generators performing the function of RCS Heat Removal, while the functions of RCS Inventory and Pressure Control are being controlled separately. With the initiation of RCS feed and bleed all three safety functions would now rely on a single process with no degree of independent control. The extreme difficulty in dealing with the competing demands of RCS Heat Removal, Pressure and Inventory Control by regulating a single process has been clearly demonstrated at TMI-2 and Ginna.

7.0 PROBABILISTIC JUSTIFICATION (REVIEW OF DRAFT PRA)

The January 29, 1982 memorandum from F. Rowsome and J. Murphy entitled "Feed and Bleed Issue for CE Applicants" included a draft PRA by the NRC Division of Risk Analysis (DRA) attempting to demonstrate that the CE plants which lack a capability for core cooling via feed and bleed operation will not meet the NRC's proposed plant performance guidelines. This guideline is that "the likelihood of a nuclear reactor accident that results in a large-scale core melt should normally be less than one in 10,000 per year of reactor operation". Additionally, the DRA study made a case for incorporating feed and bleed capability to partially alleviate the perceived problem, and presented analysis to show that such a change is cost beneficial to the utilities.

Review of the draft PRA (which has since been characterized as "overstated" by the author) indicates that the recommendations are not well supported by the analyses. This is most succinctly presented in the Staff's "Affidavit of Richard Lobel, Brian Sheron and Ashok Thadani Concerning Feed and Bleed and Emergency Feedwater System Reliability" filed with the Waterford Licensing Board on April 12, 1982:

> "The probability of complete losses of the EFWS in the Rowsome and Murphy memorandum was based on past operating experience as reported in an ORNL report (CR-2497). The staff is aware of ten events in which there was a loss of all emergency feedwater (two more events than are listed in ORNL report CR-2497). An analysis of this past experience as it relates to the Waterford Unit 3 EFWS design results in the following conclusions. First, post-TMI recommendations should greatly lower the probability of occurrence for several of these events. For example, in several events, the EFWS pumps did not start on the automatic initiation signal; safety-related EFWS flow indication must now be provided, and an indicator such as that available at Waterford Unit 3 would alert the operator immediately that the EFWS was inoperable so that he could initiate a timely manual actuation. In addition, human error resulting in a closure of valves in the pump discharge path, such as occurred during the TMI-2 accident, should be less probable now as a result of the required increased surveillance of the EFWS flow path after system testing or extended shutdown.
>
> "Secondly, some of these events were recoverable in less time than the time calculated for loss of the secondary heat sink (i.e. steam generator dryout time). The Waterford Unit 3 steam generators have a relatively large water inventory, which provides the operator with a greater period of time to attempt a manual start in the event that the system does not start automatically. A human error resulting in a closure of valves in the pump discharge path, such as occurred during the TMI-2 accident, should have a high probability of being corrected in the Waterford Unit 3 design before the heat sink is lost.
>
> "Thirdly, some of these events involved a type of failure that could not occur in an EFWS of the Waterford Unit 3 design. For example, several of these events resulted from clogged strainers in the EFWS piping; the strainers will be removed from the Waterford Unit 3 EFWS after startup testing. In addition, one of these events resulted from the interference of a reactor control system with the function of the EFWS. This event was peculiar to reactors designed by Babcock & Wilcox and the problem was fixed following the Crystal River Unit 3 event of February 1980; accordingly, it is not applicable to the Waterford Unit 3 design.

"In conclusion, the Staff's reanalysis of these data, taking into account (1) the post-TMI modifications and corrective actions, (2) the high probability of recovery of some of these events, (3) the limited applicability of some of these events to the Waterford Unit 3 EFWS design leads the Staff to conclude that the Waterford Unit 3 EFWS is subject to a demand failure probability of less than 10^{-4} per demand."

Based on the above comments it is considered that if a corrected analysis was to be performed there would be no apparent justification for plant modification.

8.0 CONCLUSIONS

As requested, a review of the Waterford 3 design has been completed and the following determined:

1. The Waterford 3 NSSS is coupled with a highly reliable emergency feedwater system, with an unavailability in the range of 10^{-4} to 10^{-5} per demand.

2. Waterford 3 is capable of achieving cold shutdown conditions using only safety grade systems, even without offsite power and with an added single failure.

3. The Waterford 3 steam generator water chemistry program and design features will minimize steam generator tube corrosion and stress. Additionally, LP&L considers that steam generator tube degradation should not be a concern during the period the NRC questions are being addressed.

4. Even if all auxiliary feedwater supply were somehow lost, the potential exists for DHR by depressurizing the steam generators to allow use of low head pumps.

5. Contrary to the draft probability analysis developed by DRA, there is no reason to believe that installing PORVs will result in a significant improvement in safety.

Based upon the considerations listed above, it is concluded that the current Waterford 3 design provides adequate protection for the health and safety of the public and full power operation is fully justified while responses are being prepared to the NRC request for additional information associated with the rapid depressurization and decay heat removal capabilities for Waterford 3.