DAIRYLAND POWER COOPERATIVE • P.O. BOX 817 • 2615 EAST AVE. SO. • LA CROSSE, WISCONSIN 54602-0817
(608) 788-4000

July 2, 1985

Mr. John Zwolinski, Chief
Operating Reactor Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Dairyland Power Cooperative
La Crosse Boiling Water Reactor (LACBWR)
Provisional Operating License No. DPR-45
On-Site Property Damage Insurance

Reference: (1) DPC Letter, Shimshak to Zwolinski dated February 7, 1985

Dear Mr. Zwolinski:

Last Thursday we met at the NRC's Bethesda office with members of your staff, the Accident Evaluation Branch and the Office of State Programs for the purpose of discussing in detail a report on post-accident recovery costs. This report and Addendum 1 which was reviewed at the meeting, form the technical basis for our request for relief from burdensome property damage insurance requirements as set forth by 10 CFR 50.54 (w).

Reprints of Addendum 1, in final form, will be sent to you in a few days for record purposes.

During our meeting, we mentioned that a simplified risk assessment program had been conducted a few years ago by Mr. Saul Levine and Dr. Norman Rasmussen. The results of this limited study had been presented to Mr. Harold Denton by them at a meeting held at his office on April 28, 1982.

The material prepared by them on our behalf and used at the meeting was never docketed. Therefore, to satisfy this objective and to accommodate Mr. J. Hulman's interest in the report, we are sending twenty (20) copies of Mr. Saul Levine's April 23, 1982 letter to Mr. James Taylor (including view graph copies) which summarized the program plan and presented conclusions favorable to LACBWR with respect to various major risks.

Ltr Encl
Add: Jerry Hulman AEB 2 2

A001
1/1

In the event that further information is needed about this program, please contact Dr. Norman Rasmussen at the Nuclear Engineering Department, Massachusetts Institute of Technology, Cambridge, Massachusetts 02138, phone 617-253-3802.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

R. E. Shimshak

Richard E. Shimshak, Manager
Special Nuclear Projects

RES:daj

cc: J. Taylor/PGL-5075
J. Parkyn w/o report
O. Hiestand w/report
E. Tremmel w/report
H. Devine w/report
C. Ross w/report
W. Manion w/report
C. Finnan w/report
J. May w/report
R. Mueller w/report
J. Thie w/report
N. Rasmussen w/report

NUS
CORPORATION

910 CLOPPER ROAD
GAITHERSBURG, MARYLAND 20878
(301) 258-6000

CG-SL-19-82
Project No. 3369.02
April 23, 1982

Mr. James Taylor
Assistant General Manager, Power
Dairyland Power Cooperative
Box 817
2615 East Avenue, South
La Crosse, WI 54601

Dear Jim:

Per our recent discussions, I am enclosing a summary of the program plan that was developed to assess LACBWR accident sequence probabilities. The summary reflects, in the most positive way possible at this time, the favorable impression that Norm Rasmussen and I have of the LACBWR facility. I believe, as you suggested, that it will be useful to Harold Denton.

If you have any questions or comments, please do not hesitate to call me.

Sincerely,

Saul

Vice President and
Group Executive
Consulting Group

JC/m

Enclosure

Summary of "Program Plan to Evaluate Accident Sequence Probabilities at the La Crosse Boiling Water Reactor."

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) in the Systematic Evaluation Program (SEP) is currently reviewing eleven early generation nuclear power plants. The purpose of these SEP reviews is to identify areas where plants may need to be upgraded to meet existing regulatory requirements. These requirements, which have evolved over the years, were developed essentially for large power reactors, e.g., of the 400 MWe to 1300 MWe range.

For some reactors, particularly those with a low power level, and an associated low radionuclide inventory, the imposition of backfits, developed to meet requirements for large power reactors as a means of achieving low public risk, may be of questionable validity. This appears to be especially true for the La Crosse Boiling Water Reactor (LACBWR), for which a consequence analysis has been performed by Sandia National Laboratories. That analysis predicts that, even for the most severe type of accidents postulated for large power reactors, no early fatalities and only 125 latent cancer fatalities may occur.

Conversely, even though the public consequences of a severe accident at LACBWR have been predicted to be extremely small, it is important to know whether undesireable events such as core melt accidents and failure of containment, are of fairly low probability. To investigate these questions, Dairyland Power Cooperative invited S. Levine, Vice President and Group Executive, Consulting Group, NUS Corporation, and N. C. Rasmussen, Professor of Nuclear Engineering,

Massachusetts Institute of Technology to visit and review the LACBWR facility. This meeting occurred on March 22, 1982, at which time Dairyland personnel conducted a tour of LACBWR and answered questions posed by Messrs. Levine and Rasmussen. Based on this meeting and additional interaction between NUS and Dairyland staff, Messrs. Levine and Rasmussen felt that the probability of a core melt with a radionuclide release to the environment, resulting only from internal plant failures at LACBWR, is likely to be quite low. This belief is based on the extensive system redundancies in the LACBWR design and the relatively modest amount of energy generated by the core, both of which allow a great deal of flexibility in mitigating the effects of accidents.

The development of quantitative estimates of core melt and containment failure probabilities will, however, require some further analysis. Because of the extremely low consequences that could result from even the most serious potential accident at LACBWR, the level of detailed technical analysis required in probabilistic risk assessments performed for larger reactors would not be appropriate for LACBWR; rather a simpler methodology, relying more on judgment and experience, would be more appropriate. If this analysis proceeds, the event trees that will be constructed will illustrate the strength of safety system designs in the LACBWR plant.

2.0 TECHNICAL CONSIDERATIONS

Although the integrated plant response to a severe accident cannot be determined without a more complete analysis, a preliminary review of the plant has revealed that significant system redundancy exists to perform various post-accident mitigating functions. The ability to perform these functions is further enhanced because the very low power level of the

core and the correspondingly small decay heat allows both time and flexibility to use the many alternate methods provided in the plant to mitigate accident effects. Post-accident functions and the available means of performing them have been catalogued in Tables 1 and 2 for various initiating events, based on a brief review of the LACBWR facility. A number of unusual characteristics of the LACBWR design which lead to increased flexibility of plant response in some situations can be seen in Tables 1 and 2. For example, for LOCAs, (i) the two train high pressure core spray system has adequate flow capacity to provide core cooling over the entire range of pipe break sizes, rather than just for smaller breaks as in larger reactors, (ii) the alternate core spray system, a low pressure ECC system, has redundant trains, each of whose pumps is powered by a dedicated diesel; this system, combined with the manual depressurization system, can also provide core cooling over the entire pipe break spectrum, (iii) the low pressure core spray system is a gravity fed system supplied by water from the overhead storage tank; this system, when combined with the manual depressurization system, can provide core cooling, for some period, over the entire break spectrum and (iv) portable emergency service water supply system pumps can also be used with the manual depressurization system to provide core cooling over the entire break spectrum. Thus, it can be seen that significantly more system capability in providing emergency core cooling exists at LACBWR than at most other reactors.

Most modes of containment failure at LACBWR are expected to be less likely than at large power reactors because of the following considerations (i) the ratio $(P_{CD} V_F/P)$[1], which is a

[1] Where P_{CD} = Containment design pressure, V_F = containment free volume and P = reactor power level.

measure of the energy absorbtive capacity of the containment, is comparable to or greater than even large dry PWR containments, (ii) the ratio of containment basemat thickness to core power is much greater for LACBWR than large power reactors, (iii) the amount of energy potentially released to the containment after an accident is so small that heat transfer methods like radiation from the steel shell or convection through the component cooling water piping can transfer decay heat to the environment and (iv) the containment is designed to be flooded to the level of the core mid-plane. Thus, it is expected that containment failures due to basemat melt-through or overpressure due to non-consensables generation or hydrogen burn are unlikely. Recent work in the area of vessel steam explosion indicates that such a failure mode would also be unlikely. However, because the containment is ventilated during normal operation, it is expected that the possibility of a containment that cannot be isolated at the time of an accident would be more likely than for other reactors; even so, this failure mode would probably not be greater than 10^{-2} per demand.

The preceding discussion indicates the basis for the feeling that severe accidents due to internal plant failures are relatively low likelihood events. The question is then raised about whether common mode failures due to earthquakes and other "external" events contribute significantly to the likelihood of severe accidents. However, the potential for certain external events to provide a core melt probability of greater than about 10^{-4} per reactor year can be immediately dismissed, principally on the basis of the frequencies of these events. Their contribution to core melt would be even less when the potential for plant systems to mitigate the

event is considered. The external events which have been considered and concluded to be unimportant contributors to a high core melt probability at LaCrosse are:

(i) External floods
(ii) Tornadoes
(iii) Aircraft related hazards
(iv) Truck transportation related hazards

It is also likely that barge and rail transportation hazards are negliglble contributors to a high core melt probability. Seismic occurrences and internal fires require further consideration to determine their potential contribution to the core melt probability.

3.0 TECHNICAL APPROACH TO A SIMPLIFIED PRA

Even though the public consequences associated with potential accidents at LACBWR are predicted to be small, it is important to know the likelihood of severe events such as core melt and containment failure, given core melt. More exactly, for LACBWR, it is important only to ascertain whether the probabilities of these severe events are likely to be fairly small. To acquire this information, a simplified probabilistic analysis could be conducted. This analysis could be done on a limited basis only to demonstrate that the core melt probability is not likely to exceed some value and to make an evaluation of the likely containment failure modes and probabilities. Levine and Rasmussen suggest that the demonstration of a core melt probability in the range of 10^{-3} to 10^{-4} per reactor year would be appropriate for LACBWR. The simplified analysis would use plant-specific event trees to identify potential core melt accident sequences. To determine the containment failure modes associated with these core melt

sequences, an event tree modeling containment failure would also be developed. The failure probabilities of accident mitigating systems would not be determined by fault trees, but rather by a combination of simpler techniques: expert judgment, the construction of Boolean expressions which model system failure modes or the use of plant operating experience data.

The simplified approach suggested was developed from the insights and techniques used in risk assessments previously performed by the NRC and others; specifically, the Reactor Safety Study (WASH-1400), the Reactor Safety Study Methodology Applications Program (RSSMAP) and the Integrated Reliability Evaluation Program (IREP). Expert judgment would be used by this approach both as the primary means for performing various tasks or as verification of the results of analyses. The expert judgment that will be used is that of recognized authorities in the risk assessment field and other areas as needed for the study. The expert judgment used for the probabilistic analysis will be provided by S. Levine and N. Rasmussen. It is also believed that expert judgment would be particularly important to evaluate the effects of potential seismic events on plant components. This judgment will be provided by Dr. Robert Kennedy, President, Structural Mechanics Associates and Dr. Spencer H. Bush, Senior Staff Consultant, Battelle Northwest Laboratories.

4.0 CONCLUSION

The judgment of Levine and Rasmussen is that the probability of a core melt and containment failure at LACBWR is likely to be relatively low. These probabilities cannot be quantified without further analysis. However, if it is desired only to ascertain whether the probability of a core melt is not greater than some value, say 10^{-3} to 10^{-4} per reactor year,

then a simplified probabilistic analysis can be used. Containment failure can also be assessed in a similar manner. Such a simplified probabilistic analysis would use insights and techniques from previously documented risk assessments; it would also rely on expert judgment in the area of risk and reliability and in other technical areas, as indicated. As stated earlier, this less rigorous approach is believed to be appropriate for the LACBWR plant in view of the fact that there would be negligible consequences if a severe accident were to occur.

TABLE 1

PRELIMINARY SUCCESS CRITERIA FOR SYSTEMS PERFORMING POST-LOCA FUNCTIONS

LOCA SIZE	Subcriticality	Emergency Core Cooling (ECI)	Containment OP Protection (< 126 Hours)	Emergency Core Cooling (Long Term)	Containment OP Protection (> 126 hours)	Post Accident Radioactivity Removal*
STEAM BREAK	RPS or BIS	1 of 2 HPCS Trains or 1 of 2 ACS trains or LPCS and MDS or MDS and 3 of 4 ESWSS	Containment Design	1 of 2 HPCS and OHST Makeup or 1 of 2 ACS trains or MDS and LPCS and OHST Makeup or 3 of 4 ESWSS	CSS and OHST Makeup or Component Cooling Water System or External Water on Containment	CSS and OHST Makeup
SMALL LIQUID BREAK	RPS or BIS	1 of 2 HPCS Trains or MDS and 1 of 2 ACS Trains or SC and 1 of 2 ACS Trains or MDS and LPCS or MDS and 3 of 4 ESWSS	Containment Design	1 of 2 HPCS and OHST Makeup or 1 of 2 ACS trains or LPCS and MDS and OHST Makeup or 3 of 4 ESWSS	CSS and OHST Makeup or Component Cooling Water System or External Water on Containment	CSS and OHST Makeup
LARGE LIQUID BREAK	RPS or BIS	1 of 2 HPCS Trains or LPCS or 1 of 2 ACS trains or 3 of 4 ESWSS	Containment Design	1 of 2 HPCS and OHST Makeup or 1 of 2 ACS trains or LPCS and OHST Makeup or 3 of 4 ESWSS	CSS and OHST Makeup or Component Cooling Water System or External Water on Containment	CSS and OHST Makeup

Key:

ACS:	Alternate Core Spray System
BIS:	Boron Injection System (uses HPCS pumps)
CSS:	Containment Spray System (manual)
ESWSS:	Emergency Severe Water Supply System
HPCS:	High Pressure Core Spray System (harsh environment considered in design)
LPCS:	Low Pressure Core Spray System
MDS:	Manual Depressurization System
OHST:	Overhead Storage Tank
RPS:	Reactor Protection System

* Operation of the CSS is not necessary to meet 10CFR100 dose requirements.

TABLE 2

PRELIMINARY SUCCESS CRITERIA FOR SYSTEMS PERFORMING POST TRANSIENT FUNCTIONS

Transient	Subcriticality	RCS Overpressure Protection	RCS Integrity	Emergency Core Cooling (Short)	Residual Heat Removal
General (T_1)	RPS or Recirculation Pump Trip and Boron Injection	PCS or SC and/or Safety Relief Valves or MDS	Closure of all Open Relief Valves	PCS or SC and SC Makeup or 1 of 2 HPCS trains or MDS and 1 of 2 ACS trains or 3 of 4 ESWSS trains	PCS or SC and SC Makeup or Decay Heat Removal System and component cooling water systems
Loss of Power Conversion System (T_2)	RPS or Recirculation Pump Trip and Boron Injection	SC and/or Safety/ Relief Valves or MDS	Closure of all Open Relief Valves	SC and SC Makeup or 1 of 2 HPCS trains or MDS and 1 of 2 ACS trains or 3 of 4 ESWSS trains	SC and SC Makeup or Decay Heat Removal System and component cooling water systems
Loss of Offsite Power (T_3)	RPS or Recirculation Pump Trip and Boron Injection	SC and/or Safety/ Relief Valves or MDS	Closure of all Open Relief Valves	SC and SC Makeup or 1 of 2 HPCS trains and Emergency Diesel Generator or MDS and 1 of 2 ACS trains or 3 of 4 ESWSS trains	SC and SC Makeup or Decay Heat Removal System, component cooling water systems and Emergency Diesel Generator

Key:
- ACS: Alternate Core Spray System
- HPCS: High Pressure Core Spray System
- MDS: Manual Depressurization System
- PCS: Power Conversion System (Main Steam and Feedwater)
- SC: Shutdown Condenser

BACKGROUND

- o BECAUSE OF ITS LOW POWER, THE CONSEQUENCES OF A SEVERE ACCIDENT AT LACBWR ARE PREDICTED TO BE EXCEEDINGLY SMALL
- o HOWEVER, IT IS IMPORTANT TO KNOW WHETHER THE LIKELIHOOD OF THE UNDESIREABLE EVENTS (CORE MELT AND CONTAINMENT FAILURE) ARE OF FAIRLY LOW PROBABILITY
- o BECAUSE OF THE LOW POWER AND THE EXTENSIVE SYSTEM REDUNDANCIES IN LACBWR, IT APPEARS THAT THE LIKELIHOOD OF THESE EVENTS (DUE TO INTERNAL PLANT FAILURES) WOULD BE QUITE LOW

EMERGENCY CORE COOLING

MORE ECCS CAPABILITY THAN MOST LARGE PLANTS

- o FOUR SYSTEMS CAN COOL CORE OVER ENTIRE SPECTRUM OF PIPE BREAK SIZES:
  - o 1 OF 2 TRAINS OF HPCS
  - o 1 OF 2 TRAINS OF LOW PRESSURE ACS AND MDS
  - o GRAVITY FED LPCS AND MDS
  - o 3 OF 4 PORTABLE ESWSS AND MDS
- o MANUAL DEPRESSURIZATION NOT REQUIRED FOR LARGE BREAKS AND CERTAIN SMALL BREAKS
- o SOME COMMON PIPING TO CONSIDER

CONTAINMENT CHARACTERISTICS

LOW POWER LEVEL MAKES CONTAINMENT FAILURE UNLIKELY (SE, BCM)

- ENERGY ABSORPTIVE CAPABILITY ($P_{CD} x V_F/P$) COMPARABLE TO LARGE DRY PWR CONTAINMENTS
- DECAY HEAT SO LOW THAT IT CAN BE REMOVED THROUGH STEEL SHELL OR COMPONENT COOLING WATER SYSTEM
- CONTAINMENT BASEMAT THICKNESS TO CORE POWER RATIO GREATER FOR LACBWR THAN FOR LARGE REACTORS
- CONTAINMENT CAN BE FLOODED TO CORE MIDPLANE TO PROVIDE LARGE HEAT SINK
- GRAVITY FED CONTAINMENT SPRAY SYSTEM AVAILABLE FOR POTENTIAL PRESSURE REDUCTION AND RADIOACTIVITY REMOVAL
- SINCE CONTAINMENT IS VENTILATED IN NORMAL OPERATION, CLOSURE SYSTEM PROBABILITY OF FAILURE HAS TO BE EXAMINED

REACTOR SHUTDOWN

- WELL DESIGNED SCRAM SYSTEM
- MANUAL BORON INJECTION SYSTEM CAN QUICKLY SHUTDOWN REACTOR
- RECIRCULATION PUMP TRIP LIMITS POWER

EXTERNAL EVENTS

- o THESE EVENTS MAY CAUSE COMMON MODE FAILURES OF SYSTEMS
- o USING FREQUENCY OF 10^{-4} PER RY SCREENING ALLOWS MANY EXTERNAL EVENTS TO BE ELIMINATED FROM FURTHER CONSIDERATION: EXTERNAL FLOODS, TORNADOES, AIRCRAFT CRASHES, TRUCK ACCIDENTS, AND PROBABLY BARGE AND RAIL ACCIDENTS
- o SEISMIC EVENTS AND INTERNAL FIRES ARE MORE DIFFICULT TO ESTIMATE AND WOULD REQUIRE FURTHER ANALYSIS

SUMMARY

- FROM A BRIEF REVIEW OF THE LACBWR PLANT, IT IS OPINION OF OF LEVINE AND RASMUSSEN:
  - THAT THE CORE MELT PROBABILITY AT LACBWR, DUE TO INTERNAL INITIATING EVENTS, IS NOT LIKELY TO BE GREATER THAN 10^{-3} PER RY
  - THAT MANY EXTERNAL EVENTS ARE LIKELY TO CONTRIBUTE A FREQUENCY OF LESS THAN 10^{-4} PER RY TO THE CORE MELT PROBABILITY. MORE ANALYSES WOULD BE NEEDED TO CONSIDER THE EFFECTS OF EARTHQUAKES AND FIRES
  - THAT MANY FAILURE MODES OF THE LACBWR CONTAINMENT ARE LESS LIKELY THAN FOR OTHER REACTORS
- IF ADDITIONAL BASES FOR THESE CONCLUSIONS IS NECESSARY, A MORE THOROUGH ANALYSIS CAN BE CONDUCTED BY A SIMPLIFIED PROBABILISTIC ANALYSIS

APPROACH TO A SIMPLIFIED PROBABILISTIC ANALYSIS

- o IF NRC FEELS IT TO BE NECESSARY, A SIMPLIFIED PROBABILISTIC ANALYSIS CAN BE PERFORMED
  - o IT WOULD PRODUCE A ROUGH ESTIMATE OF CORE MELT AND CONTAINMENT FAILURE PROBABILITIES; LESS PRECISE APPROACH THAN TYPICAL PROBABILITY ANALYSES WOULD BE ADEQUATE BECAUSE OF LOW CONSEQUENCES
  - o IT WOULD BE CONDUCTED ONLY TO THE EXTENT NECESSARY TO ASCERTAIN WHETHER THESE PROBABILITIES ARE LIKELY TO BE FAIRLY LOW
  - o CORE MELT PROBABILITY IN THE RANGE 10^{-3} TO 10^{-4} PER REACTOR YEAR IS SUGGESTED AS AN ACCEPTABLE RANGE FOR LACBWR

MAJOR CHARACTERISTICS OF ANALYSIS

- o INTERNAL AND EXTERNAL INITIATING EVENTS WOULD BE CONSIDERED
- o METHODOLOGY ADAPTED FROM PREVIOUSLY CONDUCTED RISK ASSESSMENTS I.E. WASH-1400, RSSMAP, IREP
- o EXPERT JUDGMENT WOULD BE USED EXTENSIVELY

INTERNAL INITIATING EVENTS

- o PLANT SPECIFIC EVENT TREES WOULD BE CONSTRUCTED TO COVER SPECTRUM OF LOCAS AND TRANSIENTS
- o SYSTEM FAULT TREES NOT USED; SIMPLER TECHNIQUES, INCLUDING EXPERT JUDGMENT, WOULD BE USED
- o GENERIC DATA, MODIFIED BY LER AND LACBWR EXPERIENCES AND EXPERT JUDGMENT, WOULD BE USED
- o UNCERTAINTIES WOULD NOT BE ESTIMATED
- o ACCIDENT SEQUENCES WITH PROBABILITIES LESS THAN 10^{-4} SCREENED FROM FURTHER ANALYSIS

CONTAINMENT FAILURE

- SIMPLE ANALYSES AND EXPERT JUDGMENT WOULD BE USED TO IDENTIFY LIKELY CONTAINMENT FAILURE MODES
- PROBABILITIES OF FAILURE MODES WOULD BE ESTIMATED

SEISMIC OCCURRENCE

- o SIGNIFICANT DETERMINISTIC ANALYSIS PERFORMED ON SYSTEM/ STRUCTURES STRESS RESPONSE
- o SITE SPECIFIC SEISMICITY CURVE AVAILABLE
- o COMPONENT FRAGILITIES WOULD BE DEVELOPED FROM EXISTING ANALYSES AND EXPERT JUDGMENT
- o COMBINE SEISMICITY AND FRAGILITY INFORMATION FOR "MINIMUM PATH SET" TO BOUND SEISMIC CONTRIBUTION TO CORE MELT PROBABILITY

FIRES

- o IDENTIFY AREAS WHERE POTENTIAL COMMON MODE FAILURES COULD OCCUR
- o USE GENERIC DATA FOR FIRE FREQUENCIES IN AREAS OF CONCERN
- o IDENTIFY SYSTEMS AFFECTED BY FIRE
- o MAKE CONSERVATIVE ASSUMPTIONS REGARDING SYSTEM DAMAGE TO DETERMINE IF SPECIFIC SEQUENCES ARE SIGNIFICANT CONTRIBUTORS TO CORE MELT PROBABILITY

RESULTS

- POINT ESTIMATES OF CORE MELT AND CONTAINMENT FAILURE PROBABILITIES
- SIGNIFICANT ACCIDENT SEQUENCES AND CONTAINMENT FAILURE MODES DEFINED