DAIRYLAND POWER COOPERATIVE • P.O. BOX 817 • 2615 EAST AV SOUTH • LA CROSSE, WISCONSIN 54601
(608) 788-4000

August 13, 1982

In response, please
reply to LAC-8497

DOCKET NO. 50-409

Public Document Room
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: DAIRYLAND POWER COOPERATIVE (DPC)
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
INACCURACIES REPORTED IN NUREG/CR-2497

REFERENCES: (1) NUREG/CR-2497, Volume 1, ORNL/NSIC-182/V1, Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report, Appendices A, C, D, and E
(2) NUREG/CR-2497, Volume 2, ORNL/NSIC-182/V2, Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report, Appendix B

Gentlemen:

The publishing of NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report" was brought to our attention. Upon request, we obtained a copy from the Nuclear Regulatory Commission.

We are disappointed at the irresponsibility of the Nuclear Regulatory Commission, their contractor, Oak Ridge National Laboratory, and subcontractor, Science Applications, Inc. in publishing this report with the substantial inaccuracies it contains.

The major errors involved in the analyses of incidents at the La Crosse Boiling Water Reactor are significant. One simple telephone call or letter to Dairyland Power Cooperative could have eliminated the notable misassumptions made with regards to our plant. We do not know if the analyses of our plant's incidents are indicative of the quality of the entire report, but we definitely question the remainder of this study. When dealing with as important and controversial a subject as the probability of core damage, an effort must be made to utilize correct facts, especially since the accuracy of the methodology and equipment failure rates is not absolute.

Three loss of offsite power type events which occurred at LACBWR were analyzed in detail. The descriptions of the events were fairly accurate. The conversion of each occurrence into the standard BWR Loss of Offsite Power event tree demonstrated a total lack of understanding of LACBWR systems and the absence of any effort to gain knowledge of our plant in spite of the report summary which stated:

> *All LERs selected for detailed review were subjected to an in-depth evaluation, which included:*
>
> *(1) a review of the accident sequence (if there was one) as described in the LER,*
> *(2) a review of the design of systems in the reactor plant reporting the LER to determine the impact of the failure on the operation of these systems, and*
> *(3) a review of the plant accident analyses to determine the extent to which affected systems would be required to function for different off-normal and accident conditions.*

Section 3 of the reports states:

> *Standardized event trees were used for the sequence of interest trees when the initiating event was a loss of feedwater, loss of offsite power, small LOCA, or steam line break. To permit this, mitigating systems were functionally represented on these trees. The success criteria for those functions vary from plant to plant, and certain functions do not exist at every plant (for example, runback following a load rejection). The effect of the LER event on the specific safety-related systems of the plant at which the failure occurred was carefully considered when the failed or degraded states of the event tree functions were determined. The standardized event trees used in the study are described in Appendix A. (Emphasis added).*

There is absolutely no evidence of careful consideration of the effect of the events on the specific safety-related systems at LACBWR.

The three analyzed events at LACBWR were the 1/20/71 Loss of Offsite Power (Accession No. 61043), the 3/24/71 Loss of Offsite Power (63129), and the 8/17/72 Loss of Load (75074). The 3/24/71 incident was listed in Table 4.2, "Precursors Listed by Significance Category", as being the ninth most significant event.

The event trees used for the sequence of interest and analyses for the three incidents are attached. The assumptions were made that, (1) the shutdown condenser and condensate pumps were the equivalent of RCIC and HPCI, and (2) operation of the shutdown condenser is dependent on the condensate pumps. Both of these assumptions are erroneous.

The combination of the LACBWR shutdown condenser and the High Pressure Core Spray (HPCS) System could be used as the equivalent of RCIC and HPCI. To leave out HPCS is neglecting the primary source of emergency cooling water at LACBWR and is not understandable.

The dependence of the shutdown condenser on the condensate pumps is totally nonexistent except possibly in the minds of the authors of this report. The condensate pumps are associated with the main condenser. Since they are not supplied by emergency power, they are always unavailable during a loss of offsite power. The descriptions of two of the incidents, including the March 1971 event, specifically mention that the shutdown condenser and the emergency core spray pump were used. Yet the event tree for the March, 1971 incident has the shutdown condenser as failed. The shutdown condenser has never failed to operate when needed or during testing. The failure rate for the shutdown condenser and the High Pressure Core Spray System should be re-evaluated for all three events, rather than using the RCIC/HPCI failure rate of 3.9×10^{-3} or 1.0, as used in the March 1971 event analysis. There are significant differences between the shutdown condenser and HPCS at LACBWR and GE's HPCI and RCIC.

An additional error contained in the event is the failure rate used for the reactor being made subcritical by the SBLCS or rods being manualy driven into the core. Table C.1, "Initiating Event Frequencey and Function Failure Probability Estimates" states that for a BWR loss of offsite power, the failure to make reactor subcritical by means other than a scram is taken as 5×10^{-3} per demand. In the event trees, however, a failure probability of 0.1 was used. LACBWR has a Boron Inject Sytem, rather than a SBLCS, but the failure rate should not be affected to that extent by the difference.

The event trees do not allow for the use of a low pressure core spray system if the emergency diesel generator fails to start. At LACBWR, the Alternate Core Spray System does not require offsite or onsite AC power to function. River water is supplied to the reactor from one of two diesel-driven pumps. Therefore, it is available for use if the diesel generators do not start. Also, the Low Pressure Core Spray Subsystem can be used at low reactor pressure. It consists of a gravity feed system from the Overhead Storage Tank in the Containment Building. The Low Pressure Core Spray Valve fails open on loss of power, so it is usable once the reactor is depressurized, even if the diesel generators do not start. There are additional differences between LACBWR's systems used for low pressure emergency cooling (Manual Depressurization System, Alternate Core Spray System and Low Pressure Core Spray Subsystem) and the Automatic Depressurization System, Low Pressure Coolant Injection, and Low Pressure Core Spray utilized in the study. These differences may affect the failure rate which was dominated by ADS failures. There has never been a simultaneous failure of both MDS Valves at LACBWR.

The compilation of errors made in the analyses of the loss of offsite power events at LACBWR totally invalidates those analyses. The extent of the false assumptions made is incredible considering the vast quantities of docketed material which has been submitted during the past twenty years and the care the authors of the study used in considering the specific systems at the plants being studied.

Dairyland Power Cooperative requests that NUREG/CR-2497 be retracted with explanation and completely reviewed and corrected prior to reissue. The subject of core damage should not be treated lightly or carelessly, nor should a plant's reputation be damaged by a fallacious study.

If there are any questions concerning this letter, please contact us.

Yours truly,

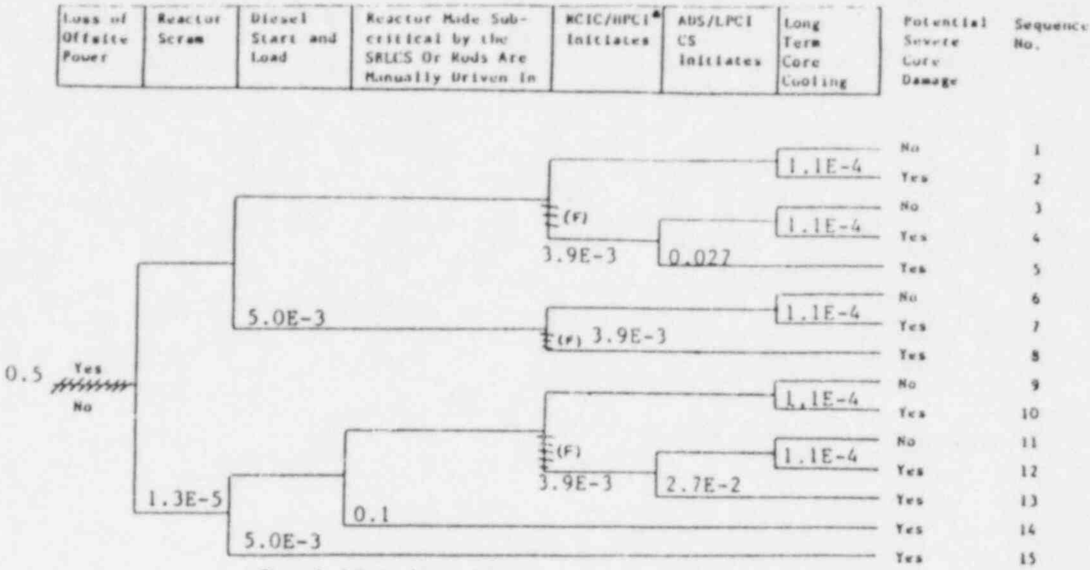
DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder, General Manager

FL:LSG:eme

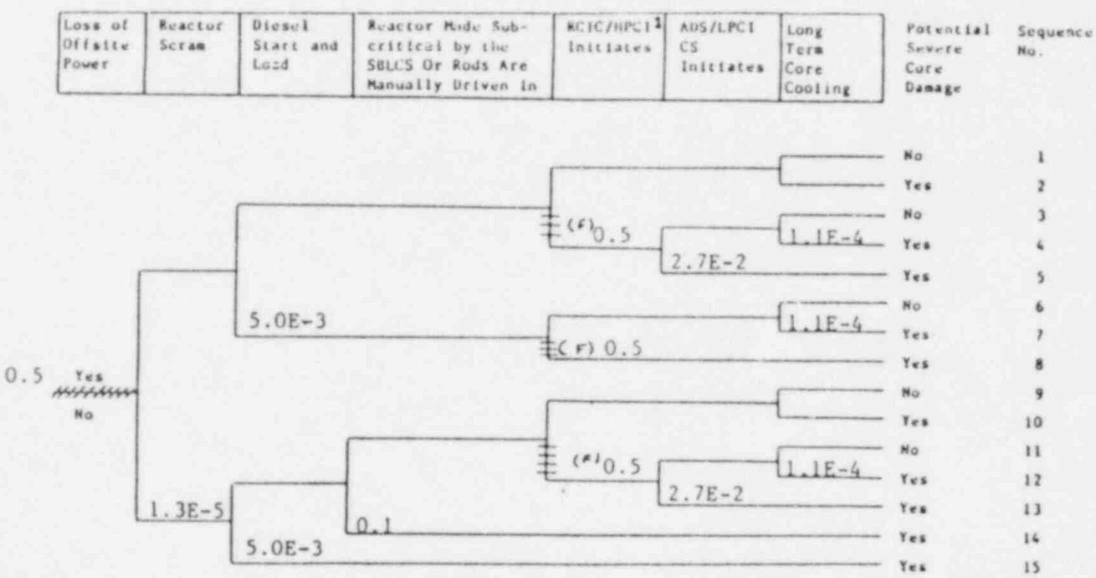
Enclosures (3)



P = 1.2E-4 (S = 39)

NSIC 61043 – Sequence of Interest for Plant Electrical Power Failure at La Crosse

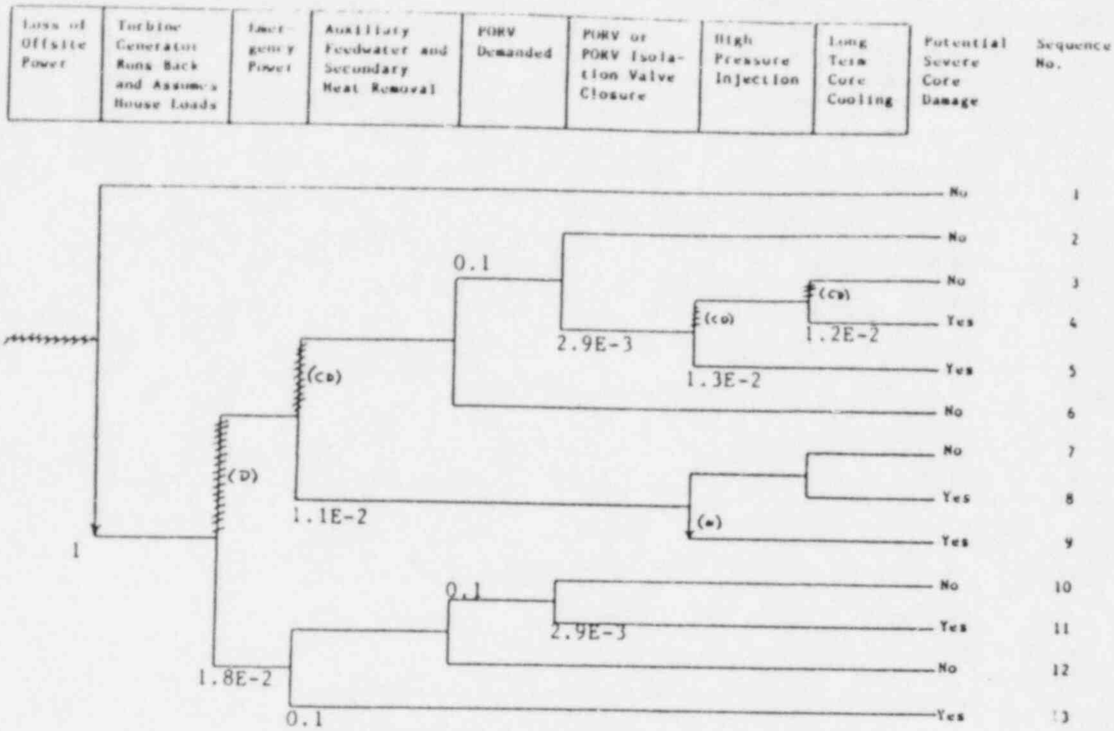
*La Crosse utilizes shutdown condenser and condensate pumps instead of RCIC and HPCI.



P = 8.8E-3 (SC = 21)

NSIC 61434 – Sequence of Interest for Loss of Offsite Power and Failure of an Emergency Condenser Valve to Open at Humboldt Bay

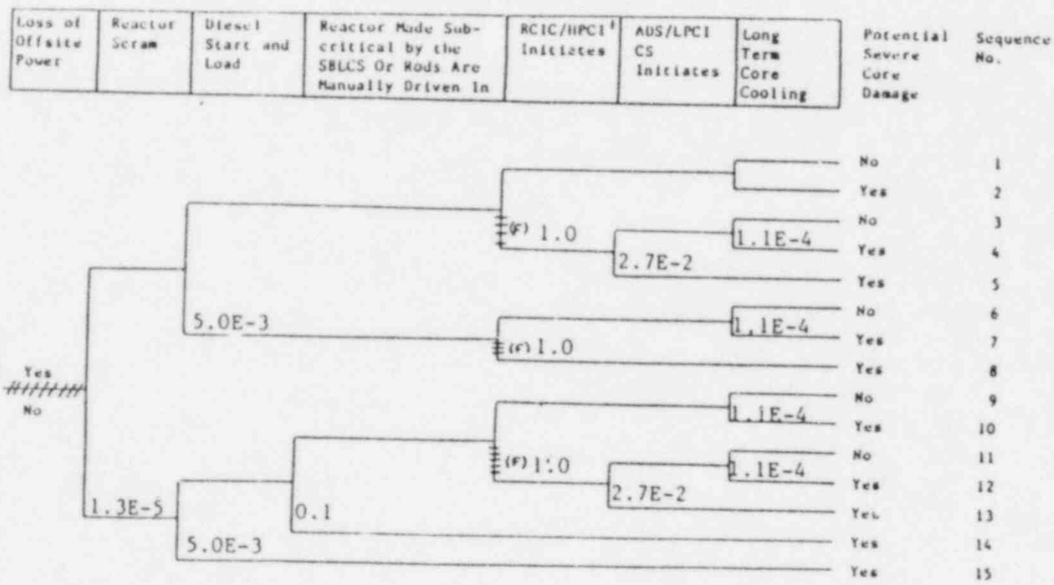
[1]Humboldt Bay utilized an emergency condenser and CRD hydraulic pumps/safety valves for decay heat removal



P = 6.4E-3 (SC = 22)

NSIC 61565 - Sequence of Interest for Loss of Offsite Power and Failure of a Diesel Generator to Load at Palisades

*Use of HPI following AFW failures not included in mitigation procedures.

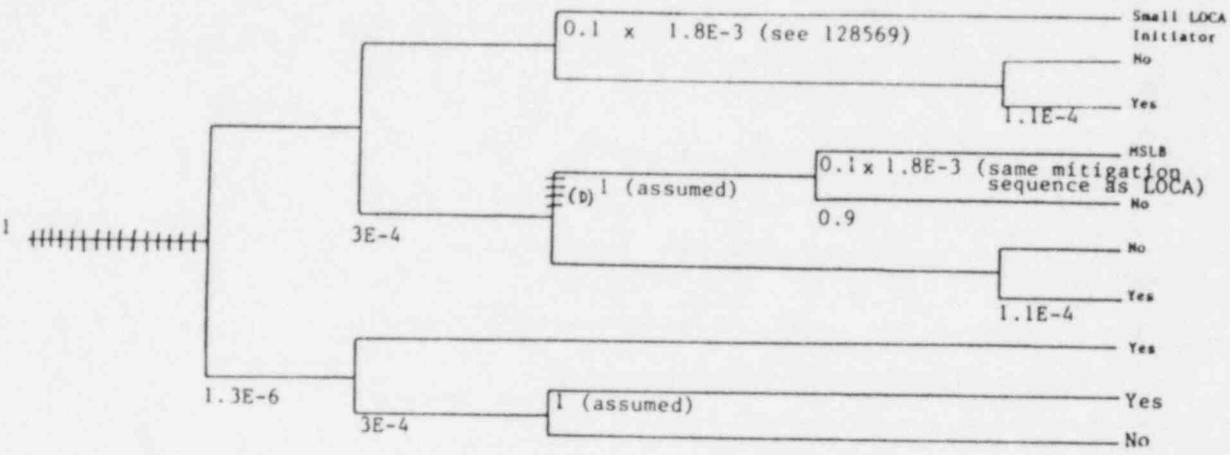


P = 1.8E-2 (SC = 18)

NSIC 63129 - Sequence of Interest for A Scram Caused by Electrical Load Rejection at La Crosse

[1] La Crosse utilizes shutdown condenser and condensate pumps instead of RCIC and HPCI.

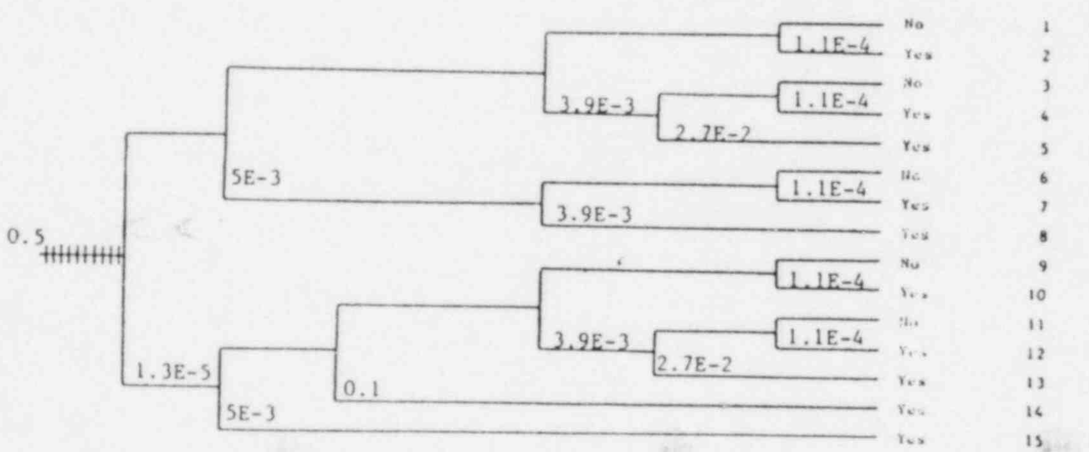
Excessive Coolant Inventory	Reactor Scram	Reactor Vessel or Turbine Isolate	Reactor Coolant Overflows Into Steam Lines, Is Discharged Through Relief Valves, Which Stick Open	Steam Line Break Due to Turbine Missiles, etc.	Long Term Core Cooling Success	Potential Severe Core Damage



P = 2.8E-4 (SC = 36)

NSIC 74242 – Sequence of Interest for High Coolant Level at Nine Mile Point

Loss of Offsite Power	Reactor Scram	Diesel Start and Load	Reactor Made Sub-critical by the SBLCS Or Rods Are Manually Driven In	RCIC/HPCI Initiates	ADS/LPCI CS Initiates	Long Term Core Cooling	Potential Severe Core Damage	Sequence No.



P = 1.2E-4 (SC = 39)

NSIC 75074 – Sequence of Interest for Loss of Load and Iodine Release at LaCrosse

DAIRYLAND POWER COOPERATIVE • P.O. BOX 817 • 2615 EAST AV SOUTH • LA CROSSE, WISCONSIN 54601
(608) 788-4000

August 13, 1982

In response, please
reply to LAC-8497

DOCKET NO. 50-409

Public Document Room
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: DAIRYLAND POWER COOPERATIVE (DPC)
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
INACCURACIES REPORTED IN NUREG/CR-2497

REFERENCES: (1) NUREG/CR-2497, Volume 1, ORNL/NSIC-182/V1, Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report, Appendices A, C, D, and E
(2) NUREG/CR-2497, Volume 2, ORNL/NSIC-182/V2, Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report, Appendix B

Gentlemen:

The publishing of NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report" was brought to our attention. Upon request, we obtained a copy from the Nuclear Regulatory Commission.

We are disappointed at the irresponsibility of the Nuclear Regulatory Commission, their contractor, Oak Ridge National Laboratory, and subcontractor, Science Applications, Inc. in publishing this report with the substantial inaccuracies it contains.

The major errors involved in the analyses of incidents at the La Crosse Boiling Water Reactor are significant. One simple telephone call or letter to Dairyland Power Cooperative could have eliminated the notable misassumptions made with regards to our plant. We do not know if the analyses of our plant's incidents are indicative of the quality of the entire report, but we definitely question the remainder of this study. When dealing with as important and controversial a subject as the probability of core damage, an effort must be made to utilize correct facts, especially since the accuracy of the methodology and equipment failure rates is not absolute.

Three loss of offsite power type events which occurred at LACBWR were analyzed in detail. The descriptions of the events were fairly accurate. The conversion of each occurrence into the standard BWR Loss of Offsite Power event tree demonstrated a total lack of understanding of LACBWR systems and the absence of any effort to gain knowledge of our plant in spite of the report summary which stated:

All LERs selected for detailed review were subjected to an in-depth evaluation, which included:

(1) a review of the accident sequence (if there was one) as described in the LER,
(2) a review of the design of systems in the reactor plant reporting the LER to determine the impact of the failure on the operation of these systems, and
(3) a review of the plant accident analyses to determine the extent to which affected systems would be required to function for different off-normal and accident conditions.

Section 3 of the reports states:

Standardized event trees were used for the sequence of interest trees when the initiating event was a loss of feedwater, loss of offsite power, small LOCA, or steam line break. To permit this, mitigating systems were functionally represented on these trees. The success criteria for those functions vary from plant to plant, and certain functions do not exist at every plant (for example, runback following a load rejection). The effect of the LER event on the specific safety-related systems of the plant at which the failure occurred was carefully considered when the failed or degraded states of the event tree functions were determined. The standardized event trees used in the study are described in Appendix A. (Emphasis added).

There is absolutely no evidence of careful consideration of the effect of the events on the specific safety-related systems at LACBWR.

The three analyzed events at LACBWR were the 1/20/71 Loss of Offsite Power (Accession No. 61043), the 3/24/71 Loss of Offsite Power (63129), and the 8/17/72 Loss of Load (75074). The 3/24/71 incident was listed in Table 4.2, "Precursors Listed by Significance Category", as being the ninth most significant event.

The event trees used for the sequence of interest and analyses for the three incidents are attached. The assumptions were made that, (1) the shutdown condenser and condensate pumps were the equivalent of RCIC and HPCI, and (2) operation of the shutdown condenser is dependent on the condensate pumps. Both of these assumptions are erroneous.

The combination of the LACBWR shutdown condenser and the High Pressure Core Spray (HPCS) System could be used as the equivalent of RCIC and HPCI. To leave out HPCS is neglecting the primary source of emergency cooling water at LACBWR and is not understandable.

The dependence of the shutdown condenser on the condensate pumps is totally nonexistent except possibly in the minds of the authors of this report. The condensate pumps are associated with the main condenser. Since they are not supplied by emergency power, they are always unavailable during a loss of offsite power. The descriptions of two of the incidents, including the March 1971 event, specifically mention that the shutdown condenser and the emergency core spray pump were used. Yet the event tree for the March, 1971 incident has the shutdown condenser as failed. The shutdown condenser has never failed to operate when needed or during testing. The failure rate for the shutdown condenser and the High Pressure Core Spray System should be re-evaluated for all three events, rather than using the RCIC/HPCI failure rate of 3.9×10^{-3} or 1.0, as used in the March 1971 event analysis. There are significant differences between the shutdown condenser and HPCS at LACBWR and GE's HPCI and RCIC.

An additional error contained in the event is the failure rate used for the reactor being made subcritical by the SBLCS or rods being manualy driven into the core. Table C.1, "Initiating Event Frequencey and Function Failure Probability Estimates" states that for a BWR loss of offsite power, the failure to make reactor subcritical by means other than a scram is taken as 5×10^{-3} per demand. In the event trees, however, a failure probability of 0.1 was used. LACBWR has a Boron Inject Sytem, rather than a SBLCS, but the failure rate should not be affected to that extent by the difference.

The event trees do not allow for the use of a low pressure core spray system if the emergency diesel generator fails to start. At LACBWR, the Alternate Core Spray System does not require offsite or onsite AC power to function. River water is supplied to the reactor from one of two diesel-driven pumps. Therefore, it is available for use if the diesel generators do not start. Also, the Low Pressure Core Spray Subsystem can be used at low reactor pressure. It consists of a gravity feed system from the Overhead Storage Tank in the Containment Building. The Low Pressure Core Spray Valve fails open on loss of power, so it is usable once the reactor is depressurized, even if the diesel generators do not start. There are additional differences between LACBWR's systems used for low pressure emergency cooling (Manual Depressurization System, Alternate Core Spray System and Low Pressure Core Spray Subsystem) and the Automatic Depressurization System, Low Pressure Coolant Injection, and Low Pressure Core Spray utilized in the study. These differences may affect the failure rate which was dominated by ADS failures. There has never been a simultaneous failure of both MDS Valves at LACBWR.

The compilation of errors made in the analyses of the loss of offsite power events at LACBWR totally invalidates those analyses. The extent of the false assumptions made is incredible considering the vast quantities of docketed material which has been submitted during the past twenty years and the care the authors of the study used in considering the specific systems at the plants being studied.

Dairyland Power Cooperative requests that NUREG/CR-2497 be retracted with explanation and completely reviewed and corrected prior to reissue. The subject of core damage should not be treated lightly or carelessly, nor should a plant's reputation be damaged by a fallacious study.

If there are any questions concerning this letter, please contact us.

Yours truly,

DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder, General Manager

FL:LSG:eme

Enclosures (3)

cc:

Mr. R. M. Bernero, Director
Division of Risk Analysis
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Gary S. Burdick
Division of Risk Analysis
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

The Commissioners
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Dick Dudley "TO BE OPENED BY ADDRESSEE ONLY"
Mail Stop 314
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Public Document Room
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Wm. B. Cottrell, Director
Nuclear Operations Analysis Center
P. O. Box Y
Oak Ridge, TN 37839

Director
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

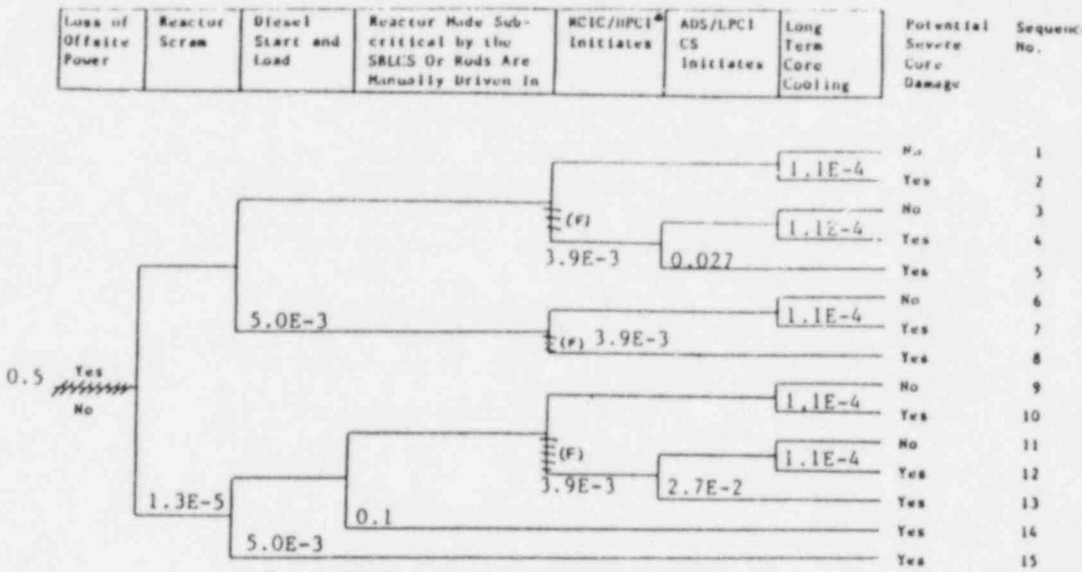
Mr. J. W. Minarick
Science Applications, Inc.
10373 Roselle St.
San Diego, California 92121

Mr. C. A. Kukielka
Science Applications, Inc.
10373 Roselle St.
San Diego, California 92121

President
Science Applications, Inc.
10373 Roselle St.
San Diego, California 92121

bcc:

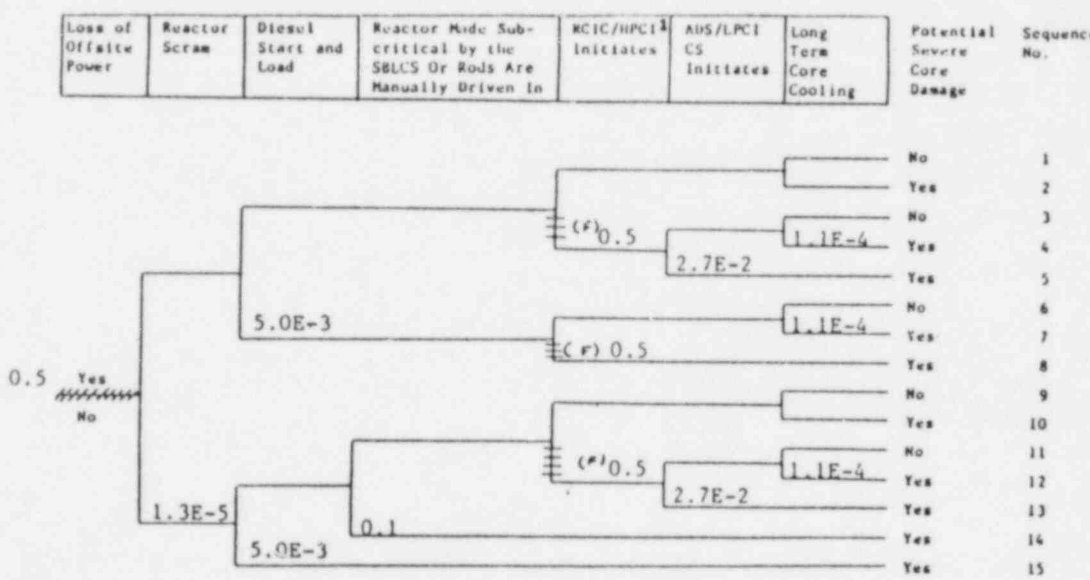
SRC
ORC (Ltd)
FILES: T-5n
Rdg.
A-11
TL



P = 1.2E-4 (S = 39)

NSIC 61043 - Sequence of Interest for Plant Electrical Power Failure at La Crosse

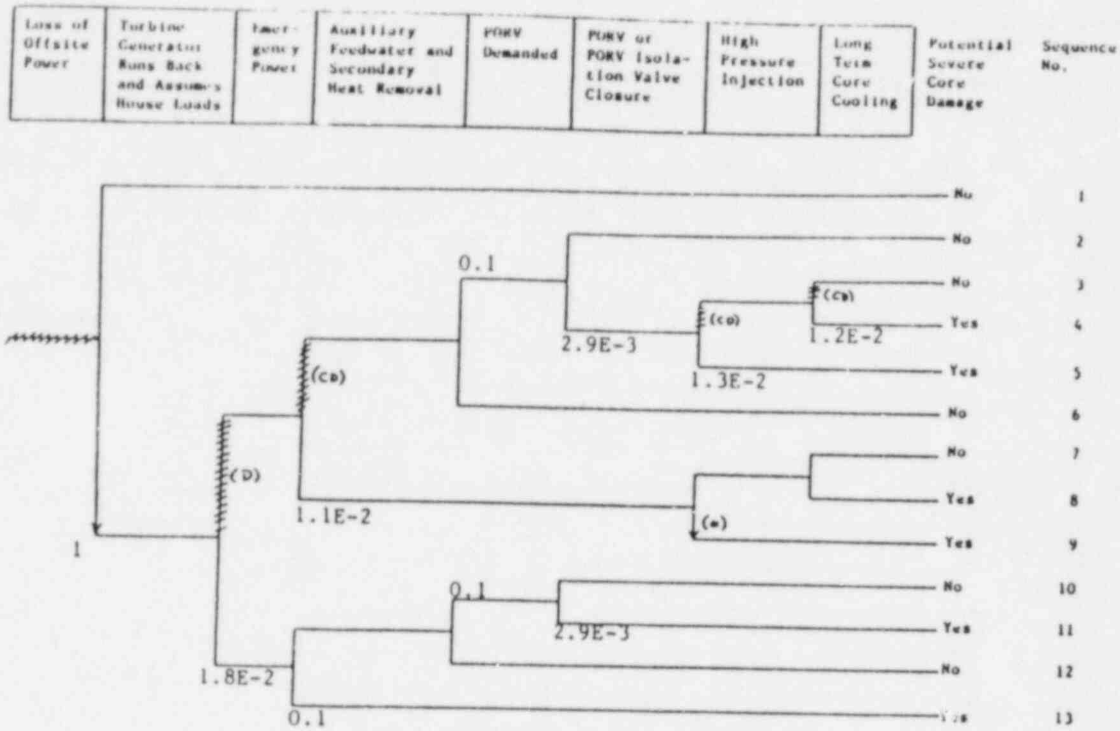
*La Crosse utilizes shutdown condenser and condensate pumps instead of RCIC and HPCI.



P = 8.8E-3 (SC = 21)

NSIC 61434 - Sequence of Interest for Loss of Offsite Power and Failure of an Emergency Condenser Valve to Open at Humboldt Bay

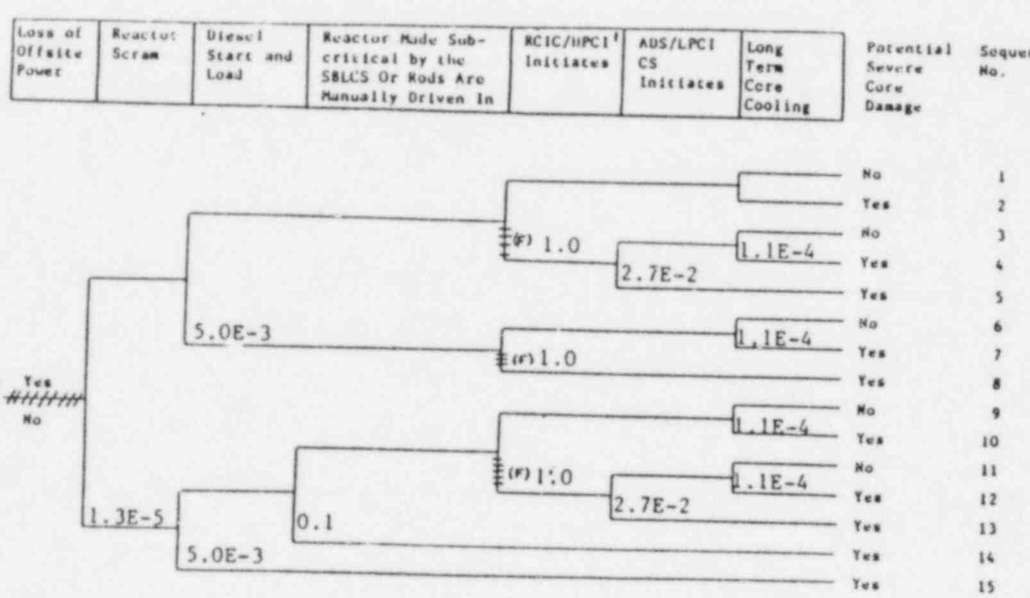
[1]Humboldt Bay utilized an emergency condenser and CRD hydraulic pumps/safety valves for decay heat removal



P = 6.4E-3 (SC = 22)

NSIC 61565 – Sequence of Interest for Loss of Offsite Power and Failure of a Diesel Generator to Load at Palisades

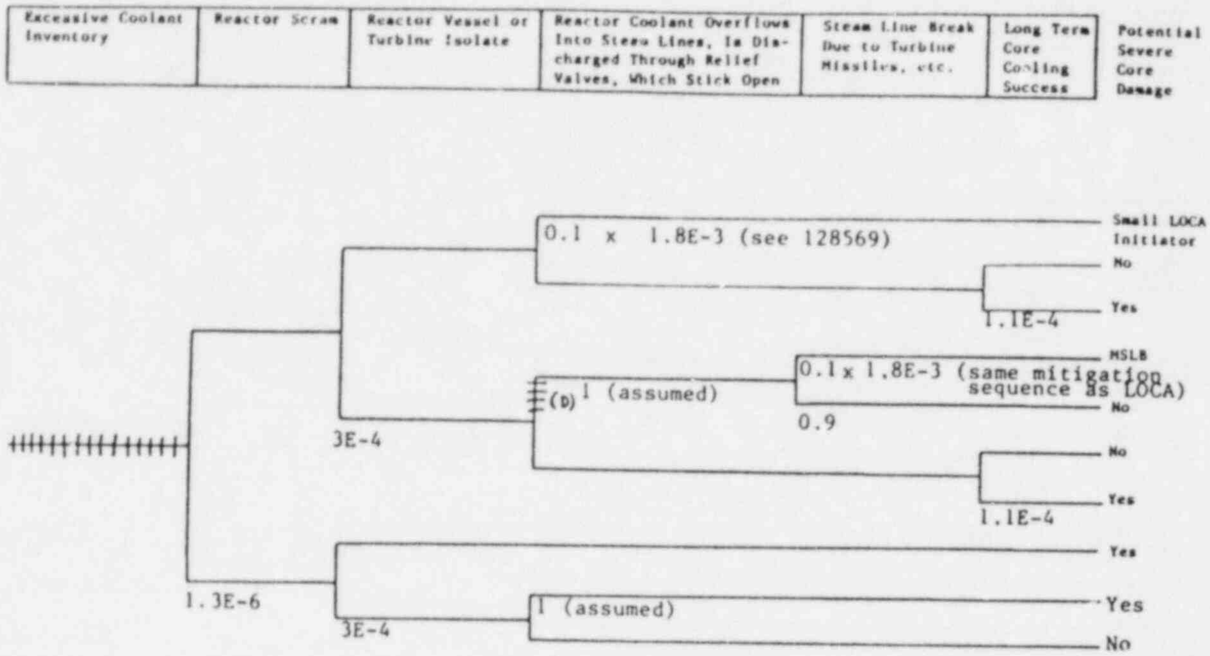
*Use of HPI following AFW failures not included in mitigation procedures.



P = 1.8E-2 (SC = 18)

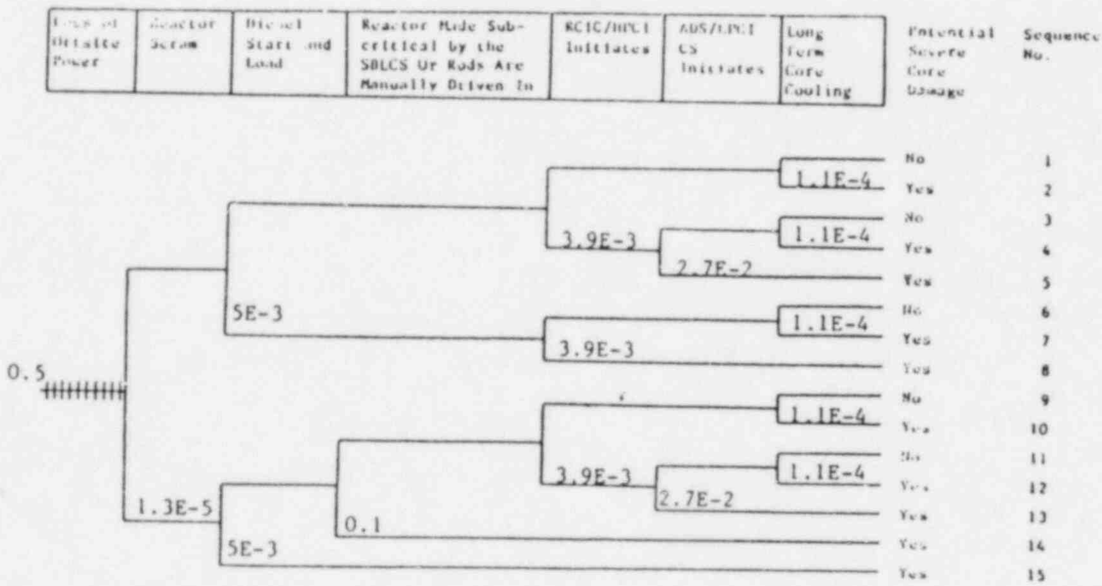
NSIC 63129 – Sequence of Interest for A Scram Caused by Electrical Load Rejection at La Crosse

†La Crosse utilizes shutdown condenser and condensate pumps instead of RCIC and HPCI.



P = 2.8E-4 (SC = 36)

NSIC 74242 – Sequence of Interest for High Coolant Level at Nine Mile Point



P = 1.2E-4 (SC = 39)

NSIC 75074 – Sequence of Interest for Loss of Load and Iodine Release at LaCrosse