UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 23 1979

TERA

Docket No. 50-341

Dr. Wayne H. Jens
Assistant Vice President
Engineering & Construction
The Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Dear Dr. Jens:

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION IN FERMI 2 FSAR

As a result of our continuing review of the Final Safety Analysis Report (FSAR) for the Enrico Fermi Atomic Power Plant Unit 2, we have developed the enclosed requests for additional information.

Please amend your FSAR to comply with the requirements listed in the enclosure. Our review schedule is based on the assumption that the additional information will be available for our review by April 27, 1979. This is the latest date for filing information to be considered in our Safety Evaluation Report for Fermi 2. If you cannot meet this date, please inform us within 7 days after receipt of this letter so that we may revise our scheduling.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

Enclosure:
Requests for Additional
Information

cc w/enclosure:
See page 2

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ENCLOSURE

REQUESTS FOR ADDITIONAL INFORMATION

ENRICO FERMI ATOMIC POWER PLANT UNIT 2

DOCKET NO. 50-341

Requests by the following branches in NRC are included in this enclosure. Requests and pages are numbered sequentially with respect to previously transmitted requests.

Branch	Page No.
Effluent Treatment Systems Branch	010-4
Power Systems Branch	222-21 through 222-29
Analysis Branch - Reactors Analysis Section	230-2 230-3 230-4
Accident Analysis Branch	
Accident Analysis Section	310-9 through 310-13
Emergency Planning Section	422-4 through 422-9

010.0 EFFLUENT TREATMENT SYSTEMS BRANCH

010.2B (RSP) It is our position that the process control program (PCP) for the solidification of radioactive waste be provided for review at least 6 months prior to the fuel loading date. This information will be required to complete a review on the proposed Radiological Effluent Technical Specifications.

222.0 POWER SYSTEMS BRANCH

222.31A
8.2
RSP

Your response to Item 222.31 states that manual operator action is required to isolate the emergency busses from a degraded voltage condition. We find this to be unacceptable and require the installation of an automatically initiated protection scheme which shall satisfy the following criteria:

a) Class 1E equipment shall be utilized and shall be physically located at and electrically connected to the emergency switchgear.

b) An independent scheme shall be provided for each division of emergency power.

c) The selection of voltage and time delay set points shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels;

d) The time delay selected shall be based on the following conditions:

(1) The allowable time delay, including margin, shall not exceed the maximum time delay associated with the availability of power that is assumed in the accident analyses;

(2) The time delay shall minimize the effect of short duration disturbances from reducing the availability of the offsite power source(s); and

(3) The allowable time duration of a degraded voltage condition at all distribution system voltage levels shall not result in failure of safety systems or components;

e) The voltage monitors shall automatically initiate the disconnection of offsite power sources by tripping the emergency bus feeder breaker whenever the voltage set point and time delay limits have been exceeded and the associated diesel generator shall be signalled to start and accept load;

f) The set points for this scheme shall be design dependent but should approximate the following envelopes:

1) Voltage set point between 87 and 90 percent of nominal
2) Time delay setting of between 6 and 10 seconds

g) Capability for test and calibration during power operation shall be provided.

h) Annunciation must be provided in the control room for any bypasses incorporated in the design.

i) The technical specifications shall include limiting conditions for operation, surveillance requirements, and trip set points with minimum and maximum limits.

222.32A
8.2
RSP

Your response to Item 222.32 does not address verification by actual measurement during pre-operational testing that the optimum voltage profiles were identified by your analyses. We require this verification be performed and the results be available for audit at the plant site.

222.33
8.2
RSP

It is the staff's position that the diesel generator bus load shedding feature be automatically bypassed once the diesel generator is supplying power to the bus. This is required so that the voltage drops encountered during load sequencing on the diesel generators will not interact with the load shedding feature and negate the loading sequence. It is further required that the load shedding feature be automatically reinstated should the diesel generator circuit breaker be subsequently opened.

It appears from the logic diagram in FSAR Figure 8.3-2 that your design complies with this position. Provide verification of your compliance or modify your design accordingly.

222.1(1)B
(3.11)

As stated in the Fermi 2 Interim Safety Evaluation Report, the staff has undertaken a generic review of environmental qualification of Class 1E equipment. As one result of this action, we have formulated a standard question, on a plant-specific basis, that addresses the total information required by the staff to complete the review. This question is as follows:

In order to ensure that your environmental qualification program conforms with General Design Criteria 1, 2, 4 and 23 of Appendix A and Sections III and XI of Appendix B to 10 CFR Part 50, and to the national standards mentioned in Part II "Acceptance Criteria" (which includes IEEE Std 323) contained in Standard Review Plan Section 3.11, the information noted below on the qualification program is required for all Class 1E equipment.

1. Identify all Class 1E Equipment, and provide the following:

   a. Type (functional designation)
   b. Manufacturer
   c. Manufacturer's type number and model number
   d. The equipment should include the following, as applicable:

      1) Switchgear
      2) Motor control centers
      3) Valve operators
      4) Motors
      5) Logic equipment
      6) Cable
      7) Diesel generator control equipment
      8) Sensors (pressure, pressure differential, temperature and neutron)
      9) Limit Switches
      10) Heaters
      11) Fans
      12) Control Boards
      13) Instrument racks and panels
      14) Connectors
      15) Electrical penetrations
      16) Splices
      17) Terminal blocks

2. Categorize the equipment identified in (1) above into one of the following categories:

a. Equipment that will experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.

b. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand any accident environment for the time during which it must not fail with safety margin to failure.

c. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, and whose failure (in any mode) is deemed not detrimental to plant safety or accident mitigation, and need not be qualified for any accident environment, but will be qualified for its non-accident service environment.

d. Equipment that will not experience environmental conditions of design basis accidents and that will be qualified to demonstrate operability under its normal or abnormal service environment. This equipment would normally be located outside the reactor containment.

3. For each type of equipment in the categories of equipment listed in (2) above provide separately the equipment design specification requirements, including:

   a. The system safety function requirements.

   b. An environmental envelope as a function of time which includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown, normal operation, abnormal operation, and any design basis event (including LOCA and MSLB), including post event conditions.

   c. Time required to fulfill its safety function when subjected to any of the extremes of the environmental envelope specified above.

   d. Technical bases should be provided to justify the placement of each type equipment in the categories 2.b and 2.c listed above.

4. Provide the qualification test plan, test set-up, test procedures, and acceptance criteria for at least one of each group of equipment of (1.d) as appropriate to the category identified in (2) above. If any method other than type testing was used for qualification (operating experience, analysis, combined qualification, or on-going qualification), describe the method in sufficient detail to permit evaluation of its adequacy.

5. For each category of equipment identified in (2) above, state the actual qualification envelope simulated during testing (defining

the duration of the hostile environment and the margin in excess of the design requirements). If any method other than type testing was used for qualification, identify the method and define the equivalent "qualification envelope" so derived.

6. A summary of test results that demonstrates the adequacy of the qualification program. If analysis is used for qualification, justification of all analysis assumptions must be provided.

7. Identification of the qualification documents which contain detailed supporting information, including test data, for items 4, 5 and 6.

In addition, in accordance with the requirements of Appendix B of 10 CFR 50, the staff requires a statement verifying: 1) that all Class 1E equipment has been qualified to the program described above, and 2) that the detailed qualification information and test results are available for an NRC audit.

In this regard, we request that you supplement as necessary, the information contained in the responses to Item 222.1 as well as Section 3.11 of the FSAR so as to provide a response for each of the items noted. (It is not necessary to provide a response to any asepct of these items which has been previously addressed).

222.35 (7.7) During the Hatch 2 operating license review the following item was identified and a staff position was taken. It was noted that the non-Class 1E startup monitoring and recording system used to monitor and record selected parameters during reactor startup could, if not properly designed and installed, degrade the operation of systems required for safety. The applicant was requested to provide a description of the system and a commitment to remove all connections at the sources of their signals upon the completion of startup and warranty testing.

We understand the Fermi 2 design to be identical to the above described Hatch 2 design. We therefore request a brief description of the physical implementation of this design at Fermi 2 with specific emphasis on the measures you have taken to assure that the cable routing does not violate your separation criteria nor jeopardize Class 1E circuits.

222.36 (App. 10A) Appendix 10A of the FSAR, Steam Bypass System, states in section 10.A.4.1: "The system is designed so that any postulated failure will not cause both valves to fail to open in the fast opening mode of operation coincident with the closure of the turbine-stop or throttle valves."

Provide a failure mode and effects analysis which demonstrate the above referenced capability.

Section 10A.4.2 states: "Each module of each channel has its own power supply which is connected to two independent ac sources. Each module power supply can use either source to supply its requirements."

Provide or reference the appropriate electrical one-line diagrams and cable routing diagrams to allow an independent evaluation of the above referenced statements. Supplement the drawings with a brief narrative explaining the intended modes of operation.

230.0 Analysis Branch, Reactor Analysis Section

230.2 (4.4.1.1) Under item c of your design bases, you state that "thermal-hydraulic design of the core shall establish the nuclear system so that it exhibits no inherent tendency toward divergent oscillations that would compromise the integrity of the fuel or nuclear system process barrier." This is not consistent with the Standard Review Plan requirement that "The reactor should be demonstrated to be free of undamped oscillations or other hydraulic instabilities for all conditions of steady-state operation, for all operational transients, for all load-following maneuvers, and for partial loop operation." Either revise your design basis and show compliance with the Standard Review Plan or justify the deviation from the Standard Review Plan.

230.3 (4.4.2.2) The GEXL data base (for the approved correlation) is for 7 x 7 and 8 x 8 one water rod bundles. No substantial data base has been provided to support the 8 x 8, two water rod design. The GEXL correlation must be demonstrated to be applicable to the new 8 x 8 design, by comparison to applicable data, prior to issuance of an operating license for Fermi-2. Alternatively, the MCPR limit may be increased by 0.05 to accommodate GEXL uncertainties.

230.4 (4.4.2.5) You state on page 4.4-6 that "There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor." Does this refer specifically to Fermi-2 calculations? What operating reactor was used for the data comparison?

230.5 (4.4.4.5) Page 4.4-16 of the FSAR states that "the nominal expected bypass flow fraction is approximately 10 percent." What is the calculated bypass flow fraction for Fermi -2 and what is its uncertainty?

230.6 (4.4.4.5) What fraction of the fuel bundle flow is "water rod flow"?

230.7 (4.4.4.5) Page 4.4-17 states "Analytical models of the individual flow paths were developed as an independent check of the tests. ... when using these models for hydraulic design calculations, nominal drawing dimensions are used." Provide the assumptions and equations comprising the model and a comparison of model predictions with data.

230.8 (4.4.4.5) In discussing thermal-hydraulic stability, applicants for BWR's have traditionally included operational design guides for decay

*Question asked on one or more other dockets.

ratios and damping factors. These design guides have been omitted from your discussion of stability. Are the design guides no longer applicable? If not, explain why.

230.9 (4.4.4.6) In discussing the FABLE code on page 4.4-20, you state that "As new experimental or reactor operating data are obtained, the model is refined to improve its capability and accuracy." This means that comparison of old versions of the model with data, as given in Figure 4.4-8, are meaningless for Fermi-2 if it has been analyzed with an updated version. Are the comparisons of the model with data, as given in Figure 4.4-8, based on the same version of the model as was used for Fermi-2? If not, provide comparisons using the Fermi-2 model. In addition, provide a description of the code or reference a prior licensing submittal (other than the KAPL reports on STABLE).

230.10 (4.4.4.6) On pages 4.4-20 and 4.4-21, the REDY code is referenced as the model used to perform system stability calculations. You also state that the model is periodically refined as new experimental or reactor operating data are obtained. Is the version of REDY used for Fermi-2 described in NEDO-10802? If not, describe the changes.

230.11 (4.4.4.6) On page 4.4-21 you state that "the most limiting condition occurs near the end-of-cycle one." This is contrary to experimental results obtained from Peach Bottom stability tests and is contrary to analyses previously submitted for Fermi-2 (Amendment 1 - November 1975 of Fermi-2 FSAR, page 4.4-25). While the condition may be the most limiting for cycle one, cycle one conditions are not the most limiting. Provide the power profile and the void reactivity coefficient used for the analysis.

230.12 (4.4.4.6) What is meant by "most responsible attainable mode" as used on page 4.4-21 in the discussion of the stability analysis results.

230.13 Provide the assumptions used for the amount of crud buildup in the design calculations and the sensitivity of CPR and core pressure drop to variations in the amount of crud present. Also, provide data supporting the assumption on crud thickness and discuss how crud buildup in the core would be detected.

230.14 (4.4.4.6) We will require that a loose parts monitoring system be installed and operational prior to startup testing. Therefore, a description of the system is needed for our evaluation prior to issuance of an operating license. Guidance regarding the design requirements for a LPM system can be found in draft Regulatory Guide 1.133 (Loose-Part Detection Program for the Primary System of Light-Water-Cooled-Reactors). Provide a schedule for submittal of your LPM system description.

230.15 (Table 4.4-6) Table 4.4-6 describes uncertainties used in the statistical analysis which is performed to establish the fuel cladding integrity safety MCPR limit. Provide a discussion of and reference where possible the experimental data bases used to derive the uncertainty values listed. In particular, describe the applicability of these values to the 8 x 8, two-water rod assembly design.

310.0 ACCIDENT ANALYSIS BRANCH

310.15 It is our position that the RHR mechanical draft towers must either be protected against vertical tornado missiles or the applicant must demonstrate that appropriate action can be taken to ensure that the plant can be shutdown and maintained in a safe shutdown condition utilizing safety grade equipment that is protected from the effects of tornadoes.

310.16 The applicant has requested that the MSIV technical specification leakage be increased from 46 scfh to 100 scfh (combined leakage from all four MSIV's).

The applicant has committed to install a positive seal MSIV-LCS that removes the MSIV's as a potential fission product leak path. Therefore, it is our position, that the requested increase in the MSIV leakage technical specification is acceptable provided that

(1) the applicant can demonstrate that there is no measurable increase in containment leakage due to air infiltration at 100 scfh and

(2) the proposed positive seal MSIV-LCS meets the criteria as specified by the appropriate Standard Review Plans and Regulatory Guides as to operation and testing

310.17 Having reviewed the FHA analysis in FSAR, Section 15B.7, we find that additional information is necessary to adequately evaluate the consequences of a FHA inside containment. Describe all instrumentation which would detect a fuel-handling accident (FHA) inside containment. Your response should include the following information:

(1) instrumentation function, e.g., close containment isolation valves;

(2) type of instruments and setpoints, e.g., mr/hr, and normal background reading;

(3) safety class, redundancy, power sources and technical specification requirements;

(4) a description of instrument sensitivity.

(5) response time for the instrument to signal containment isolation after the FHA.

310.18 Describe the response of the containment isolation and ventilation valves following the FHA. Include valve closure times indicating the expected valve closure time as well as technical specification requirements.

310.19 Provide the transit time from the point where a monitor can respond to a release from the FHA to the inboard isolation valve based on the maximum air velocity (peak centerline velocity) at maximum exhaust flow. Also include the transit time based on average velocity and normally expected air flows. Conservatively assume that the FHA results in a puff release from the pool at a point closest to an exhaust grill.

310.20 Provide drawings of the containment which clearly show the location of the radiation monitors relative to the ventilation exhaust system including all exhaust inlets, filters, dampers

and duct arrangement up to the outboard isolation valves.

310.21 If the summation of the instrument response time (question 310.17(5)) and valve dampers closure time (question 310.18) is greater than the gas transit time (question 310.19), provide an analysis of the curies of radioactivity that could be released in the exhaust air. Your response should include the methodology used to calculate gas transit times from the pool surface to the exhause system and the air velocity profiles over the pool surface. You should consider the effects of pool water temperature on air flow trajectories where these effects may be significant.

310.22 (6.1.2) For the organic coatings listed in Table 6.2.8, indicate whether each coating (paint and method of application) is qualified according to the recommendations of Regulatory Guide 1.54. If the recommendations of Regulatory Guide 1.54 are not followed, describe the quality assurance program you propose to use for each coating.

310.23 Experience with present leak test intervals for MSIV's would indicate that, for extended intervals of MSIV leak testing, gross MSIV leakage should be considered in evaluating accidents.

Therefore, it is our position that the standard technical specification values for permissable leak test intervals shall be utilized unless the following can be demonstrated:

(1) Gross leakage of the MSIV's would not result in an increased containment leakage from excessive pressurization of the

containment that the calculated radiological consequences of the LOCA will not be in excess of 10 CFR Part 100 guidelines.

(2) Gross leakage would not result in a degradation of the proposed positive seal system and result in untreated leakage downstream of the third block valve resulting in calculated radiological consequences for the LOCA in excess of 10 CFR Part 100 guidelines.

310.24 The earlier review of aircraft hazards identified only three small airports in the site vicinity and identified the largest aircraft likely to over fly the site at low altitude as weighing 3400 lbs. Changes since that time have occurred which require re-examination of that review.

There are five airports within 20 km of the Fermi-2 site, listed below:

Airport	Direction from Site	Range	Length of Longest Runway
Marshall	SW	4 km	2100 feet
Carl	NNW	8	2300
Wickenheiser	NW	11	2500
Custer-Monroe Municipal	SW	15	3500
Gross Ile Municipal	N	16	5000

Only the last two are among the 3137 airports in the National Airport System Plan and both are scheduled to be expanded to Basic

Transport Airports, i.e., capable of use by aircraft up to 60,000 lbs.

Survey each of the three smaller airports and report the distribution by type of aircraft based and operations per year expected to be flown. Also report any plans for increasing available series, facilities, or runway length and investigate the likely effect upon these three airports future operations of the expansions of the two municipal airports.

422.0 EMERGENCY PLANNING

Listed below are requests for additional information in the area of emergency planning. Response should be incorporated in page changes to FSAR Appendix 13A, Emergency Manual and Protective Action Guidance. The specific section or subsection of Appendix 13A is identified by the number in parenthesis. The references to Regulatory Guide 1.101, Emergency Planning for Nuclear Power Plants, are to Revision 1 dated March 1977.

422.13 (13A.1.1.6, 13A.1.1.7) Provide the information for these two sections: Definitions and abbreviations.

422.14 (13A.1.4.2.3) Provide a listing of the technical support personnel, by job title, available to supplement the emergency organization.

422.15 (13A.9) Replace the Radiological Assistance Handbook in this section with the current ERDA Appendix 0601, "Emergency Planning, Preparedness and Response Program" dated December 1, 1976.

422.16 (13A.10) Provide a copy of an agreement with the Department of Energy Chicago Operations Office and the Michigan Department of Natural Resources which identifies their role in radiological emergency response for the Fermi site and environs.

422.17 (13A.10) Submit an updated agreement with the following offiste agencies included in Section 13A.10:

Monroe City - County

Consumers Power Co., Detroit Edison Co., and Toledo Edison Co.

Navarre Monroe County Ambulance Service

Monroe County Sheriff

Seaway Hospital

Michigan State Police

United States Coast Guard

422.18 (13A.10.1) The regulations, at 10 CFR 50, Appendix E, Part III require that the applicant provide reasonable assurance that appropriate and timely response measures can and will be taken in the event of an emergency to protect individuals in the environs of the site. The State of Michigan Nuclear Facility Emergency Plan (November 3, 1977) and the Monroe City-County Emergency Plan (October 2, 1975) are not complete in this regard. Listed below are the elements that should be addressed (as appropriate) for each state and local agency having a response role in support of the Fermi-2 emergency plan. Please provide this information, or ensure that any revised state and local plans submitted have been reviewed for completeness with respect to these elements.

1. The identity of the agency.
2. A description of the authority and responsibility for emergency response functions.
3. A description of the concept of operations including the operational interrelationships of all organizations having emergency response roles.
4. The designation and location of the Emergency Operations Center for the direction and/or coordination of emergency support activities.
5. The established relationship and interface with State and/or local government emergency response plans.
6. The provisions established with the Department of Energy Regional Coordinating Office for radiological assistance under the RAP and IRAP programs.

7. A description of the communication plan for emergencies including titles and alternates for both ends of the communication links, and primary and backup means of communication. Where consistent with the agency function, include the following:

   a. Provision for 24-hour/day manning of communication link.

   b. Provision for administrative control methods for ensuring the effective coordination and control of the emergency support activities.

   c. Provision for communications arrangements with contiguous local governments where applicable.

   d. Provision for communications arrangements with Federal emergency response organizations.

   e. Provision for communications with the nuclear facility, State and/or local emergency operations centers, and field assessment teams.

8. A description of the communications methods for issuing emergency instructions to the public in the potentially affected environs of the nuclear facility.

9. A description of the methods, including meteorological assessment methods, and equipment to be employed in determining the magnitude and disposition of liquid or gaseous radioactivity releases.

10. The designation of protective action guides and/or other criteria to be used for implementing specific protective actions and the informational needs (e.g., dose rates, projected dose levels, contamination levels, airborne or waterborne activity levels) for implementing such actions.

11. A description of the methods for protecting the public from consumption of contaminated foodstuffs.

12. A description of the evacuation plans for the Low Population Zone (LPZ) including survey maps for the facility environs showing evacuation routes as well as relocation and shelter areas. The plans may extend to areas beyond the LPZ and should include the following:

   a. Population and their distribution around the nuclear facility.

b. Means for notification of the potentially affected population.

c. Disabilities, institutional confinement, or other factors which may impair mobility of parts of the population.

d. Means of effecting relocation.

e. Potential egress routes and their traffic capacities.

f. Potential impediments to use of egress routes.

13. The provisions for maintaining dose records of all potentially exposed emergency workers involved in response activities.

14. The provisions for emergency drills and exercises to test and evaluate the response role of the agency, including provisions for critique by qualified observers.

15. A description of the training program established for those individuals having an emergency response assignment.

16. The provisions for periodic review and updating of the emergency response plans of the agency.

422.19 (13A.10.1) Note that the Emergency Classification in the Fermi-2 emergency plan and in the Monroe City-County Emergency Plan (Class I through IV) is not consistant with the State of Michigan Nuclear Facility Emergency Plan and the rules of the State Department of Public Health (Class C through A). This inconsistency may lead to confusion in emergency response. Please correct this situation.

422.20 (13A.10.1) Provide the "Fold out County Map", referenced as Enclosure II to the Monroe City-County Emergency Plan (Three copies only required on the docket.)

422.21 (13A.10.5) Submit a copy of a letter of agreement with the University Hospital and the Bennett Ambulance Service to provide evidence of the support

expected in the event the Fermi-2 plant requests assistance.

422.22 (13A.10.11) Provide a copy of the most current State of Michigan Nuclear Facility Emergency Plan. Include Section XVI and Attachments A through J, missing from the document dated November 3, 1977. (This material was not included with the Working Draft, November 3, 1977).

422.23 It appears that portions of previous question 422.2 have not been answered. These portions are repeated below:

Provide an analysis including specific information and findings that will be needed to ensure adequacy of emergency planning with respect to evacuation of persons from the exclusion area and from any potentially affected sector of the environs, including the following:

(1) The expected accident assessment time. This figure should incorporate the time required to identify and characterize the accident, the time needed to predict the projected doses resulting from the accident, the time to notify off-site authorities, and the time required to determine the appropriate protective measures for the affected areas. Include sufficient information to support the estimate.

(2) An estimate of the time required to notify the population at risk and the means assumed for such estimate.

(3) An estimate of the evacuation times, measured from the time of initial warning, needed to remove persons from the exclusion area and from each "sector", or increment thereof, of the environs out to a distance determined by the 8-hour terminus of the 5 rem whole body dose curve, the 25 rem thyroid dose curve, or the outer low population boundary, whichever is the greatest. The "sectors" chosen for analysis may be bounded by certain geographical or man-made features, but they should cover an arc of at least 45°. Population data should include both resident and transient persons, including those resulting from the facilities described in Chapter 2 of the PSAR. Population levels projected as peak values during the expected life of the plant should be used.

(4) The identity and estimates of vehicular traffic capacities of each egress route at the point of departure from the area to be evacuated. If means for effecting the physical evacuati ..s other than the use of private automobiles are used in estimating any of the foregoing evacuation times, these should be specified.

TERO

DAIRYLAND POWER COOPERATIVE

La Crosse, Wisconsin

54601

January 30, 1979

In reply, please
refer to LAC-6095

DOCKET NO. 50-409

Mr. James G. Keppler
Regional Director
U. S. Nuclear Regulatory Commission
Directorate of Regulatory Operations
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
IE BULLETIN NO. 78-12A - ATYPICAL WELD
MATERIAL IN REACTOR PRESSURE VESSEL WELDS

Reference: (1) NRC Letter, Keppler to Linder, dated November 24, 1978.
(2) DPC Letter, LAC-6012, Linder to Keppler, dated November 27, 1978.
(3) NRC Letter, Keppler to Madgett, dated September 29, 1978.

Dear Mr. Keppler:

In response to your letter (Reference 1) which contained IE Bulletin No. 78-12A, in which the response requirements for IE Bulletin No. 78-12 have been modified, please refer to our letter (Reference 2) which contains all available information requested by IE Bulletin No. 78-12 (Reference 3).

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

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder, General Manager

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