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

DEC 5 1978

Docket No. 50-358

Mr. Earl A. Borgmann
Vice President - Engineering
Cincinnati Gas & Electric Company
P. O. Box 960
Cincinnati, Ohio 45201

Dear Mr. Borgmann:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION
(Wm. H. Zimmer, Unit No. 1)

In order that we may continue our review of your application for a license to operate the Zimmer Nuclear Power Station, Unit No. 1, your response to the enclosed positions (and requests for additional information) are needed. The positions (and requests) are based upon information contained in your application as amended through Revision 48 and your response to our previous requests.

We will need your response to these positions (and requests) by December 29, 1978. If you cannot meet this date, please advise us within two weeks of receipt of this letter of your date for complete response.

Please contact us if you desire information or clarification regarding the enclosure.

Sincerely,

John F. Stolz

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

Enclosure:
Request for Additional
Information

cc: See page 2

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<u>ENCLOSURE</u>

<u>REQUEST FOR ADDITIONAL INFORMATION</u>

<u>WM. H. ZIMMER ROUND-TWO QUESTIONS</u>

<u>Introduction</u>

This enclosure consists of the twentieth in a series of positions (and requests for additional information). We will need your response to them in order to complete our safety evaluation of your Zimmer OL application. The position (and requests) are in the area of:

212.0 <u>Reactor Systems</u>

221.0 <u>Electrical Instrumentation and Control Systems</u>

121.0 <u>Materials Engineering - Materials Integrity</u>

110.0 <u>Mechanical Engineering</u>

It will be helpful to us if your responses are in a "Position and Response" format using the same number designation as the position. The first number designated the review area and the second (in parentheses) designated the associated section of the FSAR. Of course, your responses should include revision to the FSAR wherever appropriate.

NOTE: The positions stated in the 212.0 and 221.0 series are intended to resolve the staff's concern about the use of non-safety grade equipment for the mitigation of the excess feedwater event.

The information requested in the 121.0 series is needed for the justification of exemptions to Appendix G and H of 10 CFR, Part 50.

The information requested in the 110.0 series is relevant to the potential design deficiency recently documented by I&E in Operating Experience Memorandum #13, "Piping Support Base Plate Problem," dated June 28, 1978.

212.0 Reactor Systems

212.74 In analyzing anticipated operational transients, the applicant has taken credit for plant operating equipment which has not been shown to be reliable as required by General Design Criterion 29. The staff has discussed the application of this equipment generically with General Electric. Based on these discussions, it is the staff's understanding that the most limiting transient that takes credit for this equipment is the excess feedwater event. Further, it is the staff's understanding that the only plant operating equipment that plays a significant role in mitigating this event is the turbine bypass system and the Level 8 high water level trip (closes turbine stop valves).

In order to assure an acceptable level of performance, it is the staff's position that this equipment be identified in the plant Technical Specifications with regard to availability, set points, and surveillance testing. The applicant must submit his plan for implementing this requirement along with any system modifications that may be required to fulfill the requirements.

221.0 Electrical Instrumentation and Control Systems

221.387(RSP) (7.7) (F7.7-8) It is the staff's position that the Zimmer FSAR is not sufficiently complete so as to demonstrate that feedwater system level sensors N004A, N004B and N004C are electrically isolated from each other. Therefore, the staff requires that you:

(1) Revise the FSAR, Section 7.7 to clearly describe the design and qualification of the circuitry and equipment which is common to two or more of the level 8 sensor/alarm trip unit channels utilized in the feedwater control system. This revision should contain sufficient information and drawings to permit the staff to review the feedwater control system as specified in Section 7.7 of the Standard Review Plan.

(2) Demonstrate by using the material which is provided in the response to (1) above, that the N004A, N004B and N004C signal paths are independent.

(3) Justify not removing the plant process computer inputs A1723 and A1727 from the feedwater control system.

121.0 Materials Engineering Branch - Materials Integrity Section

121.7 In the response to Question 121.2(2) submitted as an attachment to your letter dated February 13, 1978, you stated that the shell plate NDTT was determined by drop weight testing. Provide the Charpy V-notch test results for the shell plates if the testing was performed.

121.8 Since the testing to determine fracture toughness properties was performed prior to Appendix G of 10 CFR Part 50 and therefore may not be consistent with requirements of Appendix G, provide details of methodology to establish the initial RT_{NDT} considering the current Appendix G requirements. Provide a technically justified estimate of the unirradiated CVN minimum upper shelf energy. Based on the above, state if the adjusted RT_{NDT} is $67^{o}F$ as previously stated and estimate the decrease in the upper shelf energy at the EOL. Consideration should be to testing a sufficient number of the unirradiated Charpy samples from the surveillance program to aid in the estimation of the initial upper shelf energy.

121.9 Question 121.3 requested that any deviations in the surveillance program from Appendices G and H, 10 CFR Part 50 be stated. Since no response to this item was received, provide this information.

121.10 FSAR paragraph 5.2.4.4.1 states the capsule holder brackets which were designed, fabricated, and analyzed to ASME Code Section III are attached to the vessel cladding. Provide details of the welding and inspection of the attachment of the brackets to the cladding.

121.11 State if all material for bolting and other fasteners greater than one inch diameter in the reactor coolant pressure boundary meet the exact fracture toughness requirements of 10 CFR 50, paragraph IV.A.4. Provide the fracture toughness data available for all bolting greater than one inch diameter used for the reactor coolant pressure boundary. Indicate the ASME Code Edition used for these components.

121.12 State if the requirements of 10 CFR Part 50, Appendix G, paragraphs IIIB.3., 4, and 5 were met for the fracture toughness tests which were performed. Clarify if the testing was performed by an organization with a quality assurance program in conformance with 10 CFR Part 50, Appendix G. Provide sufficient detail to clarify what calibration practices, personnel qualifications and reporting of results were actually utilized.

110.0 Mechanical Engineering Branch

110.29 Appendix XVII-2461.1 of the ASME Code Section III requires that bolt loads in bolted connections for linear component supports include prying effects due to the flexibility of the connection.*

1. Provide confirmation that the loads in bolted connections for linear component supports were determined by considering the deformation of the connection and tension-shear interaction for the bolts.

   For connections of supports which are anchored to a concrete structure provide in addition:

   a. The type of anchor bolt.

   b. The factors of safety (and their bases) against pullout under static, repeated and transient loading.

   This information should include representative diagrams of the connections, material properties, and interaction diagrams, the analytical techniques and models used, and the maximum stresses in the bolts and the connections under both static, repeated, and transient type loading.

2. If any connection was assumed to be rigid, provide complete analytical or experimental justification for this assumption.

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*Similar requirements for structural joints are also stated in the AISC Manual of Steel Construction, 1970 Edition for plants in which support design predates Subsection NF of Section III of the ASME Code.

121.13 Provide the fracture toughness data available for ferritic piping, pumps and valves used for the reactor coolant pressure boundary. Indicate the code requirements used for these components. A summary of test results, rather than the copies of the certified test reports, is acceptable.

121.14 The FSAR, paragraph 5.24.1.2, states that the specific temperature limits for operation when the core is critical were based on analysis of the vessel heads and shell areas remote from discontinuities. Provide sufficient details of the analysis to support the pressure/temperature limits which were established.