NUREG-0528
Supplement No. 3

Safety Evaluation Report

related to the operation of Wm. H. Zimmer Nuclear Power Station, Unit No. 1

Docket No. 50-358

Cincinnati Gas and Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

August 1982

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Safety Evaluation Report for the Zimmer Nuclear Power Station, Unit 1 (NUREG-0528), Docket No. 50-358, was issued January 1979. Supplements 1 and 2 to that report were issued June 1981 and October 1981, respectively. This supplement (Supplement 3) addresses the resolution of outstanding and confirmatory items that remained in our review at the time Supplement 2 was issued. Each item for which there is a change in status since issuance of Supplement 2 is addressed in the appropriate subsection which is numbered to correspond to the subsection in which it was discussed in Supplement 2. The following table summarizes those items for which the status has changed in this supplement.

Subsection	Topic	SSER 2 Status	SSER 3 Status
4.6.2	Scram Discharge System	Confirmatory	Resolved
4.6.2	Scram System Pipe Break	--	Confirmatory
6.2	Adequacy of BWR Pressure Suppression Containment	--	Outstanding
6.2.1	LOCA Pool Dynamic Load	--	License Condition
6.2.1	Vacuum Breaker Performance	--	Outstanding
6.2.6	Containment Leakage Testing	Outstanding	Resolved
6.3.4	Implementation of NUREG-0803 Guidelines (Scram System Pipe Break)	License Condition	Moved to Section 4.6.2
7.1.3	Physical Separation of Associated Cables	Confirmatory	Resolved
7.5.3	Loss of Power to Instruments and Control Systems	--	Resolved
8.1.2	Station Blackout Events	License Condition	Confirmatory Item
13.7	Industrial Security Plans	--	License Condition
II.E.4.1	Dedicated Hydrogen Penetration	Confirmatory	Resolved
II.F.1	Containment High Range Radiation Monitor	Confirmatory	Resolved
II.F.2	Incore Thermocouples	Confirmatory	License Condition
II.K.3.27	Common Water Level Reference	Confirmatory	Resolved

A summary of the remaining open items, confirmatory items, and license conditions is provided in Sections 1.8, 1.9, and 1.11, respectively. In addition, six subsections are included in this supplement for the purpose of clarification; these are subsections 4.4.1.1, 4.4.1.2, 5.2.4, 6.2.3, 7.5.3, and 13.7.

The Zimmer plant is currently undergoing a thorough quality assurance investigation because of construction difficulties. These activities are being undertaken by the NRC regional office and will be processed and reported through their normal channels.

The NRC Project Manager assigned to the Operating License application for Zimmer is Dr. Gordon E. Edison. Dr. Edison may be contacted by calling (301) 492-7219 or writing:

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Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

This Safety Evaluation Report Supplement is a product of the NRC staff and its consultants. NRC staff members and consultants who were principal contributors to this report are identified in Appendix I.

1.8 Summary of Outstanding Issues

The resolution of the one remaining outstanding issue listed in subsection 1.8 of NUREG-0528, Supplement 2, is discussed in Section 6.2.6 of this supplement.

Two new outstanding issues have been identified in this supplement to NUREG-0528, and are listed below. These issues will be resolved prior to a decision to issue an operating license.

Issue	Subsection
· Wetwell-to-Drywell Vacuum Breaker Performance Under Accident Conditions	6.2.1
· Adequacy of the BWR Pressure Suppression Containment Systems	6.2

1.9 Summary of Confirmatory Items

Those confirmatory items listed in subsection 1.9 of Supplement 2, for which a change in status has occurred, are discussed in the corresponding subsections of this supplement.

Items which still require confirmation at the time of this supplement are listed below. If implementation of some staff requirements is not confirmed prior to a decision to issue an operating license, the items remaining to be confirmed may be made a condition of the operating license.

Issue	Subsection
• Toxic Chemicals	2.2.1, 6.4.2
• Suppression Pool Hydrodynamic Loads on Concrete Containment and Internal Structures	3.8.1, 3.8.2
• Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment	3.10
• Environmental Design of Mechanical and Electrical Equipment	3.11
• Seismic and LOCA Loading	4.2.3
• Scram System Pipe Break	4.6.2
• Control System Failures Resulting From Loss of Common Electric Power Sources, Sensors, or Exposure to High Energy Line Breaks	7.7.3
• Station Blackout Events	8.1.2

1.11 NUREG-0737 "Clarification of TMI Action Plan Requirements"

Listed below are the NUREG-0737 requirements for Zimmer which need further consideration by the staff and applicant in order to achieve confirmation of full implementation on the Wm. H. Zimmer Nuclear Power Station. We will report further on these matters in a future supplement to this report.

Item (NUREG-0737)	Short Title	Additional information from applicant	Confirmation by the staff
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	X	X
III.A.1.1	Upgrade Emergency Preparedness	X	X
III.A.2	Improving Licensee Emergency Preparedness-Long Term	X	X
III.D.3.4	Control Room Habitability	X	X

1.12 Operating License Conditions

The following requirements will be made a condition of the operating license.

Requirement	Subsections
• Implementation of NUREG-0577 Guidelines	5.2.3
• Mark II Containment LOCA Pool Dynamic Load	6.2.1
• Degraded Grid Voltage	8.1.2

1.13 Other Matters Resulting From the Staff Updated Review

By letter from E. A. Borgmann to Harold Denton, dated November 2, 1981, Cincinnati Gas & Electric Company stated the company's position that the Wm. H. Zimmer Nuclear Power Station, Unit 1 complies with the applicable regulations of 10 CFR Parts 20, 50, and 100 except in those cases where specific exemptions have been granted by the Nuclear Regulatory staff. "Conformance with Applicable Regulations" is summarized in Appendix N to the Final Safety Analysis Report.

2 SITE CHARACTERISTICS

2.2 Nearby Industrial, Transportation and Military Facilities

2.2.1 Transportation of Toxic Chemicals

On January 6, 1982, the applicant submitted a report on the results of a hazardous cargo truck survey on U.S. Route 52 in the vicinity of the Zimmer plant site. The applicant has committed to provide appropriate protection for the Zimmer control room if the results of the survey show that the probability of a toxic chemical release (for which the control room does not have protection), which could jeopardize the control room, exceeds 10^{-6}. The staff is continuing to review this report.

Results of this evaluation will be provided in a future supplement.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.8 Design of Seismic Category I Structures

In Supplement Nos. 1 and 2 it was indicated that the applicant had not completed his assessment of the containment structure and its internal structures for the Pool Dynamic Loads. In Amendment 17 to the Design Assessment Report dated February 16, 1982, the applicant submitted for staff review the results of the assessment. The following is the staff's evaluation of the applicant's assessment.

3.8.1 Concrete Containment

The Zimmer Unit 1 containment was originally designed to resist various combinations of dead loads, live loads, environmental loads, including those due to wind and tornado, operating basis earthquake (OBE) and safe shutdown earthquake (SSE), and loads generated by the design basis accident (DBA) resulting mainly in high pressure and temperature. However, it was later identified that besides these loads traditionally associated with normal operation and DBA, additional suppression pool hydrodynamic loads had not been included explicitly in the original design basis of all Mark II containments of BWR plants under construction (including Zimmer 1). These additional loads can occur not only from postulated LOCA but also as a result of the actuation of safety relief valves (SRV) in normal plant operation. The required consideration of the additional loads is generic to all plants using Mark II containments. In an attempt to resolve the issue generically, a Mark II Owners Group was formed; the Zimmer applicant is a member of the group.

The initial effort of the group resulted in issuance of the Dynamic Forcing Function Information Report (DFFR). The information contained in DFFR either is preliminary or needs further verification. Consequently, there is some uncertainty in the information contained in the DFFR and it can only be resolved on a long-term basis. In order to meet the needs of the lead plants, that is, plants in the later stage of construction, the staff issued NUREG-0487 report entitled "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," dated October 1978, with Supplements 1 and 2 dated August 1980 and January 1981, respectively. In August 1981, the regulatory staff issued NUREG-0808 report entitled, "Mark II Containment Program Load Evaluation and Acceptance Criteria," in which the staff concludes that the improved condensation-oscillation and chugging loads for the suppression pool boundary as proposed by the Mark II Owner's Group and the lead plant pool-swell load adopted by the Mark II owners as the final load specifications are conservative.

On the basis of DFFR and NUREG-0487, the structural components forming the boundary of the suppression pool were evaluated by the applicant for their capability to resist the effects of the additional hydrodynamic loads and were found to have adequate margins of safety. Through the use of a finite element model with the inclusion of the water as fluid mass, the effect of fluid-structure interaction was considered in the evaluation. The evaluation is contained in the applicant's Design Assessment Report (DAR). The staff has reviewed the DAR and found that Amendment 17 of the DAR did not incorporate the criteria delineated in NUREG-0808. In view of this fact, the applicant has committed to make an in-plant SRV test and a confirmatory assessment of Zimmer 1 containment with respect to the effect of revisions to load definitions as delineated in NUREG-0808 as a confirmation of the adequacy of the evaluation as presented in the revised DAR. With this commitment, the staff considers this to be a confirmatory item.

3.8.2 Concrete and Structural Steel Internal Structures

The containment interior structures consist of sacrificial shield wall, radial beam framing and stabilizer truss in the drywell, and drywell floor and its support columns, reactor support, cable tray supports, and catwalks in the suppression pool.

The containment concrete and steel internal structures are designed to resist various combinations of dead and live loads, accident-induced loads, including pressure and jet loads, and seismic loads. The assessment of design adequacy of the containment internal structures to withstand the effects of suppression pool hydrodynamic loads was accomplished in the same manner as that for the containment structure as described in Section 3.8.1. The detailed reevaluation of the capability of containment internal structures to resist the newly identified loads is described in the applicants' revised Design Assessment Report. The staff has reviewed the design and analysis procedures and criteria that were used for the original design and for the reevaluation of the internal structures in the suppression pool. The containment internal structures were designed and proportioned to remain within limits described in the Zimmer 1 FSAR and accepted by the Regulatory staff under various load combinations. These limits as well as the design and analysis procedures are described in the staff's Zimmer 1 Safety Evaluation Report. The applicant indicated that structures or structural elements which were found not meeting the established criteria have been modified or strengthened.

As indicated in Section 3.8.1 of this supplement, the applicant is committed to perform an inplant SRV test and also a confirmatory assessment of the containment and its internal structures in conformance with the criteria delineated in NUREG-0808 in order to assure the design adequacy of these structures. With this commitment, the staff considers this to be a confirmatory item.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

The staff's evaluation of the adequacy of the applicant's program for qualification of electrical and mechanical equipment important to safety for seismic and dynamic loads consists of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (2) an onsite audit of selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

The Seismic Qualification Review Team (SQRT) has reviewed the equipment dynamic qualification information contained in the pertinent Final Safety Analysis Report (FSAR) Sections 3.9.2 and 3.10 and made a site visit on June 2 through June 5, 1981 to determine the extent to which the qualification of equipment as installed in Zimmer meets the current licensing criteria as described in IEEE 344-1975, Regulatory Guides 1.92 and 1.100, and the Standard Review Plan Sections 3.9.2 and 3.10. Conformance with these criteria satisfies the applicable portions of General Design Criteria in 1, 2, 4, 14, 18, and 30 of Appendix A to 10 CFR Part 50, as well as Appendix B to 10 CFR Part 50, and Appendix A to 10 CFR Part 100. A representative sample of Seismic Category I mechanical and electrical equipment, as well as instrumentation, including both NSSS and BOP scopes, were selected for the plant site review. The review consisted of field observations of the actual equipment configuration and its installation, followed by the review of the corresponding test and/or analysis documents.

In instances where components have been qualified by test or analysis to other than current licensing criteria such as IEEE Standard 344-1975, Regulatory Guides 1.92 and 1.100, and the Standard Review Plan Sections 3.9.2 and 3.10, or where equipment is affected by and was not qualified for the Mark II containment suppression pool hydrodynamic loads (associated with either safety relief valve discharge or LOCA blowdown into the suppression pool), the applicant has undertaken a reevaluation and requalification program.

In the trip report of the SQRT site visit (see Appendix H, Section 3.10, reference 1), the staff concluded that in order to complete its review, the staff would require the applicant to provide additional information and to clarify the details of the qualification for some pieces of equipment. In response to these concerns, the applicant provided post-audit submittals on July 21, August 27, 1981 as well as April 7, 1982. A number of concerns had since been resolved during several conference calls between the SQRT and the applicant. The staff's remaining concerns and the corresponding response contained in the above applicant's submittals are summarized below. (Sequence of items listed below is the same as that in reference 1.)

Generic Concerns

A. Provide results and conclusions for the following:

(a) In Plant Impedance Tests. The applicant has submitted the results and conclusions of their inplant impedance test program on 8 representative items of equipment important to safety. The objectives of this program are to assess the similarities and differences between the impedance test results and the existing qualification reports. The SQRT reviewed the summary report and concurs with the applicant's conclusion that, in general, a good agreement exists between the existing qualification reports and the results of the inplant impedance tests.

(b) Fatigue Evaluations. The concern was that the fatigue effect on equipment due to hydrodynamic loads was not adequately addressed. The applicant has examined some typical equipment under the combined seismic and suppression pool hydrodynamic loads. Three representative items of equipment (LPCS and RCIC pumps and RHR heat exchanger) were selected and evaluated by analysis. Results showed that the maximum total usage factors due to combined seismic and hydrodynamic loading for the LPCS, RCIC pumps and RHR heat exchanger are 0.221, 0.0055, and 0.373, respectively as compared to the maximum allowable limit of 1.0. In addition, 3 local instrument racks and 14 electrical instrumentation and devices (such as indicators, switches, transmitters, temperature elements, etc.) were evaluated by testing. The applicant has submitted to the SQRT on April 7, 1982 a brief outline of the procedures and underlying technical justification for their fatigue evaluation program, together with a typical fatigue evaluation report on Recirculating Core Isolation Cooling Pump (RCIC pump). Furthermore, the applicant has completed and provided results on April 7, 1982 of dynamic tests of Limitorque motor-operated valve assemblies, to demonstrate the adequacy of Limitorque motor operated valves to function during postulated safety relief valve actuation and LOCA events in addition to the seismic events. This was accomplished by testing one 16-inch gate valve with Limitorque SMB-2-40 actuator, and one 4-inch gate valve with Limitorque SMB-000 actuator, using a dynamic input which simulates the amplitude, frequency content, and duration for the most severe combination of seismic, SRV actuation, and LOCA loads. The review of the applicant's fatigue evaluation program is presently underway.

B. Provide justification regarding the amplification factor of 2 used for valve qualification:

The applicant committed to use justification regarding LaSalle Plant's valve amplification factor of 1.5 as an argument for justifying Zimmer's more conservative amplification factor of 2. The SQRT has reviewed the valve flexibility study submitted by La Salle plant on November 5, 1981, and found the amplification factor of 1.5 used by La Salle acceptable. It is thus concluded that the more conservative amplification factor of 2 used by Zimmer is also acceptable.

C. Limitorque motor operators were dynamically qualified for seismic loading only. Evidence of qualification of these operators to the additional hydrodynamic loadings (due to either the safety relief valve discharge or LOCA blowdown into the suppression pool) should be provided.

This concern has already been addressed in A.(b) above.

D. Piping analysis results should be checked when available to make sure the loading imposed by piping on all the valves do not exceed allowable g levels and nozzle loads.

In the applicant's April 7, 1982 submittal, it was stated that the piping analysis work on Zimmer is essentially complete. In addition, the applicant stated that wherever the valve allowable "g" levels and/or nozzle loads were exceeded, the valve was requalified for the higher allowables, or the piping loads were reduced by redesign of the support system. Any engineering changes to existing safety related piping will involve a review of the valve "g" levels and nozzle loads to assure that allowables are not exceeded. The SQRT regards this approach adopted by the applicant acceptable. However, the applicant is required to notify NRC in writing when this program is completed.

Specific Concerns

A. Provide clarifying details as described below:

(a) Spent Fuel Storage Rack (NSSS8). The applicant provided a report entitled, "Assessment Report - Spent Fuel Storage Rack," on April 7, 1982. The SQRT has reviewed this report and found there is insufficient documentation to arrive at a conclusion. Adequate documentation should be provided by the applicant in order for the SQRT to establish the qualification of this spent fuel storage rack.

(b) Reactor Core Cooling Bench Board (NSSS10). In the applicant's April 7, 1982 submittal, the list of the nonessential devices on the reactor core cooling bench board was provided, but their nonessentiality was not substantiated. A reference is made to a GE document (No. 328X2277U, Rev. 24) for Zimmer in this regard. The relevant part of this document should be supplied to establish non-essentiality of these devices.

(c) Flow-Indicator Switch (NSSS13). The applicant is committed to replace the Barton model 288 flow indicator switch by the qualified Barton model 288A by October 1, 1982. The applicant should notify the staff in writing when this replacement is completed.

(d) Bailey Alarm (NSSS15). The qualification of this item could not be established from the applicant's submittal of April 7, 1982. The applicant is required to demonstrate that side to side, non-restrained rack motion at its resonance frequency (18 Hz), and at a level which would be present at its current location (mounting location at Zimmer plant), the alarm would receive an excitation of less than 9.5 g.

B. Requalification of the following items is currently in progress. The applicant committed to requalify them by the following dates.

(a) 480-V Motor Control Center (BOP 1). Requalify by September 20, 1982.

(b) RBCCW Pumps (BOP 2). The applicant informed the staff that the engineering portion of the requalification has been completed and modification of hardware is currently underway. The applicant committed to provide a report of requalification of this item to the NRC by July 15, 1982.

Furthermore, the applicant indicated on June 16, 1982 that as of that date, essentially all NSSS and BOP equipment important to safety are qualified to the SQRT criteria. Approximately 90 to 95 percent of the equipment are qualified to the SQRT criteria; however, only about 70 percent of the BOP instrumentation is qualified to the SQRT criteria. The applicant stated that these equipment yet to be qualified or requalified will be qualified by October 1982 which is several months before the expected fuel loading date. The applicant committed to provide to the staff by July 2, 1982 a list of equipment yet to be qualified or requalified, together with a schedule for qualification or requalification.

The staff will complete its review when the applicant has provided the required information as stated above and has documented the completion of their seismic and dynamic qualification program. The staff will report on the results of its final evaluation of the applicant's program in a future supplement to the Safety Evaluation Report.

3.11 Environmental Design of Mechanical and Electrical Equipment

The applicant provided the staff with an environmental qualification report by letter dated July 16, 1982. Upon completion of the review of this information, the staff will conduct an audit and prepare the environmental qualification safety evaluation report. The staff considers this to be a confirmatory item.

4 REACTOR

4.2 Fuel System Design

4.2.3 Design Evaluation

Seismic and LOCA Loadings

No change in status from Supplement 2 to NUREG-0528.

4.4 Thermal and Hydraulic Design

4.4.1 Evaluation

4.4.1.1 Core Flow Monitoring

In the latter part of Section 4.4.1.1 in NUREG-0528, a sentence, "Technical Specifications will require that the core flow be checked every 24 hours and the average power range monitor flow biased scram be recalibrated every month," should be corrected to read "Technical Specifications will require that the core flow be checked every 24 hours." The staff reviewed the original TS and found that the channel calibrations for the Average Power Range Monitor (APRM) flow-biased scram specified in Item 2.6 of Table 4.3.1.1 are appropriate to account for possible effects of crud deposition. As specified in the original TS, full-channel calibration of the APRM flow-biased scram should be made on a semi-annual basis instead of a monthly basis. No further discussion of this topic is required in future supplements unless new information is identified.

4.4.1.2 Core Spray Distribution

Introduction

The staff recently reopened its review of core spray distribution based on information provided by the ACRS on spray distribution tests conducted in a foreign country for a simulated BWR/5 configuration in steam using a 60° sector test facility. The test data show that central bundles receive low core spray flow due to maldistribution. Although no specific data are available, the staff has also been told that the 360° tests with 5/6 of the spray nozzles blocked gave similar results to the 60° sector tests. This information is of concern since credit is taken for core spray heat transfer using a minimum spray to each bundle in the General Electric (GE) Emergency Core Cooling System (ECCS) Evaluation Model. This results in a heat transfer coefficient of 1.5 Btu/hr-ft^2-°F for core spray heat transfer, which is the minimum value specified in Appendix K.

Evaluation and Conclusions

The 60° sector test data are not the first to show low spray flow to some sections of BWR cores. Thus, as described above, the staff has previously considered the effect of low core spray flow to individual channels on calculated peak clad temperature (PCT). In our evaluation of NEDO-20566 Amendment 3, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K - Effect of Steam Environment on Core Spray Distribution," it was concluded that minimum spray flow to any channel following a LOCA would not be less than half of the design flow that was demonstrated to be available by tests and calculations. The tests and calculations did not include steam effects on nozzle spray patterns and flow rate. Based on measurements of minimum bundle spray flow for each BWR size and type for one sparger only, in air, the minimum flow for BWR/2 through BWR/5 designs was calculated to be 1.3 times the flow necessary to remove decay heat by vaporization (reference flow). Thus, the steam effects on spray distribution would not result in less than 0.65 times the minimum reference flow (or 1.3 times with both spray spargers operating). BWR FLECHT data (see Appendix H, Sec. 4.4, Reference 1) show little degradation in heat transfer for flow as low as 0.38 times the reference flow, or approximately 1 gpm. As far as we have been told, the minimum flow observed for any bundle in the 60° sector tests was 1 gpm. The heat transfer coefficients in GE's ECCS Evaluation Model are based on the FLECHT data, and a minimum bundle flow of 1 gpm would justify the heat transfer coefficient for core spray cooling (1.5 Btu/hr-ft^2-°F) used in that model.

During the BWR core spray injection, spray injected in the upper plenum will either be distributed to the core or bypass the core and drain to the lower plenum region, which results in a rapid bottom reflood rate. Presently credit was not taken for this rapid bottom reflood effect in the GE ECCS Model. Any liquid in excess of the minimum required for core spray heat transfer is assumed lost from the system and does not contribute to the reflood. Preliminary results from the 30° SSTF Counter Current Flow Limiting (CCFL) tests performed in Lynn, Massachusetts show that the spray flow injected in the upper plenum actually drains to peripheral bundles and increases the bottom reflood rate. In response to our request, GE presented result for reanalysis of limiting BWR/4 and BWR/5 cases to assess the effect of no core spray cooling on the peak clad temperature, assuming that the core spray coolant drains to the lower plenum and increases the reflood rate as observed in the Lynn tests. The calculated peak clad temperature did not exceed the 10 CFR 50.46 limit of 2200°F with no credit taken from the spray cooling effect.

The staff concludes that the new information from the 60° sector tests does not pose a safety concern for BWR/4 and BWR/5 reactors including Zimmer for the following reasons:

(a) Core spray flow maldistributions resulting in flows on the order of 1 gpm per bundle (apparently consistent with those obtained in the 60° sector tests) would remain consistent with the core spray cooling assumptions employed in the present GE ECCS Evaluation Model.

(b) New analyses performed by GE have shown that for limiting BWR/4 and BWR/5 cases with core spray assumed to flow down peripheral channels to increase the reflood rate as observed in the Lynn test, the calculated peak clad temperature did not exceed the 10 CFR 50.46 limit of 2200°F with no credit taken for the spray cooling effect.

4.6 Functional Design of Reactivity Control System

4.6.2 Control Rod System

Scram Discharge System

On December 22, 1980 the staff forwarded its December 1, 1980 Report, "BWR Scram Discharge System Safety Evaluation" to the applicant. This report addressed concerns raised by the partial scram event at the Browns Ferry Plant in 1980. This report contained the staff criteria relating to the scram discharge system. The staff further clarified these criteria in a letter dated March 30, 1981.

In Supplement 1 to the Staff Safety Evaluation Report (SER) for Zimmer Unit 1 (NUREG-0528, June 1981) it was stated that the staff would review conformance of the Zimmer scram discharge volume (SDV) design with the generic study presented in the December 1, 1980 report and provide the results of that review in a future supplement to the SER.

The applicant responded in a letter dated July 30, 1981, taking exception to the following criteria recommended in the December 22, 1980 report.

1. Safety Criterion 1
2. Safety Criterion 2
3. Safety Criterion 3
4. Safety Criterion 4
5. Design Criterion 3
6. Design Criterion 7
7. Design Criterion 10
8. Surveillance Criterion 3

The staff reviewed the applicant's response of July 30, 1981 and reported the results of that review (see Appendix H, Sec 4.6.2, Reference 1) on November 6, 1981. The staff concluded that the applicant's exceptions were not valid with the exception of the one involving Design Criterion 7. It was agreed that Design Criterion 7 does not apply to the Zimmer Plant Design.

In a letter dated April 8, 1982 (Appendix H, Sec. 4.6.2, Reference 2) the applicant withdrew its exceptions to all of the outstanding items which include Safety Criterion 1, 2, 3, and 4, Design Criteria 3 and 10, and Surveillance Criterion 3.

Therefore, the staff now finds the applicant's response to the December 1, 1980 NRC staff report, "BWR Scram Discharge System Safety Evaluation" to be acceptable. This item is considered to be resolved.

Scram System Pipe Break

NUREG-0803 provides guidance for an acceptable plant-specific resolution of the issues related to this concern. This guidance is grouped into three major areas:

(1) Piping Integrity - Licensees and applicants are to verify proper scram discharge valve (SDV) piping installation by as-built inspection and propose

an inservice inspection program of the SDV system which meets the requirements of ASME Section XI for Class 2 piping.

(2) Mitigation Capability - Licensees and applicants are to implement revised emergency procedures for pipe break in the scram system.

(3) Environmental Qualification - Licensees and applicants are to identify equipment needed to:

(a) detect an SDV system break, and

(b) mitigate the consequences of such a break and propose a program for qualifying such equipment (if not environmentally qualified).

In a letter dated July 26, 1982, the applicant provided an evaluation of this concern. The staff is reviewing this evaluation and will confirm in a future supplement to the Safety Evaluation Report that the concerns in NUREG-0803 are adequately addressed for the Zimmer plant.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.3 Reactor Coolant Pressure Boundary Materials

No change in status from Supplement 2 to NUREG-0528.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

Inservice Testing of Pumps and Valves

In Supplement 1 to the SER, the staff stated it had not completed detailed review of the applicant's submittal. Therefore, the staff could only grant relief from the ASME Code, Section XI, for an interim period based on preliminary review. Since that time the staff has completed a detailed review. Therefore, pursuant to 10 CFR Part 50.55a, the relief that the applicant has requested from the Code is granted for the initial 120 month inspection interval. The staff has concluded that granting these relief requests will not endanger the health and safety of the public.

However, during the detailed review, the staff identified certain valves which are not considered to be safety-related by the applicant and are not included in their inservice testing program. Specifically, these are gate valves which are part of the cross connection between the Service Water System (SW) and the Residual Heat Removal System (RHR) and check valves that are part of the hydraulic control units utilized for rod control and scram. The staff has determined that these valves are safety-related and requires that they be included in the applicant's inservice testing program and be tested as follows. For the cross connect gate valves, 1WS012A & B and 1WS013A & B, it was concluded that leakage testing is not required but they should be full stroke exercised at cold shutdowns and refueling outages.

The applicant has proposed to verify the operational readiness of the check valves in the hydraulic control units by scram tests which are performed quarterly in accordance with the Standard Technical Specifications (NUREG-123, Revision 3). The staff finds this acceptable provided that the applicant does not take exception to technical specification 4.1.3.2 of the Standard Technical Specifications.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 General

Mark II Containment LOCA Pool Dynamic Load

Discussion of Mark II Containment LOCA Pool Dynamic Loads appeared in Section 6.2.2 of NUREG-0528, Supplement 1; however, it is included in this section at this time in order to be consistent with the Standard Review Plan (NUREG-0800) and NUREG-0528.

The Zimmer Nuclear Power Station utilizes the Mark II type of containment. The adequacy of the Mark II containment for the Zimmer Station was reviewed against the staff's lead plant acceptance criteria as identified in NUREG-0487 and its Suppelements 1 and 2. In NUREG-0487, it was stated that at the conclusion o the Mark II Long Term Program (LTP) the applicants for all Mark II facilities would be required to conduct a confirmatory review of their containment against the long-term program load specifications. This confirmatory review for the Zimmer Station is limited to the chugging, condensation oscillation, vent lateral and diaphragm reverse pressure loads. The staff requested that the results of this review be submitted by the applicant within 1 year of its receipt of NUREG-0808 (a copy of which was forwarded to the applicant by letter dated September 24, 1981 from D. Eisenhut to E. A. Borgman).

In its letter of April 20, 1982, the applicant proposed to provide such an evaluation 6 months after completion of the SRV tests which will be after October 1, 1982. These tests are now scheduled to be conducted soon after fuel load.

The applicant's load specifications were previously reviewed and found to be conservative and acceptable as noted in the Safety Evaluation Report. Also, the finalization of the chugging load specification as set forth in NUREG-0808 was in doubt until recently. Based on these considerations, the staff agrees with an extension beyond 1 year from the date NUREG-0808 was issued as specified in the letter from D. Eisenhut to Mr. Borgmann, dated September 24, 1981 to assess the containment against NUREG-0808 acceptance criteria. However, the staff does not believe it is necessary to tie this commitment to the SRV test reporting schedule.

Therefore, the upcoming operating license for the Zimmer Nuclear Station will be conditioned on the applicant's furnishing the results of its confirmatory assessment of Zimmer Station's containment against the specifications for pool dynamic loads (chugging, condensation oscillation, vent lateral, and diaphragm reverse pressure) developed in conjunction with the LTP and reported in NUREG-0808, within 9 months from the fuel load date instead of the previously established date of prior to October 1, 1982.

Wetwell-to-Drywell Vacuum Breaker Performance During the Pool Swell Phase of a LOCA

This additional issue has been developed since issuance of SER Supplement 2 and will require staff evaluation prior to the fuel load date for Zimmer 1. The Zimmer Atomic Safety and Licensing Board has been notified of a new concern regarding vacuum breaker performance under accident conditions. The staff plans to complete its evaluation of this open item prior to the Zimmer 1 fuel load date.

Adequacy of the BWR Pressure Suppression Containment Systems

Several concerns have been identified by a former employee of General Electric Company regarding the Mark III containment design capability. The staff is currently assessing the applicability of the concerns to the Mark II containments. The applicant has been requested to evaluate the applicability of the concerns to the Zimmer plant. This open item will be discussed further in a future supplement to the Safety Evaluation Report.

6.2.3 Containment Isolation System

Control Rod Drive Insert and Withdrawal Lines

The following discussion has been included to state the staff rationale for finding acceptable the containment isolation provisions for the control rod drives (CRD) in the Zimmer plant. The design represents a departure from the explicit requirements of the GDC.

Four types of valves associated with the CRDs are discussed below: control valves, scram valves, manual shutoff valves, and ball check valves.

Both the CRD insert and withdrawal lines are provided with normally closed, fail-closed, solenoid-operated directional control valves, which open only during routine movement of their associated control rod. The normally closed, fail-open air-operated scram inlet and exhaust valves open only when required to effect a rapid reactor shutdown (scram). In addition, manual shutoff valves are provided for positive isolation in the unlikely event of a pipe break within a hydraulic control unit. (These units and the valves described above are located outside containment to satisfy testing, inspection, and maintenance requirements.) In addition, each CRD insert line is provided with an automatically actuated ball check valve inside containment. We find that the system design represents a departure from the explicit requirements of general design criteria (GDC). However, in accordance with the provisions of Appendix A to 10 CFR Part 50 and GDC 55 which permits departure from its explicit requirement, the staff finds that the CRD containment isolation provision stated above is acceptable on the basis stated in NUREG-0803, "Safety Evaluation Report Regarding Integrity of BWR Scram System," dated August 1981. The implementation of NUREG-0803 is a staff requirement that is described in more detail in Section 4.6.2 (scram system pipe break) of this supplement.

6.2.6 Containment Leakage Testing

The applicant has reviewed its procedures for leak testing the ECCS Injection Line Isolation Valves and has committed to air testing these valves by pressurizing the bonnets of the valves. Based on this commitment, the staff concludes that Zimmer is now in conformance with Appendix J to 10 CFR Part 50 as regards the leak testing of these valves. The staff requires that Table 6.2-8 of the FSAR be revised accordingly.

6.3 Emergency Core Cooling Systems

6.3.4 Performance Evaluation

Scram System Pipe Break

Discussion of this concern is in Section 4.6.2 of this supplement.

6.4 Habitability Systems

6.4.2 Toxic Gas Protection

See Section 2.2.1 of this supplement for the status on this matter.

7 INSTRUMENTATION AND CONTROLS

7.1 General Information

7.1.3 Specific Findings

Physical Separation of Associated Circuits

The applicant has completed a 100% analysis of the physical separation of associated circuits. The purpose of this analysis was to verify the design criteria and the implementation for divisional associated circuits. The acceptance criteria for the analysis was no single event coupled with a single failure would result in the loss of a safety function. The analysis identified limited areas which required a field check to assure that the design criteria had been properly implemented.

Deviations identified during the field check will be corrected. In addition, the applicant has identified and provided justification in FSAR Revision 83 for any differences between their design for the independence of electrical systems and that currently recommended in Regulatory Guide 1.75.

The staff concludes that the results of the analysis, justification provided, and actions taken provide reasonable assurance that adequate independence of electrical systems is provided for the Zimmer-1 facility.

7.5 Safety Related Display Instrumentation

7.5.3 Specific Findings

Loss of Power to Instruments and Control Systems

The staff requested that the applicant review the adequacy of emergency operating procedures to be used by control room operators to attain safe shutdown upon loss of any class 1E or non-class 1E buses supplying power to safety or nonsafety-related instruments and to control systems (This issue was addressed for operating reactors through I&E Bulletin 79-27). The applicant has demonstrated that sufficient equipment for safe shutdown would remain available subsequent to loss of any 1E or non-1E electrical bus. The staff also requested that the adequacy of the existing emergency operating procedures for achieving cold shutdown be assessed assuming the loss of an electrical bus and its effect on safety or nonsafety instruments and control systems.

The applicant identified in FSAR Revision 83 the operating procedures, power sources, support systems, and operator information necessary to achieve cold shutdown using current (as-built) drawings and documents. Utilizing the information obtained, a single failure analysis was performed. The results of the analysis indicate the minimum equipment required to obtain a cold shutdown condition remains functional and the existing operating procedures adequately guide the operator in performing the necessary actions. The staff, therefore, considers this issue resolved.

7.7 Control Systems Not Required For Safety

7.7.3 Specific Findings

Control System Failures and Effects of High Energy Line Breaks

The analyses reported in Chapter 15 of the Final Safety Analysis Report are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents, including those related to control systems. Based on the conservative assumptions made in defining these "design bases" events and the detailed review of the analyses by the staff, it is likely that they adequately bound the consequences of single control system failures including those which result from the harsh environments associated with high energy line breaks (HELB).

The applicant has performed a Failure Mode Effect and Consequences Analysis (FMECA) of the plant control systems for common sensors or common power source failures and HELB which might cause multiple control system malfunctions and result in consequences more severe than those of Chapter 15 analyses or beyond the capability of operators or safety systems.

The applicant utilized current (as-built) drawings, documents and physical plant walkdowns to identify the control system equipment in the vicinity of potential HELBs and the sensors or power sources common to two or more control systems. The results of the FMECA indicates the FSAR Chapter 15 Analyses bound the consequences of control system actions caused by the failure of common sensors, power sources or the effects of an HELB. However, the applicant has not finalized that portion of the FMECA which considers the effects of malfunctions in common hydraulic or impulse lines feeding pressure, temperature, level or other signals to two or more control systems. The preliminary results of this portion of the analysis indicate the effects are also bounded by the FSAR Chapter 15 Analysis.

The staff will report on this confirmatory item in a future supplement to NUREG-0528.

8 ELECTRIC POWER

8.1 Introduction

8.1.2 General Findings

Degraded Grid Voltage

The applicant has reanalyzed the voltage levels on the Class 1E buses, and in Amendment 127, Revision 76 to the Final Safety Analysis Report, dated August 1981 provided the results of his analysis. The relay trip setpoint for the required second level undervoltage protection will be increased to approximately 3800 volts. In addition, the applicant has changed the time interval for actuation of the alarm on the second level undervoltage relay from zero seconds to time interval greater than a motor starting transient (variable range of 0 to 20 seconds) and trip offsite sources in 5 minutes.

The proposed second level undervoltage scheme and relay setpoints will protect the Class 1E equipment from operation under sustained degraded grid voltage conditions within the expected range of grid voltage limits. The proposed modifications conform to part (a) of position 1 on degraded grid voltage. We find this acceptable and therefore consider this item resolved.

Because of the long delivery time to the equipment, the applicant states that this modification will not be implemented until the first refueling outage. We find this acceptable. We will condition the operating license upon the satisfactory implementation of this modification per design, prior to restart following the first refueling outage.

Protection of Reactor Containment Electrical Penetrations

The applicant in Amendment 127, Revision 76 to the Final Safety Analysis Report, dated August 1981 provided the maximum fault current versus time profiles and time current characteristics (I^2Rt) of the penetration conductors. In addition, the applicant has submitted drawings showing proposed designs for the protection of electrical penetrations.

Except for the 6.9-kV reactor coolant pumps penetrations, all other protective devices will be fault current actuated and will require no external power source. This includes the 480-volt drawout case type breaker which receives tripping power from the fault via current transformers.

The 6.9-kV penetrations are the only items where control power for primary and backup protection devices will be provided from independent Class 1E DC sources, i.e., powered from DC buses 1D and 1E. This conforms with Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants," and is acceptable. We consider this item closed.

We will require the applicant, by a condition on the license to satisfactorily implement the proposed modifications for the protection of electrical penetrations as stated above, prior to restart following the first refueling outage.

Station Blackout Events

The applicant is proceeding with response to generic letter 81-04, "Emergency Procedures and Training for Station Blackout Events," and has reported the status in Amendment 127, Revision 76, to the Final Safety Analysis Report. The staff requires that interim emergency procedures and operator training for blackout events be implemented prior to fuel loading. This is discussed further in Appendix C of NUREG-0528, Supplement 1. The staff will confirm that emergency procedures and operator training have been implemented prior to fuel load.

In Supplement 2, "other requirements which may result from the staff's review" were stated to be a condition of licensing. However, the license condition is being eliminated because no specific "other requirements," have yct been identified and because any such requirements will be dealt with generically as part of the staff's Unresolved Safety Issue Program.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.4 Fuel Handling System

Control of Heavy Loads

In Supplement No. 2 to NUREG-0528 the staff stated that the applicant had committed to implement the interim actions of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," as required by the staff's general letter dated December 22, 1980. The staff concluded that implementation of the interim actions prior to the final implementation of NUREG-0612 guidelines and prior to the receipt of their operating license provided reasonable assurance of safe handling of heavy loads until full implementation of NUREG-0612 would be possible. This item was listed as an operating license condition in Section 1.12 of Supplement No. 2. In a letter from B. R. Sylvia to H. Denton, dated July 9, 1982, Cincinnati Gas and Electric committed to provide documentation by October 1982 indicating that the interim actions defined in the staff's December 22, 1980 letter are complete. Upon receipt of acceptable documentation, the need for a license condition on control of heavy loads will be deleted by the staff in the next SER supplement.

9.6 Diesel Generator Systems

In Amendment 127, Revision 76, to the Final Safety Analysis Report, the applicant reported the status of implementation of the guidelines presented in NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." Full implementation of the staff's position is required prior to startup following the first refueling outage and will be made a condition of the operating license.

13 CONDUCT OF OPERATIONS

13.7 Industrial Security

Physical Security Plan

The staff has reviewed the applicant's Physical Security Plan and provides the following approval.

The applicant has submitted security plans entitled "Wm. H. Zimmer Nuclear Power Station, Unit 1 Industrial Security Plan," "Zimmer Nuclear Power Station Safeguards Contingency Plan," and "Zimmer Nuclear Power Station Security Training and Qualification Plan," for protection against radiological sabotage. The plans were reviewed in accordance with Section 13.6 "Physical Security" of the July 1981 edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." (SRP, NUREG-0800)

As a result of the staff evaluation, certain portions of these plans were identified as requiring additional information and upgrading to satisfy the requirements of Section 73.55 and Appendices B and C of 10 CFR Part 73.

The applicant filed revisions to these plans which satisfied these requirements. The revised plans are considered to comply with the Commission's regulations contained in 10 CFR Part 73.

An ongoing review of the progress of the implementation of these plans is performed by the staff to assure conformance with the performance requirements of 10 CFR Part 73.

The identification of vital areas and measures used to control access to these areas, as described in the plan, may be subject to amendments in the future.

The applicant's security plans are being protected from unauthorized disclosure in accordance with Section 73.21 of 10 CFR 73.

The staff will condition the license as follows. The licensee shall fully implement and maintain in effect all provisions of the Commission approved physical security, guard training and qualification, and safeguards contingency plans, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans, which contain information described in 10 CFR 73.21, are collectively entitled "Wm. H. Zimmer Nuclear Power Station - Unit 1 -Amendment 118 - Revision 12 to the Industrial Security Plan, dated April 28, 1981," 'Zimmer Nuclear Power Station Safeguards Contingency Plan Appendix E, dated March 23, 1979 and revised by letters dated March 13, 1981 and May 27, 1981," and "Zimmer Nuclear Station Security Training and Qualification Plan dated March 13, 1982 and revised by letters dated May 27, 1981 and July 2, 1981.

22 TMI-2 REQUIREMENTS

22.2 TMI Action Plan Requirements for Applicants for Operating Licenses

I.C.5 Procedures for Feedback of Operating Experience to Plant Staff

The applicant reported in Amendment 127, Revision 76, to the Initial Safety Analysis Report that development of procedures for feedback of operating experience to plant staff are under development pending final reorganization of the corporate structure for the nuclear facility. It is anticipated that these procedures will be completed in 1982. The staff will review and report on these procedures when they are completed.

II.E.4.1 Dedicated Hydrogen Penetrations

Discussion and Conclusions

The Zimmer Nuclear Power Station presently has an external recombiner in its Flammability Control System (FCS), which takes suction from the drywell through a dedicated penetration and isolation system, and meets all redundancy and single failure requirements. Furthermore, it is not connected to, nor is it a branch line of, the large containment purge penetration. This system also discharges into the wetwell through safety grade piping and valves. In addition to the FCS, Zimmer has three other systems available to mitigate the consequences of hydrogen in the primary containment. They are the Standby Gas Treatment System; the Primary Containment Purge System and the Nitrogen Inerting System. Therefore, the staff concludes that Zimmer complies with the provisions of Item II.E.4.1 of the TMI Action Plan.

II.F.1 Additional Accident-Monitoring Instrumentation

Attachment 3, Containment High-Range Radiation Monitor

In Supplement 1 of the SER, the staff stated that the applicant should provide methods to correlate the readings from high-range containment radiation monitors located in the primary containment penetrations to actual dose rates inside containment. In Revision 83 (April 1982) to the FSAR, the applicant amended the response to Item II.F.1 (see Section 22 of Supplement to NUREG-0528). This revised response includes a correction curve to be used to correlate the high-range containment monitor readings (in R/hr) with the actual exposure rate (in R/hr) in the drywell for various times after an accident. Such a curve is necessary to correct for the attenuation due to the sleeves in which the monitors are located. Appendix B of the Applicant's Emergency Plan contains a graph that relates containment monitor readings (in Rads/hr) to activity of ionizing krypton and xenon in containment. Use of these correction curves, in conjunction with grab sampling and onsite isotopic analysis capabilities will permit the applicant to more accurately identify the actual containment radiation levels following an accident. The applicant's response meets the positions set forth in NUREG-0737 for Item II.F.1.

II.F.2 Instrumentation for Detection of Inadequate Core Cooling

Representatives from The BWR Owners Group met with the staff on December 17, 1981, January 27, 1982, and April 5, 1982 to discuss the staff requirements as specified in SECY 81-582 (October 7, 1981) and the BWR Owners Group position. As a result of these meetings, agreement was reached to broaden the issue from the specific requirements for incore thermocouples to that of monitoring inadequate core cooling. The BWR Owners Group agreed to actively participate in the analysis of inadequate core cooling instrumentation requirements and expects to submit a final report for staff review in late 1982. The staff expects the calculations reached from that evaluation will be applied to the Zimmer plant.

The staff concludes that the existing level instrumentation and the BWR Owners' Group emergency operating procedure guidelines (which will be based on the inadequate core cooling instrumentation requirements, to be developed and implemented upon completion of the staff review and approval of the applicant's report) will satisfy the requirements of Item II.F.2. Accordingly, the staff will condition the Zimmer operating license to require conformance to II.F.2 requirements which result from staff evaluation of the BWR Owners Group report.

II.K.1 IE Bulletins on Measures to Mitigate Small-Break LOCAs and Loss-of-Feedwater Accidents

No change in status from Supplement 2 to NUREG-0528.

II.K.3.27 Provide Common Reference Level for Vessel Level Instrumentation

In its letters of August 31, 1981 and April 16, 1982, transmitting revisions 76 and 83 to the FSAR, the applicant committed to implement the following modifications prior to loading fuel:

(1) The scales on level indicator 1B21-R610 and recorder 1B21-R615 will be changed to use the wide and narrow range instrument zero as a reference level. This instrument zero is located at the bottom of the steam dryer skirt (vessel elevation 516.75 inches). The new scales will have a range of -110 inches to -310 inches.

(2) Operating procedures will be modified to incorporate the above changes.

(3) Appropriate retraining of operators will be provided.

The staff finds that the applicant's proposed midifications and implementation schedule are acceptable and meet the requirement of TMI task Item II.K.3.27. This confirmatory item is now resolved. The implementation of the modifications will be verified by the NRC prior to fuel load.

III. Emergency Preparations and Radiation Protection

III.A.1.1 Upgrade Emergency Preparedness

No change in status from Supplement 2 to NUREG-0528

III.A.2 Improving Licensee Emergency Preparedness - Long Term

On January 22, 1982, the staff transmitted a list of all the outstanding items that must be accomplished before a favorable finding can be made on the status of emergency preparedness at the Zimmer Nuclear Power Station. This list was based on a review conducted in accordance with Section 13.3, Emergency Planning, of the July 1981 edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP, NUREG-0800). The major outstanding items included the following:

(1) Submission of the final State and local emergency response plans in accordance with 10 CFR 50.33.

(2) NRC staff receipt and review of FEMA findings and determinations as to whether the State and local emergency response plans are adequate.

(3) Correction of the deficiencies identified in the November 22, 1981, and January 22, 1982, letters to the applicant to ensure conformance with the standards in 10 CFR 50.47(b).

(4) Completion by the NRC of an onsite appraisal and correction of any deficiencies that indicate the applicant's plan cannot be implemented.

(5) Receipt of supporting information for the evacuation time estimate study.

By a letter dated February 12, 1982, the applicant provided additional information regarding the evacuation time estimates. This information adequately addressed item five of the above listed outstanding issues. By a letter dated April 19, 1982, the applicant provided a response to most of the remaining outstanding items. However, these responses must be incorporated into the site emergency plan before the staff review can be completed. The applicant has indicated that a revised plan incorporating these revisions will be provided to the NRC in August 1982.

On June 21, 1982, the Atomic Safety and Licensing Board issued an Initial Decision which, among other issues, concluded that the state of offsite emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. In particular, the Board found shortcomings in the planning for evacuation of local schools. The Board also stated that prior to operation of the Station at power levels in excess of 5% of rated power, the following deficiencies must be satisfactorily resolved: (1) the availability and responsibility of volunteers, (2) the transport of disabled individuals, (3) inadequacies in radio communications. amd (4) revisions and issuance of public educational material.

The staff is continuing to work with the applicant and FEMA toward final review and resolution of these matters and will report its conclusion in a future supplement to NUREG-0528.

III.D.3.4 Control Room Habitability

See Subsection 2.2.1 of this supplement for the status of final resolution of this matter.

23 CONCLUSIONS

Based on the staff's evaluation of the application as set forth in NUREG-0528 and in its supplements (Supplements 1 through 3), the staff is able to affirm the conclusions presented in Section 22 of NUREG-0528.

APPENDIX A

CHRONOLOGY OF NRC STAFF RADIOLOGICAL REVIEW OF WILLIAM H. ZIMMER NUCLEAR POWER STATION, UNIT NO. 1

Supplement No. 2 to the Safety Evaluation Report provided a chronology of the NRC staff's radiological safety review for the period for March 30, 1981 to September 24, 1981 in Appendix A. The purpose of this appendix is to update that chronology through July 8, 1982.

September 25, 1981	Letter from applicant transmits Amendment 128 to the FSAR which consists of Revision 77.
October 1, 1981	Letter from applicant concerning Safety/Relief Valve (S/RV) Operability Test Report.
October 7, 1981	Letter from applicant transmitting a certificate of service for Amendment 128 to the FSAR.
October 16, 1981	Letter from applicant concerning Fission Gas Release Analysis.
October 26, 1981	Letter to applicant requesting additional information on fuel handling systems.
October 30, 1981	Representatives from NRC & CG & ECO meet in Bethesda, MD to discuss environmental qualification review and resolve outstanding issues. (Summary issued November 3, 1981.
October 30, 1981	Letter from applicant transmitting Amendment 129 - Submittal of Revision 78 to the FSAR.
October 30, 1981	Letter to applicant transmitting 2 copies of Supplement 2 to the Zimmer SER.
November 2, 1981	letter from applicant concerning compliance with code of federal regulations.
November 9, 1981	Letter to applicant transmitting 20 copies of the bound contractor printed Supplement 2 to the Zimmer SER.
November 12, 1981	Letter to applicant requesting additional information on the Zimmer Emergency Plan.
November 13, 1981	Letter from applicant transmitting Amendment No. 130 to the applicant consisting of Revision 5 to the ER-OL.
November 17, 1981	Letter from applicant concerning containment leakage testing.

November 19, 1981	Letter from applicant transmitting a certificate of service for Amendment No. 130.
November 30, 1981	Letter from applicant transmitting Amendment No. 131 consisting of Revision 79 to the FSAR.
December 3, 1981	Letter from applicant transmitting the certificate of service for Amendment No. 131 to the FSAR.
December 11, 1981	Letter from applicant concerning long term operability of deep draft pumps.
December 18, 1981	Letter to applicant transmitting an Order extending the latest construction completion date to December 31, 1982.
December 15, 1981	Letter from applicant concerning Pipe Breaks in the BWR Scram System (Generic Letter 81-34).
December 28, 1981	Letter to applicant concerning seismic qualification program review.
January 4, 1982	Letter to applicant concerning Final Safety Evaluation Supplement for Zimmer Operating License Review.
January 6, 1982	Letter from applicant concerning supplemental information on toxic chemicals.
January 8, 1982	Letter from applicant transmitting Amendment 132 consisting of Reviesion 80 to the FSAR.
January 13, 1982	Letter from applicant transmitting a certificate of service for Amendment 132, in the form of Revision 80 to the FSAR.
January 19, 1982	Representatives from NRC, CG&E meet in Bethesda, MD to discuss emergency planning issues relative to meteorology. (Summary issed March 23, 1982.)
January 22, 1982	Letter to applicant concerning status of the emergency preparedness review for Zimmer.
February 2, 1982	Letter to applicant concerning Evacuation Time Estimates.
February 4, 1982	Letter to applicant concerning Interlocks Dependent on Pressure Signals for Valves Connecting the Low Pressure ECCS with the Reactor Coolant System.
February 5, 1982	Letter from applicant transmitting Amendment 133 consisting of Revision 81 to the FSAR.
February 9, 1982	Letter from applicant transmitting a certificate of service for Amendment 133, in the form of Revision 81 to the FSAR.

February 11, 1982	ORDER Requiring NRC Staff Testimony with Regard to Off-site Emergency Planning Matters Under NRC Staff Review.
February 16, 1982	Letter from applicant transmitting Amendment 17 to the Mark II Containment Design Assessment Report.
February 26, 1982	Representatives from CG&E and NRC meet in Bethesda,MD to discuss emergency planning. (Summary issued March 4, 1982.)
March 8, 1982	Letter from applicant giving fuel load date.
March 29, 1982	Letter from applicant concerning Mark II containment program.
March 31, 1982	Letter from applicant submitting Revision 82 consisting of Amendment 134 to FSAR.
April 7, 1982	Letter to applicant regarding "fast scram" hydrodynamic loads on control rod drive systems.
April 8, 1982	Letter from applicant concerning BWR scram discharge system safety evaluation.
April 16, 1982	Letter from applicant submitting revision 83 consisting of Amendment 135 to FSAR.
April 21, 1982	Letter from applicant enclosing certificate of service of Amendment 135.
April 29, 1982	Letter to applicant regarding control of heavy loads.
May 3, 1982	Letter from applicant transmitting the 1981 financial statements.
May 5, 1982	Letter to applicant regarding errors in BWR water level indication.
May 28, 1982	Letter from applicant transmitting Amendment 136 consisting of Revision 84 to the FSAR.
May 28, 1982	Letter from applicant concerning fast scram hydrodynamic loads on control rod drive systems.
June 2, 1982	Letter from applicant concerning containment leakage testing.
June 3, 1982	Letter from applicant transmitting a certificate of Service for Amendment 136.
June 23, 1982	Letter to applicant concerning evaluation of offsite emergency preparedness for the Zimmer Station exercise.

July 8, 1982	Letter to applicant - concerns regarding the adequacy of the design margins of the Mark I and II containment systems.

APPENDIX H

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3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electric Equipment

1. T. Y. Change (NRC) memorandum for Z. R. Rosztoczy (NRC), "Trip Report Seismic Criteria Implementation Review Meeting with Cincinnati Gas and Electric Company on Wm. H. Zimmer Nuclear Power Station," October 1, 1981.

4.4 Thermal and Hydraulic Design

1. F. A. Schraub and J. E. Leonard, APED-5529, "Core spray and Core Flooding Heat Transfer Effectiveness in a Full-Scale Boiling Water Reactor Bundle," June 1979.

4.6 Functional Design of Reactivity Control System

1. L. S. Rubenstein (NRC) memorandum for R. L. Tedesco (NRC), "Evaluation of Applicants Response to the December 1, 1980 NRC Staff Report, 'BWR Scram Discharge System Safety Evaluation,'" November 6, 1981.

2. E. A. Borgmann (CG&E) letter to H. Denton (NRC), "RE: Wm. H. Zimmer Nuclear Power Station, Unit 1 BWR Scram Discharge System Safety Evaluation," April 8, 1982.

APPENDIX I

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This Safety Evaluation Report is a product of the NRC staff and their consultants. The following NRC staff members were principal contributors to this report. A list of consultants follows the list of staff.

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16. ABSTRACT (200 words or less)

The Safety Evaluation Report for the Zimmer Nuclear Power Station, Unit 1 was issued in January 1979. At the time of issuance there were two outstanding issues. Supplement No. 1, issued in June 1981 discussed the resolution of these issues and the concerns of the Advisory Committee on Reactor Safeguards, which issued a favorable report on March 13, 1979. Supplement No. 2, issued in October 1981 discussed subsequent outstanding issues since June 1981. This Supplement closes out outstanding issues and concludes that the facility can be operated by the applicant without endangering the health and safety of the public. The Zimmer Station is located in Washington Township, Clermont County, Ohio.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT: UNLIMITED

19. SECURITY CLASS (This report): UNCLASSIFIED

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